Doc. No. 12-9051191-001

NGNP with Hydrogen Production

Preconceptual Design Studies Report

June 2007

Disclaimer

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, nor any of their employees, nor their contractors and subcontractors, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

BEA Contract No. 000 60209

AREVA NP Inc.

AREVA

Record of Revisions

Revision	Date	Pages/Sections Changed	Brief Description
000	6/05/2007	None.	Original Issue
001	6/22/2007	Global	Changed revision number in headings; minor editorial changes, spelling corrections and format adjustments throughout document.
001	6/22/2007	Section 1.1, first ¶	Corrected sentence fragment in first ¶.
001	6/22/2007	Section 2.6	Added Appendix E to list of appendices.
001	6/22/2007	Section 6.1.3.1, last ¶	Added statement on expected control rod coolant channel temperatures during normal operation. Revised expected post-accident control rod temperature (from 1400ºC to 1600ºC).
001	6/22/2007	Section 6.2.1.2	Added sentence stating expected margin to allowable stress for Mod 9 Cr 1 Mo material.
001	6/22/2007	Section 8.6.3.2, Page 133	Clarified definition of efficiency for FOAK and NOAK plants.
001	6/22/2007	Section 9.2.2, 2nd to last ¶	Corrected the capacity of the bridge crane to 300 tonnes.
001	6/22/2007	Section 12, Page 191	Added sentence clarifying ultimate goal of NGNP licensing strategy
001	6/22/2007	Section 12.2, Page 196	Deleted reference to a non-existent appendix
001	6/22/2007	Section 12.9, Page 205	Revised 2nd bullet to indicate that the initial NGNP license will be a non-LWR prototype license in lieu of a Class 104 (c) license.
001	6/22/2007	Section 12.9, Page 205	Added sentence to last ¶ stating that NGNP experience will support future licensing of HTR technology (i.e., design certification, license applications).
001	6/22/2007	Section 18	Added new subsection (Section 18.2.3) that discusses overall project risk.
001	6/22/2007	Section 22.4	Added bullet identifying nitriding as a key risk.
001	6/22/2007	Section 22.4	Added a final ¶ on managing project risk.
001	6/22/2007	Section 22.7	Added risk perspective discussion to second ¶.

Table of Contents

Table of Contents
(continued)

Table of Contents
(continued)

Table of Contents
(continued)

Table of Contents
(continued)

Table of Contents (continued)

Table of Contents
(continued)

Table of Contents
(continued)

Table of Contents
(continued)

Table of Contents
(continued)

Table of Contents
(continued)

List of Tables

List of Tables
(continued)

Page

List of Figures

List of Figures (continued)

Page

List of Figures (continued)

Page

1.0 SUMMARY

The AREVA Next Generation Nuclear Plant with Hydrogen Production Preconceptual Design Studies Report documents the preconceptual design engineering studies performed by the AREVA NGNP team for the Battelle Energy Alliance (BEA), the Management & Operating Contractor of the Idaho National Laboratory (INL) as part of the Department of Energy's Next Generation Nuclear Plant (NGNP) project.

1.1 Scope of Work

The Scope of Work assigned to the AREVA NGNP Team consisted primarily of the preparation of a preconceptual design studies report based upon the adaptation of AREVA's ANTARES HTR and the preparation of supporting technical documents (i.e., special studies).

The key elements of the AREVA scope of work are:

- Four special studies:
  - Reactor Type Selection Study
  - Prototype Power Level Study
  - Power Conversion System Study
  - Primary and Secondary Cycle Concept Study
- Development of an NGNP preconceptual design concept based on the adaptation of AREVA's ANTARES HTR concept to the NGNP requirements. This constrains the resulting NGNP concept to
  - Prismatic core
  - Indirect Cycle
  - Combined Cycle Gas Turbine (CCGT) electricity generating cycle
- Identify project R&D needs, risks, and overall schedule
- Develop project cost estimate
- Economic analysis for a commercial NOAK plant
- Document results in the PCDSR

Of the above special studies, only the results of the reactor power level and primary-secondary systems study influence the adapted preconceptual design reported herein. The direction to adapt ANTARES to the NGNP made the results of the reactor type and power conversion system moot relative to the preconceptual design reported herein; nevertheless, they were performed due to the valuable insights they would provide BEA/INL for the NGNP conceptual design.

1.2 Special Studies

AREVA was commissioned to perform the following special studies to either directly guide the design adaptation or provided valuable insight to BEA/INL on NGNP project direction.

1. Reactor Type Comparison Study – This study answers the questions, "Which HTR technology, pebble bed reactor or prismatic block reactor, best meets the needs of the NGNP." The AREVA study concluded that the prismatic block HTR was the preferred option in order to maximize the unit power level for best economic performance.

2. Prototype Power Level Study – This study answered three main questions: (1) What should be the power level of the commercial HTR plant? (2) What should the power level of the NGNP reactor be in order to adequately demonstrate the commercial reactor? And, (3) how much power should be devoted to a co-located demonstration hydrogen production facility? The AREVA study concluded that the power level of the commercial HTR plant is only constrained by the maximum achievable safe power level (i.e., 565 MWth). Also, the study tentatively concluded that the NGNP prototype should be full size (i.e., 565 MWth) in order to maximize the benefit to future commercial plants. This study also concluded that 60 MWth should be provided to the test loop supporting the hydrogen process demonstration facility.

3. Power Conversion System Study – This study reviewed current and proposed power conversion technologies to determine which technology should be coupled to the NGNP reactor. The AREVA study concluded that a steam cycle system was preferred for near-term deployment. The study also concluded that supercritical CO_2 systems may have significant potential advantages and should continue to be pursued for long-term deployment.

4. Primary and Secondary Cycle Concept Study – The purpose of this study was to determine the key operating parameters and equipment configuration for the NGNP. The AREVA study concluded that the reference NGNP should have reactor inlet/outlet temperatures of 500°C /900°C as the best compromise between hydrogen production performance and NHS feasibility. The study also concluded that a multiple loop primary circuit configuration using robust tubular IHXs for the main energy transfer to the PCS should be used to maximize feasibility and minimize risk.

1.3 NGNP Design Adaptation from ANTARES

The AREVA NGNP preconceptual design work scope required that AREVA develop an NGNP design concept adapted directly from AREVA's existing ANTARES HTR concept employing an indirect cycle combined cycle gas turbine (CCGT) energy utilization system.

AREVA's scope did not include either the hydrogen production facility or the high temperature heat transport loop. Therefore, the adapted NGNP design retains most of the ANTARES configuration with the exception of the heat transfer loop arrangement (i.e., multiple loops versus single loop).

The resulting reference NGNP design is a 565 MWth prismatic HTR with a modified 9Cr-1Mo reactor vessel. Heat is supplied to three parallel loops each with a helical coil tubular intermediate heat exchanger (IHX) and a dedicated primary circulator. Tubular IHXs are used to maximize design feasibility and component lifetime for very high temperature service. The secondary coolant from the three IHXs is combined to drive a single closed loop gas turbine. The secondary coolant is a mixture of nitrogen and helium selected to allow the use of air-breathing gas turbine technology. Residual heat from the turbine outlet drives the Rankine bottoming cycle through a heat recovery steam generator.

A fourth primary loop is included to provide heat for demonstration of high temperature hydrogen processes. Given the smaller size of this loop and the use of helium as the secondary fluid in the heat transfer loop, a compact heat exchanger is specified for this loop in order to demonstrate that new technology. The resulting configuration is the best configuration that can be achieved in the near-term for direct high temperature heat supply. It minimizes technical risk to the maximum extent possible.

The adapted NGNP design represents an advanced, near optimum multi-purpose configuration for the supply of high temperature process heat. Attendant with this advanced design are technological challenges, especially with respect to the development of IHX technology. Unless direct high temperature heat supply is the sole objective the NGNP regardless of the technical challenges and potential schedule impact, AREVA, based on its ANTARES experience, suggests an alternative configuration. As concluded in AREVA's PCS study and further recommended in the future studies section of this report, a simple steam cycle concept is the preferred configuration based on increased market flexibility, minimized technical risk, and most rapid deployment schedule.

1.4 Licensing

AREVA recommends that the NGNP be licensed under the conventional two step 10CFR50 licensing process:

1. Secure a construction permit based on the review of a preliminary safety analysis report (PSAR); and,

2. Secure an operating license based on the review of a final safety analysis report (FSAR). The license will initially be a Class 104 (c) license that will be converted to a commercial Class 103 license following a successful demonstration period.

Additionally, AREVA recommends that elements of 10 CFR 52 be carried out in parallel with the Part 50 license, in particular, maintaining close liaison with the NRC through pre-application technical exchanges and interactions that will be necessary to develop technology neutral and HTR-specific licensing bases for the NGNP demonstrator and commercial plants

This is essentially a hybrid approach that satisfies the need to initiate construction activities as early as possible but also minimizes the risk of construction prior to obtaining an operating license. However, the opportunity that the Part 50 approach offers in terms of an early construction start can only be seized if the key issue of containment versus confinement is resolved early in the licensing process.

Experience from the operation and prototype testing of the demonstration NGNP will be needed to support the design certification of future commercial versions because, unlike the NGNP, the commercial plan will follow the Design Certification Process provided in 10CFR52 [17].

1.5 Safety

The design philosophy for the AREVA NGNP concept is the application of safety features that provide a high confidence that the plant safety requirements are met with increased use of passive and inherent design characteristics. The key safety issues associated with the AREVA NGNP concept are: 1) radionuclide retention, 2) control of heat generation, 3) control of heat removal, 4) control of chemical attack on fuel particles, and 4) assurance of the reactor system geometry under a spectrum of design basis events that include event scenarios that are initiated by external hazards such airplane crash, internal and flooding, extreme weather conditions and seismic event. These issues are addressed in the design to provide a high level of safety consistent with the NGNP safety requirements.

1.6 Hydrogen Production Conclusions

While the design of the hydrogen production plant was not part of the AREVA team's assigned scope, the AREVA team did agree to perform an evaluation of high temperature steam electrolysis. This evaluation focused on overall system performance assuming heat was supplied using extraction steam from an adjacent steam cycle HTR and electric energy. Over a range of electrolyzer operating temperature of 600°C to 800°C, the best performance was predicted at 600°C. This result is based on the overall system performance, not just the electrolyzer.

1.7 Risk Conclusions

The key risks identified for the NGNP project are • Fuel performance (the probability of this risk can be reduced, but the potential consequence can not be minimized) • Heavy component procurement and fabrication (industrial capacity is limited both in forging size and supply schedule in the current market) • Licensing • Funding continuity (consistent effort is required for R&D, design, and procurement/construction in order to achieve the NGNP mission)

1.8 Research and Development Conclusions

The key R&D needs identified for the NGNP project are • Fuel manufacturing and qualification • Graphite qualification • Modified 9Cr-1Mo qualification and codification • High temperature materials • Methods qualification

1.9 Cost and Economic Conclusions

Costs were estimated for the First-of-a-Kind reference NGNP prototype, including R&D, design, capital, operation, and decommissioning. An economic analysis was also performed to estimate the output product cost for the commercial Nth-of-a-Kind VHTR plant. While the AREVA team did not assess the cost or economic performance of a comparable steam cycle NGNP alternative, it is clear that both the development cost and the capital cost of t steam cycle would be significantly lower than for the reference concept (probably 20-30%), while the reduction in electricity or hydrogen production efficiency would be relatively minor.

1.10 Future Studies

A number of future studies have been identified as being important in determining the future direction of the NGNP project, as being necessary to support anticipated conceptual design activities, or as being beneficial to the long-term deployment of future HTRs. Two key studies are important in determining the future direction of the program.

The first study is a thorough evaluation of the steam cycle as an alternative path to fulfill the NGNP mission on the desired near-term schedule with greater probability of success and greater potential for near-term commercialization. The second study further explores the NGNP power level issue in order to reconfirm the selected power level (565 MWth) or confirm a more suitable albeit lower power level that adequately addresses the potential differences between the NGNP prototype plant and possible future commercial HTR/VHTR plants.

1.11 Overall Conclusion

AREVA believes that the HTR has the potential to make a major impact on the broader energy market, if the necessary technology hurdles can be surmounted. The AREVA reference NGNP concept using multiple tubular IHXs is the best approach, if the direct delivery of nuclear high temperature heat is a fundamental requirement. However, the remaining technical challenges are significant.

As an alternative, AREVA believes that the steam cycle concept is the best path forward for near-term HTR deployment. It best meets the near-term process heat market for liquid fuels production and is suitable for steam electrolysis hydrogen production. It also provides a more solid foundation for the long-range deployment of advanced concepts as more advanced technology becomes available.

2.0 INTRODUCTION

2.1 General

This report describes the Next Generation Nuclear Plant Program (NGNP) preconceptual design studies performed by the AREVA NGNP team for the Battelle Energy Alliance (BEA), the Management & Operating Contractor of the Idaho National Laboratory (INL) as part of the Department of Energy's (DOE) Next Generation Nuclear Plant Project.

The purpose of INL in authorizing the preconceptual design studies described herein is two-fold:

1. Assist INL in focusing the technical scope and priorities of research & development activities for the NGNP.

2. Provide INL a basis for subsequent development of the technical and functional specifications for the prototype facilities for NGNP.

The preconceptual design studies as performed by AREVA within the authorized work scope and as reported herein are also consistent with the corresponding elements of the Phase I scope of work defined for the NGNP Project in the Energy Policy Act of 2005.

2.2 Energy Policy Act of 2005/NGNP Program Authorization

In July of 2005, Congress passed the Energy Policy Act of 2005 (H.R. 6), which was signed into law in August of 2005. Under Section 641, the Act states, "The Secretary [of Energy] shall establish a project to be known as the 'Next Generation Nuclear Plant Project'." It continues:

"The [NGNP] Project shall consist of the research, development, design, construction and operation of a prototype Nuclear System, including a nuclear reactor that:

- Is based on research and development activities supported by the Generation IV Nuclear Energy Systems Initiative...
- And shall be used to:
  - Generate electricity
  - Produce hydrogen, or
  - Both generate electricity and produce hydrogen."

The combination of these two objectives - to promote nuclear energy and to produce clean-burning hydrogen - can be met simultaneously with the development of new advanced reactor and hydrogen generation technology. The DOE's mission is then to develop and integrate these technologies to support its strategic goal of promoting diverse energy supply.

2.3 Scope of Work

The Scope of Work assigned to the AREVA NGNP team consists of the preparation of a preconceptual design report based upon the adaptation of AREVA's ANTARES HTR concept and preparation of supporting studies that will assist BEA/INL in focusing and prioritizing research and development work and preparing for the Conceptual Design of the NGNP. Also included in the report are the results of four special studies, two of which directly influenced the design adaptation, cost estimates for the FOAK plant (i.e., the NGNP prototype) and

NOAK commercial plant, and assessments of R&D needs, project risk and schedule, and requirements. The scope of the study is limited to the nuclear island and the power conversion system.

Because BEA/INL issued multiple awards, the final scope of work reported on herein is a reduced scope of work relative to the initial scope of work proposed by BEA/INL in their Statement of Work No. 3963, "Preconceptual Engineering Services for the Next Generation Nuclear Plant with Hydrogen Production," (Project No. 23843, July 26, 2006).

It is important to note the difference between the initial and reduced work scopes. The full scope was to be a new design based on what the contractor determined to be the best option to meet NGNP requirements. The reduced work scope (i.e., that assigned to AREVA) was that the preconceptual design work was limited to the adaptation of AREVA's ANTARES HTR to the NGNP. Furthermore, the number of special studies requested was reduced from six to four, and the hydrogen production plant/process heat scope was eliminated. Also eliminated were scope items related to high temperature heat transport, maintenance, availability, plant layout, safety, and licensing. The final agreed upon scope of work can be found in AREVA's NGNP Work Plan approved by BEA/INL on December 13, 2006. [1].

Key Deliverables

Key work elements developed within the framework of the scope of work are listed below:

- Develop NGNP preconceptual design
- NGNP System Requirements Manual
- Four supporting special studies:

1. Reactor Type Selection Study
2. Power Conversion System Study
3. Reactor Power level study
4. Primary-Secondary Systems Parameter and Configuration Study

- Identification of R&D needs and project risks
- Cost and Economics Analysis
- Project Schedule
- Issue a preconceptual design studies report on the adaptation of ANTARES to the NGNP
-

Of the above special studies, only the results of the reactor power level and primary-secondary systems study are factored into the NGNP preconceptual design. The direction to adapt the ANTARES indirect cycle CCGT concept to the NGNP made the results of the reactor type and power conversion system moot relative to the preconceptual design, nevertheless, they were performed due to the valuable insights they would provide BEA/INL for the NGNP conceptual design.

Finally, it is important to note the key items from BEA/INL's original work scope that were excluded from AREVA's work scope:

- Considerations of design options better suited to NGNP goals than the ANTARES concept
- Development of a detailed research and development plan
- Hydrogen plant

- High temperature heat transport system
- High temperature process heat, transfer, and transport study
- Licensing and Permitting Study
- NGNP byproducts study

2.4 AREVA NGNP Preconceptual Design Organization

AREVA, as the lead BEA NGNP contractor for this work scope, has the overall project responsibility. In support of the NGNP preconceptual design work, AREVA assembled a team of sub-contractor companies with the key technical competencies needed for full execution of this and follow-on phases of the NGNP project including final design, construction, and operations.

The AREVA NGNP Team includes Burns & Roe, Washington Group International, BWXT, Dominion Engineering, Air Products, Hamilton-Sundstrand-Rocketdyne, and Mitsubishi Heavy Industries (MHI). The team organization is show graphically on Figure 2-1 below:

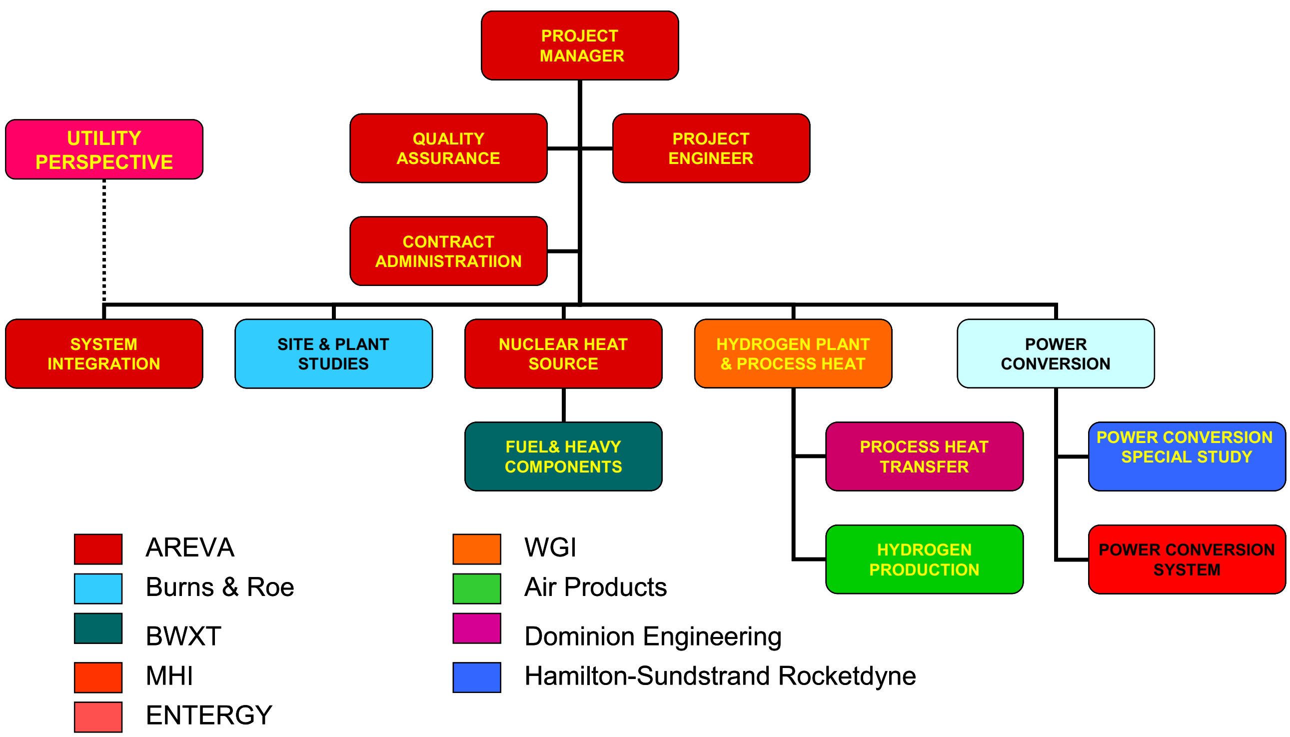


Figure 2-1: AREVA NGNP Organization

2.5 ANTARES Overview

The NGNP preconceptual design presented in this report is based on the adaptation of AREVA's ANTARES to the NGNP. Because it was the starting point for AREVA's NGNP preconceptual design work, a brief overview of ANTARES is presented below.

ANTARES is a graphite moderated, helium-cooled reactor. It uses an annular prismatic block core rated at 600 MWth. The prismatic block concept was selected for the ANTARES, because it allows a higher power level for improved economics, core management capability for greater fuel cycle flexibility, and lower core pressure drop for improved system performance.

ANTARES uses an indirect cycle configuration as shown in Figure 2-2. An intermediate heat exchanger (IHX) couples the reactor to a conventional combined cycle gas turbine (CCGT) generating system. This configuration results in an energy delivery system that can be coupled to a wide variety of applications. Table 2-1 summarizes the key features of the reference ANTARES design.

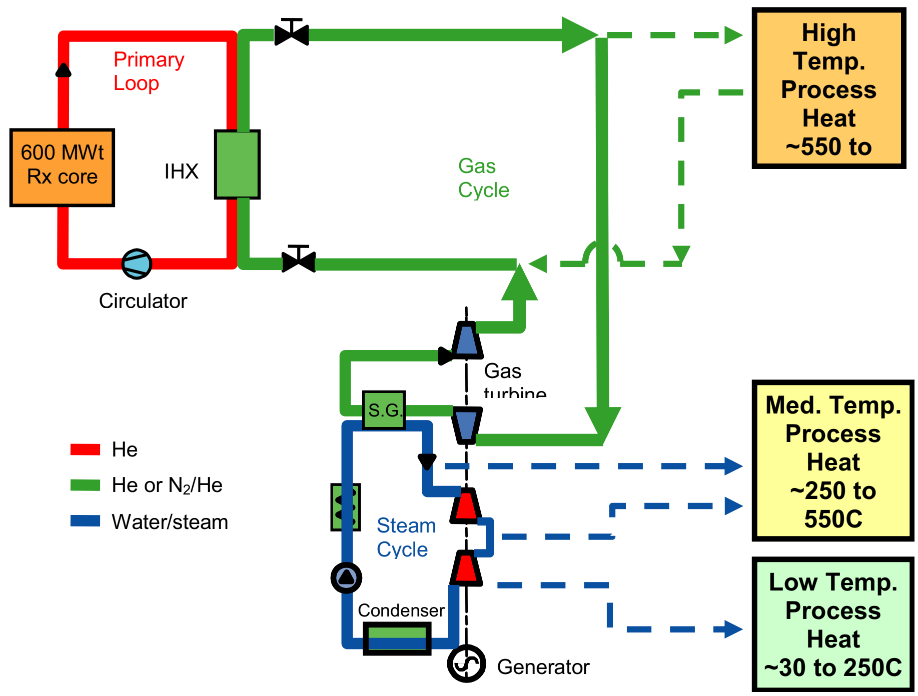


Figure 2-2: Reference ANTARES Configuration (Basis for Adapted NGNP Design)

Table 2-1: Reference ANTARES Configuration Parameters

Fuel Type	TRISO, Prismatic Block
Core Geometry	102 column annular 10 block high
Reactor Power	600 MWth
Reactor Outlet Temperature	850°C
IHX Secondary Outlet Temperature	800°C
Primary Pressure	6 MPa
Primary Coolant	Helium
Secondary Coolant	Nitrogen/Helium
Reactor Vessel	Mod 9Cr 1Mo
Power Generating System	CCGT – Gas turbine with bottoming cycle
Net Efficiency	>46%

2.6 Report Organization

The NGNP Preconceptual Design Studies Report (PCDSR) presented herein represents AREVA's adaptation of ANTARES to the NGNP requirements. The PCDSR consists of 21 sections and 3 appendices as can be viewed in the Table of Contents (TOC). The TOC organization mirrors the original outline proposed by BEA/INL in the initial statement of work; however, it has been appropriately adjusted to reflect AREVA's final work scope for NGNP preconceptual design. Briefly, the information presented is as follows:

- Sections 1 and 2 present summary and introduction
- Section 3 provides an overall description of AREVA's reference NGNP concept
- Section 4 addresses key project requirements
- Section 5 addresses hydrogen plant interface assumptions and evaluation of steam HTR system performance
- Sections 6 through 9 describe the reactor, reactor support systems, power conversion system and plant layout in detail.
- Sections 10 and 11 address overall plant operations and safety, respectively.
- Section 12 describes AREVA's licensing strategy for NGNP.
- Sections 13 and 14 address maintainability and availability.
- Section 15 presents the fuel fabrication and qualification strategy
- Sections 16, 17, and 18 examine economics, schedule, and risk management.
- Section 19 reports on research and development needs and is also supplemented by information provided in Appendix C.

- Section 20 summarizes the results of the four special studies (which are appended in their entirety in Appendix B).
- Section 21 discusses future studies and related work recommended for the NGNP conceptual design phase; or, before.
- Section 22 present overall conclusions
- Section 23 References

Five appendices are also provided:

- Appendix A - The NGNP System Requirements Manual
- Appendix B – Special Studies:
    - B1 - Reactor Type Comparison Study
    - B2 – Prototype Power Level Study
    - B3 – Power Conversion System Study
    - B4 – Primary and Secondary Cycle Concept Study
- Appendix C – Research and Development
- Appendix D - System Analysis of High Temperature Steam Electrolysis
- Appendix E - 90% Review Meeting – Comments/Resolutions

3.0 OVERALL NGNP PLANT DESCRIPTION

The NGNP is aimed at producing both electricity and hydrogen in a cogeneration mode. The plant can operate in an all-electric mode or it can also produce hydrogen and electricity simultaneously. The NGNP is envisioned as a flexible demonstration and R&D facility, and it is expected that multiple high temperature hydrogen production processes and components will be demonstrated.

The NGNP plant consists of the following:

- Nuclear Heat Source
- Power Conversion System
- High Temperature Heat Transport Loop
- Hydrogen Production Plant.
- Site facilities.

The reference AREVA NGNP design studied within the present preconceptual design work is based on an adaptation of the ANTARES design. It consists of a modular Very High Temperature Reactor (VHTR) coupled to a combined cycle gas turbine (CCGT) generating system. The design is based on an indirect cycle configuration in which heat from the reactor is transferred to a closed loop Brayton cycle through Intermediate Heat eXchangers (IHXs). A nitrogen based fluid is used in the secondary circuit in order to allow air-breathing gas turbine technology to be used. In addition, heat is also provided to a heat transport loop connected to a hydrogen production plant. The proposed concept is illustrated schematically in Figure 3-1.

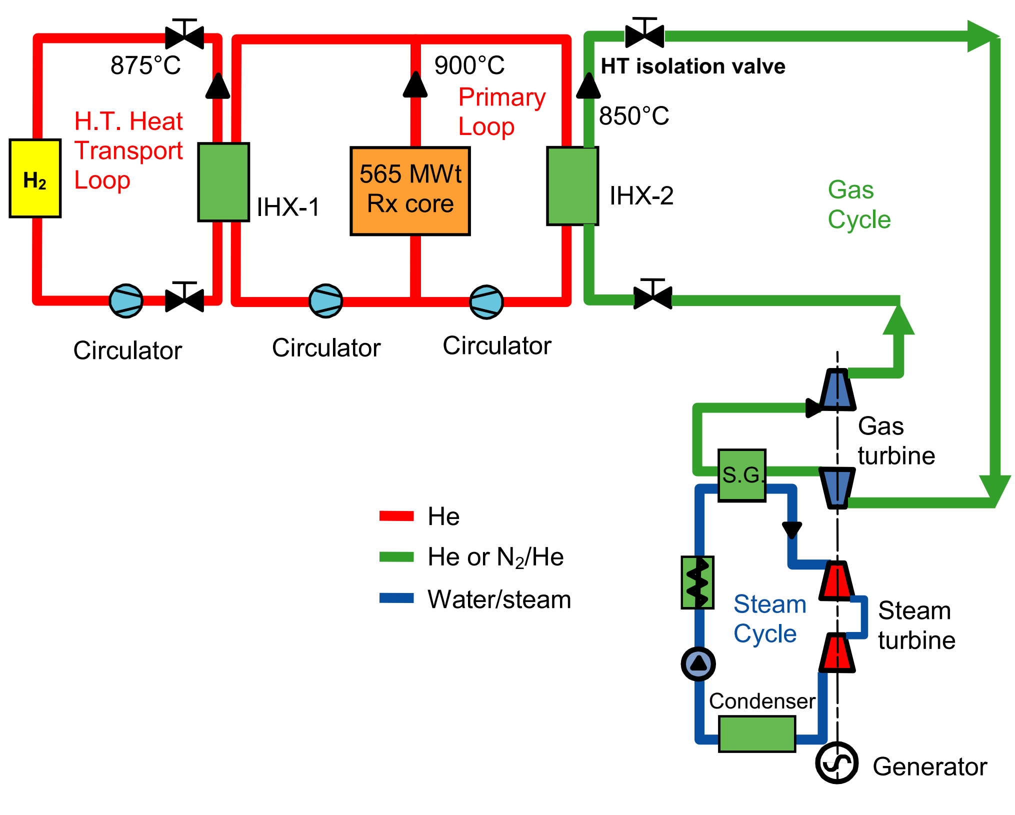


Figure 3-1: AREVA HTR Indirect Cycle NGNP Schematic

The indirect cycle offers several advantages to reduce the overall development risk in contrast to a direct cycle concept. As already noted, the indirect cycle allows the use of air-breathing gas turbine technology, avoiding the development of helium turbomachinery. The indirect cycle also makes maintenance and potential modification and adjustment of the system more practical, since the equipment is distributed rather than being in a tightly integrated configuration. Also, contamination of the power generating equipment is minimized, since any circulating radiocontaminants are confined to the primary circuit. In addition, operation and analysis of plant performance is simplified, because the dynamics of the reactor and primary circuit are partially decoupled from the power generating system. The indirect cycle also allows considering similar nuclear heat sources for a variety of applications, i.e., electricity generation or process heat application.

The values of the normal operating parameters used during the Preconceptual Design Phase are indicated in Table 3-1.

Table 3-1: Normal Operating Parameters

Parameter	Selection
Primary Side	
Primary Fluid	Helium
Reactor Power	565 MWt
Reactor Outlet Temperature	900°C
Reactor Inlet Temperature	500°C
Primary Coolant Flow Rate	272 kg/s
Primary Coolant Pressure	5 MPa at the circulator outlet
Heat transport to Hydrogen Production Plant	
Secondary Fluid	Helium
Heat Load	60 MWt
Heat transport to Power Conversion System	
Secondary Fluid	Nitrogen/helium mixture Reference: He 20% - N2 80% in mass
Heat Load	578 MWt (all electric mode)
Power Generation	
Power Generation System	Combined cycle with Brayton topping cycle (secondary circuit) and Rankine bottoming cycle (tertiary steam/water circuit)

3.1 Nuclear System Arrangement

3.1.1 Module Description

The Nuclear Heat Source (NHS) of the AREVA NGNP plant is a modular graphite-moderated, helium-cooled nuclear reactor, located in a metallic vessel. This type of reactor has the capability to supply high temperature heat for a variety of applications, and its safety characteristics provide advantages in plant siting, in protection of capital investment, and in minimizing the number of safety systems required. The AREVA NGNP design uses a completely ceramic prismatic block reactor core. The absence of metal alloys in the core allows very high reactor outlet temperatures to be achieved during normal operation.

The fuel consists of approximately 20 billion ceramic coated fuel particles, each being about 1 mm in diameter. Each "TRISO" particle has a fuel kernel at the center surrounded by three successive layers of low and high density carbon and silicon carbide. Figure 3-2 illustrates the coating layers, including a final outer layer of high density carbon which protects the SiC during manufacturing. These layers retain fission products within the fuel particles during normal operation and accident conditions. The fuel particles are molded into cylindrical rods called compacts, and loaded into the prismatic graphite fuel blocks.

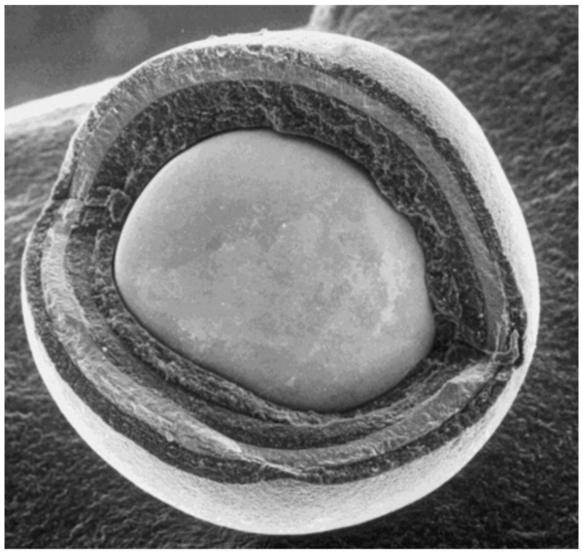


Figure 3-2: Internal Structure of TRISO Coated Fuel Particle

The AREVA NGNP concept uses an annular core. The active portion of the core consists of 102 columns of 10 blocks each, for a total 1020 fuel elements. This configuration was selected based on experience from AREVA NP participation in the early phases of the General Atomics/OKBM GT-MHR program.

Heat produced in the reactor is transferred to IHXs via the Primary Heat Transfer System (PHTS). The IHX is the functional interface between the NHS and the energy user. Two types of IHX are proposed for the NGNP: tubular IHXs for heat transfer to Power Conversion System and a compact IHX for the H2 plant.

Tubular IHXs are preferred over compact IHXs for the PCS due to the inherent robustness of this type of concept and the maturity of this design to operate at temperatures of 900°C or above for a design life of 20 years. Compact IHX technology is selected for the H2 plant due to more benign service conditions and to limited impact of a shorter design life on the overall plant costs. Moreover, it is expected that this IHX will be replaced more frequently in any event in order to test alternate technologies.

In addition to the IHXs, the PHTS includes the main helium circulators, and the required ducting to channel the coolant from the reactor to the IHX, from the IHX to the circulator, and from the circulator back to the reactor inlet.

The NHS systems are contained in a steel vessel system. The reactor pressure vessel is fabricated from Modified 9 Cr – 1 Mo. This alloy provides increased high temperature capability which is compatible with the reactor operating temperatures and provides sufficient margin for off-normal events.

The Reactor Vessel contains the reactor core and internals. Each IHX Vessel contains IHX modules, main circulator, and the necessary interconnecting primary and secondary coolant ducts. The Cross Vessels connect the Reactor Vessel and the IHX Vessels, and they maintain concentric flow paths between the reactor and IHXs. During normal operation the IHX separates the primary and secondary coolant, however isolation valves are provided to complete the primary coolant boundary in case of IHX or secondary duct leakage. More generally, the circuits interfacing the primary circuit are equipped with isolating devices for minimizing the consequences of failures.

The vessels are located in a dedicated Reactor Building. The entire Reactor Building is located below grade, except for the associated portion which houses the Reactor Cavity Cooling System water storage tanks. The Reactor Building structure provides protection against external hazards including seismic events and aircraft threats.

The Reactor Service Building is located at one end of the reactor complex. It houses the new fuel preparation and storage area and irradiated fuel storage. The Reactor Auxiliary Building houses waste processing and other necessary functions. Non-nuclear activities are housed in other adjacent buildings where the main control room is also located. Additional long-term irradiated fuel storage would be provided in a separate storage facility.

A commercial plant would be made of several NHS modules as described above, but for which support facilities and systems would be shared between several modules.

3.1.2 Safety Characteristics

The design philosophy for AREVA NGNP concept is the application of safety features that provide a high confidence that the plant safety requirements are met with increased use of passive and inherent design characteristics. The key safety objectives associated with the AREVA NGNP concept are: 1) radionuclide retention, 2) control of heat generation, 3) control of heat removal, 4) control of chemical attack on fuel particles, and 5) assurance of the reactor system geometry under a spectrum of design basis events that include event scenarios that are initiated by external hazards such as airplane crash, internal and flooding, extreme weather conditions or seismic event.

Safety of the AREVA NGNP concept stems from the following design characteristics:

Helium Coolant - The use of an inert, single phase gaseous coolant eliminates the possibility of a complete loss of coolant event. Circulator cavitations can not occur, and no chemical reaction between the coolant and graphite, or fuel is possible. Furthermore, adequate core heat removal can be achieved either actively or passively under either pressurized or depressurized primary system conditions.

Graphite Core - The high heat capacity and the low power density of the graphite core in an annular configuration results in very slow and predictable temperature transients. In addition, the strength of graphite increases with temperature up to levels well above those associated with licensing basis events.

Ceramic Fuel Particles - The primary fission product barriers in the NGNP fuel are multiple ceramic coatings covering the fuel kernel. These coatings will retain a large fraction of the fission products.

Negative Reactivity Feedback – Reactor core design retains an inherent high negative reactivity feedback characteristic. If core temperatures increase, power level decreases. This property ensures that fuel temperature rise is self-limited for loss of heat removal events.

Core Region Structural Configuration – The annular geometry of the core, low power density, and high thermal capacity ensure the cooldown of the shutdown reactor under emergency conditions by passive removal of heat from the reactor vessel using heat emission, conduction, and convection. As a result, fuel and core temperatures remain within allowable limits in all relevant accident scenarios.

In addition, the AREVA NGNP concept has multiple decay and residual heat removal systems, including the secondary nitrogen/helium loop in conjunction with the primary heat transport loop, the Start-up and Decay Heat Removal System (SDHRS), the reactor Shutdown Cooling Systems (SCS), and the Reactor Cavity Cooling System (RCCS). Coupled with the inherent safety characteristics, the peak fuel temperature that occurs during the passive residual and decay heat removal condition will always fall below that minimum fuel temperature that would cause significant fuel damage and release of fission products to the primary system. Multiple reactor trip mechanisms are also provided, including the normal shutdown system and the Reactor Reserve Shutdown System.

An added benefit of the inherent characteristic of the AREVA NGNP concept plant is its slow and benign response which, combined with passive residual heat removal, simplifies the operator's role and provides long time intervals for deliberate actions, thus minimizing the opportunity for operator error.

3.2 Power Conversion System Arrangement

The AREVA NGNP PCS system is based on the adaptation of the ANTARES combined cycle gas turbine concept to NGNP design conditions.

The Brayton cycle consists of the gas turbine unit (gas turbine, compressor and auxiliaries) and the interconnecting ductwork. The main function of the gas turbine unit is to convert the thermal energy contained in secondary circuit gas exiting the IHX into electrical power. The shaft power generated by the gas turbine drives both the gas compressor and electrical generator. The heat content of the turbine exhaust gas is significant and much of it is transferred to the tertiary steam cycle.

The major components of the tertiary circuit are the Heat Recovery Steam Generator (HRSG), the HP/IP/LP turbine units and the generator, and the condensate system. Superheated high pressure steam from the HRSG goes into high pressure (HP) turbine and then, upon exhaust, is conducted to the HRSG reheating zone. The reheat steam from the HRSG goes to the intermediate pressure (IP) turbine. The exhaust steam from IP turbine is conducted to the low pressure (LP) turbine. The exhaust steam from the LP turbine flows directly to the steam condenser. The condensate system condenses the steam from the LP turbine exhaust and supplies condensate to the feed water heaters.

Pipes are used to transport the hot gas (850ºC) from IHX outlet to gas turbine inlet and from gas turbine outlet to HRSG inlet. Pipes conveying high temperature gas are insulated on the inner surface to keep their operating temperature low.

The PCS configuration considered for NGNP is shown schematically in Figure 3-3.

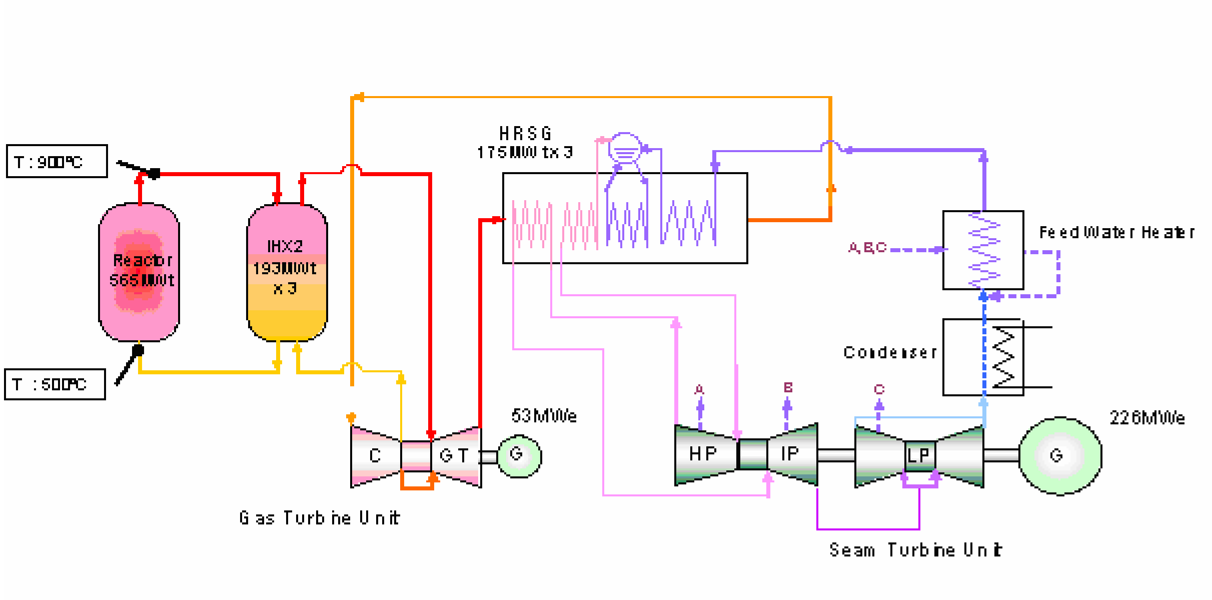


Figure 3-3: PCS Configuration

This NGNP PCS configuration allows the demonstration of separate turbine generator sets and permits each power unit to be uniquely optimized for its given conditions. Furthermore, in a multiple-module setting, it is feasible to feed a common steam-turbine unit from two or more reactor modules. The NGNP PCS configuration allows the demonstration of the key control features that would be required by such an arrangement.

Table 3-2 provides a summary of the parameters of the PCS system.

Table 3-2: Main Parameters of the PCS system

PCS Configuration		
PCS type	-	Combined cycle with Brayton topping cycle and Rankine bottoming cycle
Shaft configuration	-	Multishaft gas turbine / steam turbine
Total output	MWe	279
Gas turbine unit		
Fluid	-	He (20%) + N2 (80%)
Gas turbine inlet/outlet temperature	°C	850 / 600
Rated output	MWe	53
Speed	rpm	3600
HRSG Module		
No. of Vessels	-	3
Heat duty/vessel	MWth	175.1
Steam turbine		
Type	-	Single reheat
Rated output	MWe	226
HP steam temperature	°C	535
HP steam pressure	MPa	11.8
Speed	rpm	3600

3.3 Heat Transport Loop

Out of AREVA team's scope of work.

3.4 Hydrogen Production Plant

Out of AREVA team's scope of work.

3.5 Operation and Performance

3.5.1 Normal Nuclear System Operation

The AREVA NGNP design and operations are based on a commercial scale nuclear heat source that will be used for both electricity and process heat generation and hydrogen production.

Consistent with the BEA / NGNP requirements, the AREVA NGNP prototype facility design is based on a once-through low-enriched uranium core capable of high burnup. The plant will be designed for a 60 year life span and provisions will be made to replace components with shorter lifetimes. The number of primary system components to be maintained is minimized by the selection of the indirect cycle architecture and the remaining systems and components in the power generation and hydrogen processing sections will be readily accessible for inspection, repair and replacement. The indirect cycle design also minimizes the transient coupling of the nuclear heat source and the heat utilization systems.

The AREVA concept for the NGNP combines the VHTR NHS with a multi-purpose energy utilization architecture that will demonstrate both advanced electricity and process heat generation and hydrogen production applications. The AREVA NGNP design incorporates an indirect combined cycle power generation system that is capable of using the full thermal capacity of the reactor, 565 MWt at an efficiency level that exceeds the NGNP requirement (44%). Heat from the NHS is supplied to the Power Conversion System (PCS) through three parallel IHXs for electricity production (see Figure 3-4).

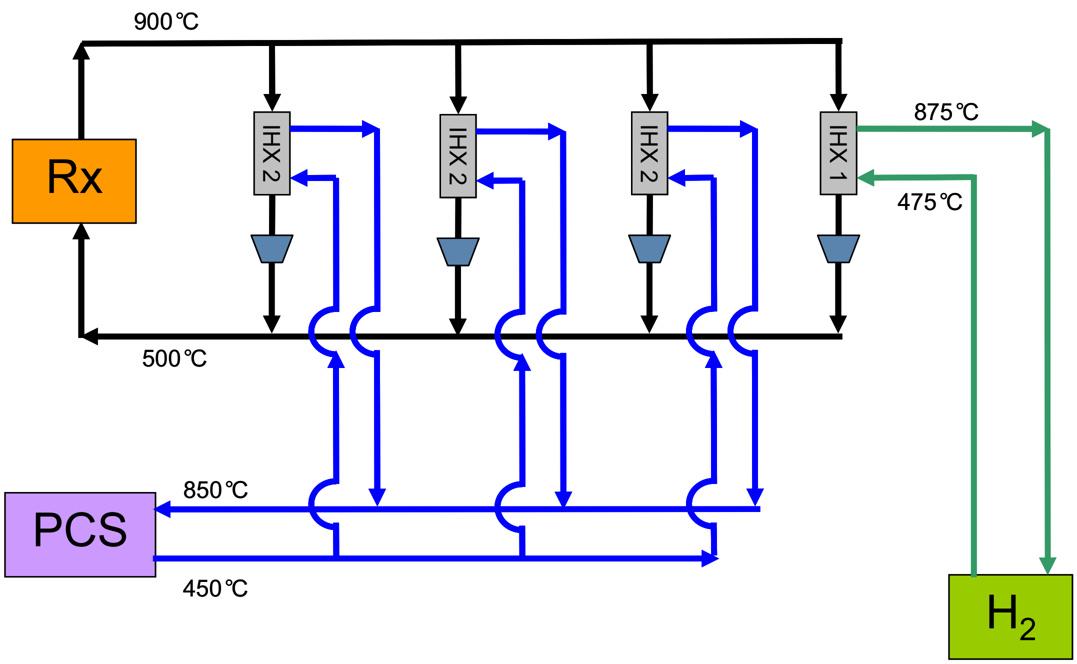


Figure 3-4: NGNP Parallel IHX Configuration

The heat transport architecture of the AREVA NGNP provides an optimum platform for the demonstration of high-temperature hydrogen production processes such as thermo-chemical and electrolysis, while allowing full utilization of the reactor output for electricity generation, when the hydrogen production section is not in service. The 60 MWt heat transport helium loop that supports the hydrogen process energy requirements minimizes radiological contamination, and allows modification of the hydrogen production section, independent of operations in the electricity generation portion of the plant. The 900ºC temperature provided at the process coupling heat exchanger fully supports high efficiency, based on data pertaining to the Sulfur-Iodine thermo-chemical water-splitting cycle. The heat transport loop and associated circulator further provide for independent control of heat transport to the hydrogen production process and minimizes the potential for adverse interactions with the remainder of the plant.

The indirect cycle architecture of the AREVA NGNP, in conjunction with the energy transport architecture provides a flexible basis for future testing, including those for demonstrating investment protection and safety margins. This is facilitated by the simple and reliable design of the reactor and primary heat transport loop that minimize the likelihood of unforeseen consequences that might result from unusual operating conditions associated with testing.

The NGNP prototype facility is designed for the control and operation of a single reactor module, two turbine-generators, and/or the delivery of process heat by operators from a central control room. Monitoring systems enable maintenance and operation tasks during normal and abnormal operation. The main objective in defining the conditions for steady-state operation is that normal operations, upsets, or unanticipated transients in either the nuclear plant or the hydrogen production plant must not compromise safe operation of the other portion of the plant.

Strategies for operational control of the NGNP plant will optimize the reliability, availability, and safety of the NGNP plant systems, structures, and components. This objective is achieved through the identification and implementation of state-of-the-art component designs, predictive maintenance methods, advanced information technologies, and incorporation of human factors and safety culture.

3.5.2 Off-Normal Operation

Plant control and protection systems are to be highly automated. An operator is expected to supervise all automatic control actions and have the means to control actions during both anticipated and unanticipated events. Each plant (nuclear and hydrogen) should have its dedicated control and protection system but a supervisory control system will be required to ensure coordinated actions.

The proposed control for electrical power transients uses a secondary by-pass that by-passes the IHX and the turbine to adapt the power conversion system to the demand. This control is performed by the Power Output Controller. The steam/water circuit is adjusted to the new conditions by means of the Steam Generator Level Controller and the Steam Pressure Controller. The primary circuit must also be controlled to adapt the primary circuit to secondary circuit and to maintain constant the core outlet temperature using the Primary Flowrate Controller and the Core Outlet Temperature Controller.

Optimization of the primary/secondary IHX pressure difference can be controlled by actuating the primary and secondary inventory valves.

Several anticipated load varying events to consider include:

- Load following –ability to maintain electric or process heat load in light of changing load demand
- Load rejection – ability to rapidly reducing power production (without a reactor trip) after total loss of electrical load.
- Reactor trip – capability to trip the reactor when safety limits are approached
- Turbine trip – capability of orderly reactor shutdown when the main turbines are tripped.

3.5.3 Nominal System Performance

With the NHS and PCS system described in Sections 3.1 and 3.2, the performance expected for the NGNP for electricity production is the following:

- Gas turbine/generator unit: 53 MWe
- Steam turbine/generator unit: 226 MWe
- Total Power Generation: 279 MWe

This would correspond to a net efficiency of the NGNP plant of 45.8 %.

4.0 KEY PROJECT REQUIREMENTS

The AREVA NGNP preconceptual design study project goal is to adapt the ANTARES high temperature modular gas-cooled reactor concept, to meet the mission of the DOE NGNP reactor. The ANTARES design has recently completed the preconceptual design stage and is well positioned to be further developed to fulfill the DOE NGNP prototype facility mission and goals.

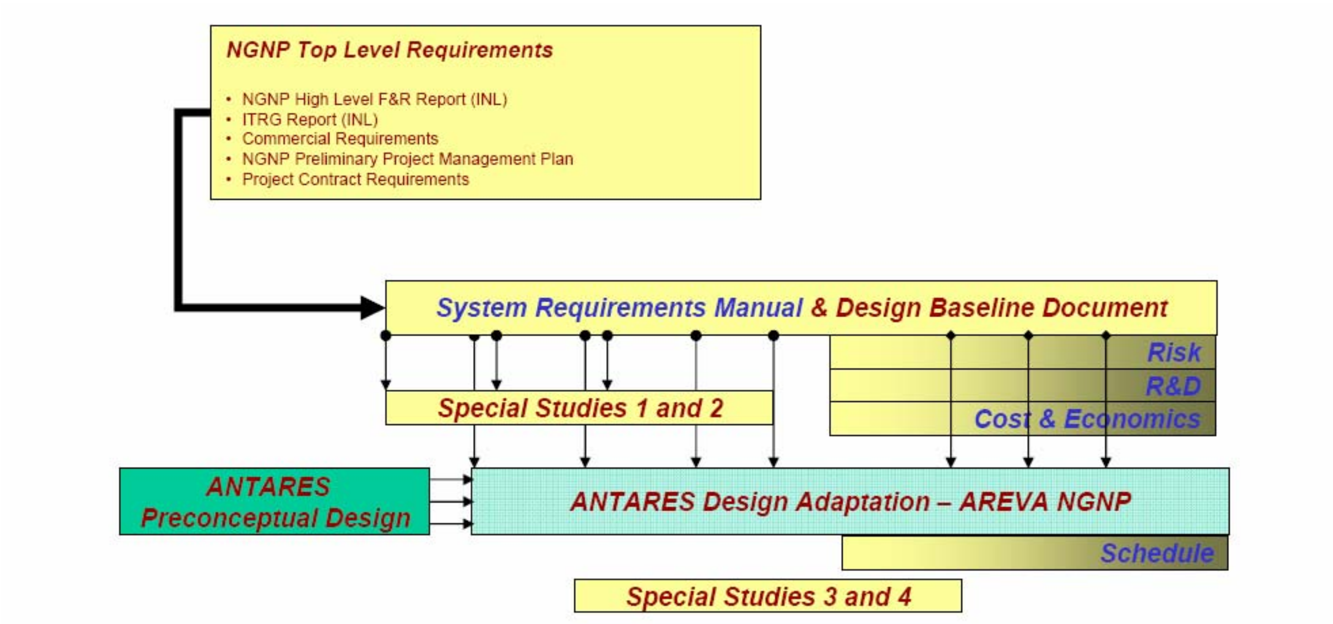


Figure 4-1: NGNP Pre-Conceptual Design Adaptation

The NGNP preconceptual design adaptation process utilizes a top down system engineering approach. As illustrated in Figure 4-1, top level NGNP requirements are taken from the NGNP High Level Functions and Requirements document [2] , ITRG Report [3] , commercial views provided by AREVA, NGNP Preliminary Project Management Plan [4] , and present project contract requirements.

Top level requirements were prepared and documented in a System Requirements Manual (Appendix A). Four special studies were also conducted to derive the NGNP power level, primary and secondary temperatures pressures and configuration.

Key top level plant requirements are provided in this chapter. The NGNP nuclear heat source is intended to drive an electric plant and a scale hydrogen production plant. However, the heat transport loop to the hydrogen plant and the hydrogen plant was not within the scope of the present contract, therefore, only certain interfacing criteria have are given in this document.

4.1 NGNP Preconceptual Design Studies Missions

The missions of the NGNP preconceptual design work described here include:

- Assisting in focusing the technical scope and priorities of research & development activities for the NGNP.
- Providing a basis for subsequent development of the technical and functional specifications for the prototype facilities for NGNP.

This work is consistent with the Phase I scope of work defined for the NGNP Project in the Energy Policy Act of 2005. Phase I will define the initial design parameters for the NGNP, select the principal hydrogen production technology (not within AREVA scope of work), and identification of supporting research and development.

Pre-conceptual design work includes evaluation of a range of design parameters and alternatives (special studies), and based on the justification for the parameters and alternatives, prepare a preconceptual design adaptation of AREVA ANTARES design for the NGNP prototype facilities.

An important purpose of the evaluation of parameters and alternatives is to determine the most appropriate configuration of the prototype to enable subsequent commercialization of NGNP technologies (e.g., scaling and licensing considerations). Research & development needs deemed necessary to select such parameters and alternatives will be identified.

The design parameters and alternatives to be evaluated will include:

- Reactor thermal power level
- Primary and secondary cycle concept studies (i.e., reactor inlet and out let temperature, secondary circuit design and temperature, and primary and secondary pressure)
- Reactor design (e.g., pebble bed compared to prismatic block)
- Power conversion concept (e.g., direct compared to indirect cycle, power conversion machinery concept)

The AREVA NGNP preconceptual design work will establish the basic reactor geometry and layout; perform reactor physics, thermal fluids studies, and heat balance calculations; provide subsystem identification and relative sizing (heat exchangers, pumps, compressors, piping, structural, etc.) including balance of plant support facilities; provide general site layout, subsystem, and plant interfaces.

The modular reactor design is based on a single module capacity AREVA ANTARES high temperature gas cooled reactor concept. The optimal size and design temperature for the reactor is determined for a “commercial scale prototype reactor” for electrical power generation and optimal Sodium–Iodine Thermo-chemical or High Temperature Electrolysis hydrogen production facilities and provisions for other industry applications of high temperature process heat.

This work includes industry literature research on: current and emerging technologies for very high temperature reactors, hydrogen and electricity production, other industry high temperature process heat applications study, capital cost estimates, operating cost estimates, life cycle cost estimates, economic analysis, and project schedule, all at a preconceptual design level of detail.

Information from the preconceptual design study will provide NGNP program management with geometric data, identification of critical Structures, Systems, and Components (SSC) and data that are needed to further direct and focus planned research and development in the areas of reactor safety and design methods, fuels, materials, and input regarding the appropriate overall licensing strategy that should be followed for the NGNP prototype.

4.2 Top Level Plant Design Requirements

The hierarchy of the NGNP plant requirements is provided in Figure 4-2. At the highest levels are the project goals and objectives as well as the regulatory requirements for the NGNP plant. From these top level requirements, the lower plant level functions and requirements are derived. The functions and requirements of each of the individual systems and sub-systems are then defined; each of these system level functions and requirements is based upon one or more of the top-level requirements defined above.

The NGNP prototype facility top level plant design requirements are:

1. NGNP shall be designed, constructed, licensed, and operating by 2020 with initial operations in 2018.
2. NGNP design configuration shall consider cost and risk profiles to ensure that NGNP establishes a sound foundation for future commercial deployment.
3. NGNP nuclear heat source shall be based on the modular high temperature gas-cooled reactor concept and utilize passive safety features to cool the core from full power to safe shutdown conditions.
4. NGNP shall produce high efficiency electricity and generate hydrogen on a scale that sets a foundation for future commercial deployment.

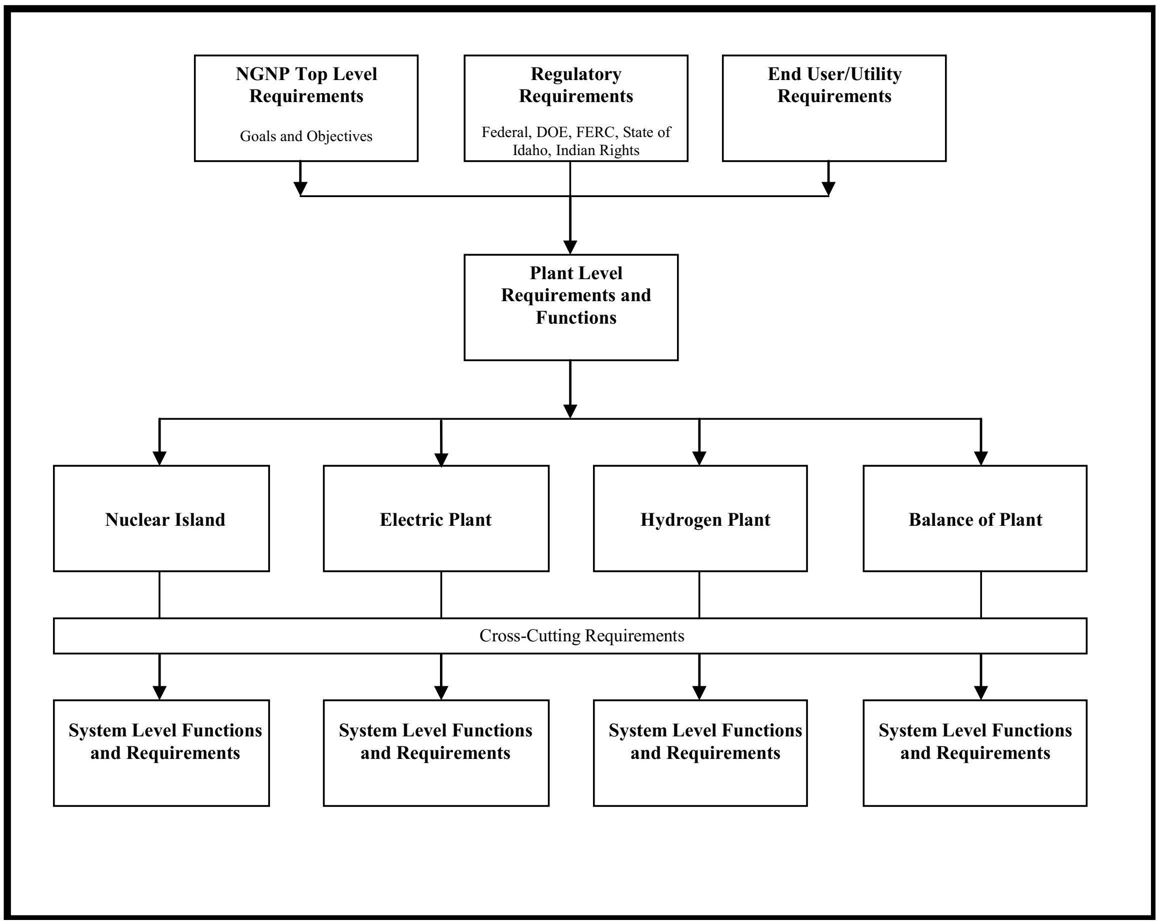


Figure 4-2: NGNP Requirements Hierarchy

5. NGNP shall be licensed by the NRC as a commercial cogeneration facility producing electricity and hydrogen.
6. NGNP shall include provisions for future testing.
7. NGNP shall enable demonstration of energy product and processes utilizing its nuclear heat source.
8. The project shall include identification of necessary and sufficient R&D technical scope and priorities.
9. NGNP plant licensing shall support potential future US-NRC technology neutral rulemaking activities (i.e., Risk-Informed, Performance-Based Alternative to 10CFR Part 50).
10. For the purposes of the AREVA preconceptual design activity, existing AREVA Nuclear Island and Power Conversion System shall be utilized as the bases for the NGNP prototype facility.

The NGNP prototype facility top level requirements flow down to the following four areas: nuclear heat source (reactor), power conversion system (PCS), heat transfer loop and hydrogen plant (not in AREVA scope of work), and balance of plant (site). Appendix A contains the detail system requirements manual (SRM) developed for the AREVA NGNP prototype facility.

4.2.1 Nuclear Heat Source

The NGNP nuclear heat source shall use modular high temperature gas cooled reactor concept that includes inherent safety features such as high negative reactivity temperature coefficient and passive methods for residual and decay heat removal during accident conditions. The reactor shall use thermal neutron spectrum, graphite moderator, low power density, TRISO coated particle fuel, low enriched uranium, and annular design core to achieve a degree of nuclear safety where public safety and protection does not depend on off-site emergency evacuation. The reactor and the intermediate heat exchanger shall use an underground silo to reject heat to the earth in case the engineered passive reactor cavity cooling system becomes inoperative. The underground silo construction shall provide the added aircraft crash protection.

The safety design of the nuclear heat source shall be such that a nuclear plant restart is would be possible after reasonable inspection and repair period, i.e., design bases accidents shall not result in total plant write-off.

4.2.2 Power Conversion System (PCS)

The NGNP power conversion system shall produce electricity at high efficiency. The minimum electricity generation efficiency shall be 44%. The power conversion system shall use minimum amount of water to reject heat to the environment. The ultimate condenser heat sink shall be via a water circuit to a natural or forced convection cooling towers. To achieve a high efficiency electricity a combined gas turbine Brayton and Rankin steam cycle with single or multiple reheat loops shall be considered. Single or multi-shaft generator arrangements shall be considered.

The high temperature process heat shall be delivered to the power conversion system through multiple intermediate heat exchangers. The secondary heat transfer medium is a mixture of 20% He and 80% Nitrogen which allows the use of air breathing gas turbine technology.

4.2.3 Heat Transfer Loop

The design of the heat transfer loop to the hydrogen plant is not within the scope of AREVA work. For interfacing purposes AREVA shall design a small compact intermediate heat exchanger to transfer 60 MWth of primary heat to the Hydrogen loop. The heat shall be delivered to the using a He heat transfer medium at 875°C. The return temperature from the Hydrogen plant is assumed to be 475°C.

4.2.4 H2 Plant

The design of the scaled hydrogen plant is not within the scope of AREVA work. For interfacing purposes a scaled hydrogen plant utilizing Sulfur – Iodine thermo-chemical hydrogen production plant and consuming 60MWth of heat at 850°C peak process temperature is assumed.

4.2.5 Site

The NGNP plant site shall be located at the Idaho National Laboratory NPR (new production reactor) site. The plant shall generate commercial quantities of electricity and be connected to the near by electrical distribution grid system. The balance of plant facilities shall include long term (15 years) storage capability for used fuel, buildings and facilities to operate the nuclear plant, the hydrogen plant and provision for improvements for facilities to test other process heat applications on an industrial scale.

The plant heat sink for electricity production shall be natural or forced draft cooling towers. The source of fresh water is limited to deep ground aquifers. The usage of fresh water shall be kept at an absolute minimum.

4.3 Intermediate Level Requirements

4.3.1 Nuclear Heat Source

Functions	Requirements
Reactor System	
Generate heat and transfer it to the primary coolant Maintain reactor shutdown	The reactor system shall be designed to provide the possibility of passive decay and residual heat removal. The reactor system shall be designed for an operational lifetime of 60 years. The reactor system shall be designed to provide dual hydrogen and electricity generation. The core shall use forced circulation helium as the heat transport medium. Non replaceable structural materials in contact with helium shall resist corrosion and erosion during plant cycle life
Reactor Core	
Generate heat Transfer heat to coolant and/or reactor internals	The decay heat removal shall be possible by passive heat transfer means (conduction and radiation) from the fuel to the reactor internals without reaching unacceptable fuel temperatures during all design basis accident conditions. The core shall utilize thermal spectrum neutrons for fission reaction The core shall be moderated with graphite. The active core height shall ensure the axial stability of the neutron flux and preclude the risk of xenon oscillations. Reference fuel shall be low enriched uranium based (UCO or UO_2) with an enrichment limited to <20.0% (in mass) and with a peak burn-up limited to 20% FIMA. The core bypass flow shall be maintained within an acceptable range which ensures a good compromise for the fuel temperature in normal and accidental conditions (existence of a minimum amount of bypass in lateral reflector).

Functions	Requirements
	The fuel handling sequences for core reloading shall be completed within a time interval consistent with the prescribed outage period. The reactivity temperature coefficient shall be sufficiently negative to shutdown the nuclear chain reaction before an unacceptable fuel temperature is reached, and maintain the core in a safe state for a time offering the certainty to reliably introduce absorber elements
Reactor Cavity Cooling System	
To protect the reactor cavity concrete structure including the support structures of the reactor pressure vessel from overheating during all modes of operation. To provide an alternate means of reactor core heat removal from the Reactor System to the environment when neither the PHTS nor the SCS is available	The RCCS shall operate continuously and maintain reactor cavity concrete temperatures less than [90ºC] during normal operations and less than [150ºC] for off-normal events (short term). The RCCS shall be designed to operate through the utility/user duty cycle events for the number of cycles specified [TBD] plus those events and even combinations determined to be required by plant transient analysis. Inaccessible parts of the RCCS shall be designed for an operating life of 60 years. The need for access to individual components during normal plant operation and under accident conditions shall be considered in developing building and component arrangements. The RCCS shall be designed to meet availability/investment protection requirements. The RCCS shall be designed to accommodate continuous operation at any power level up to 100% of rated power. Where cost effective, the design of the RCCS shall incorporate features required to implement on-line surveillance and performance monitoring. The design of the RCCS shall incorporate those features required to accomplish in-service inspection activities within the time and scheduling constraints imposed by the allotted design planned outage time. The RCCS is required to operate continuously in all plant states, including shutdown following loss of forced reactor cooling by the PHTS and SCS with simultaneous loss of pumped circulation of RCCS cooling water and a safe shutdown earthquake. All components and piping of the RCCS shall be designed against seismic loads. All components and piping inside the reactor building including the connections for emergency water supply

Functions	Requirements
	(fire brigade) are designed against external events, e.g., aircraft crash or pressure waves.
Fuel Handling System	
Remove and replace fuel from the reactor core Prepare new fuel for use in the reactor core Store spent fuel	During reactor shutdown, the FHS shall receive new and irradiated fuel, reflector blocks, and other core elements from the spent fuel storage system SFSS and place them in the reactor vessel, physically replacing and re-stacking the core. The FHS shall provide shielding to protect workers from radiation during certain fuel handling operations. The FHS shall limit the ingress of potential contaminants into the primary helium circuit from components of the FHS external to the primary helium pressure boundary. The FHS shall be designed to accomplish plant refueling within a time interval specified in planned outage allocations
Nuclear Heat Source Protection System	
Maintain plant parameters within acceptable limits established for design basis accidents	The protection system shall implement the relevant monitoring, analysis, and actuation functions which are necessary to reach the controlled state in case of abnormal events
Fuel Requirements	
Allowable [TBD] failure of fuel particle coatings at the time of manufacture Allowable [TBD] free uranium contamination in fabricated fuel	Fuel performance retention capabilities during normal operation (accounting for the failure of fuel particle coatings and, if significant, for the radio contaminants diffusion in the fuel particles). Fuel performances retention capabilities during off-normal events (accounting for any incremental failure of fuel particle coatings and, for any increased diffusion of radio contaminants in the fuel particles

4.3.2 Power Conversion System (PCS)

Functions	Requirements
Power Conversion System	
Convert heat from the PHTS into electricity for distribution on the commercial gird.	The NGNP PCS shall be connected to a local public transmission line for external distribution and sale of [276] MWe. The NGNP PCS shall produce electricity at 60 Hz. The NGNP plant electrical output shall be delivered to the operating utility at the low-voltage bushings of the main power transformer.
Steam Turbine & Generator	
Produce electricity using steam generated in the heat recovery steam generator	The steam turbine and generator shall be designed for superheated steam at [TBD-pressure] and [TBD-temperature] at the turbine throttle. The steam turbine and generator shall be designed with a high pressure, intermediate pressure, and low pressure turbine on a single shaft. The turbine shall be designed for main steam temperature variations of up to [TBD] The steam turbine shall produce [210 MW] of electricity
Gas Turbine, Compressor and Generator	
Produce electricity using the secondary gaseous fluid (nitrogen based mixture)	The turbine shall be designed to operate with the appropriate secondary gas mixture. The turbine shall be designed to operate continuously at a nominal turbine inlet temperature of [850°C] The generator rotors shall be supported by magnetic bearings.
Heat Recovery Steam Generator	
Generate steam using the transfer of heat from the secondary gaseous fluid to the main feedwater	The HRSG shall be a closed loop system. The HRSG shall be designed to a rated thermal output of [500 MW].
Main Feedwater System	
Deliver feedwater to the steam generators at the specified temperature, pressure, flow rate, and water chemistry.	Maintain feedwater chemistry within the vendor specified limits

Functions	Requirements
Provide storage to accommodate process fluid surge and volume fluctuations. Provide isolation of the feedwater to prevent water inflow to a failed steam generator.	

5.0 ASSUMPTIONS FOR HYDROGEN PLANT AND INTERFACES WITH REACTOR SYSTEMS

This chapter contains two distinct parts. The first part defines the assumptions for the interface with the hydrogen plant (Sections 5.1 and 5.2). The second part of the chapter describes an independent evaluation of high temperature steam electrolysis (Section 5.3).

As discussed previously, the AREVA NGNP preconceptual design studies work scope did not include the High Temperature Heat Transport Loop Special Study, the design of the hydrogen production plant, or the design of the high temperature heat transport loop. Therefore, assumptions must be made regarding these systems in order to define the interface conditions with the Nuclear Heat Source. This chapter very briefly describes these assumptions.

5.1 Hydrogen Production Plant Assumptions

Based on the NGNP design requirements, the hydrogen production demonstration plant is assumed to utilize both the Sulfur-Iodine thermochemical process and the High Temperature Electrolysis process.

Based on the work performed under the Primary and Secondary Cycle Concept Study, these hydrogen production processes are assumed to utilize heat at 850°C. The heat is delivered to these systems from a secondary coolant system operating at 875°C with a return temperature to the NHS of 475°C.

The impact of operational events within the hydrogen production plant on the NHS operation is bounded by full load rejection from the hydrogen plant.

The impact of NHS operating events on the hydrogen production plant was not evaluated, since the design of the hydrogen plant was not performed.

5.2 High Temperature Heat Transport Loop Assumptions

The high temperature heat transport loop is assumed to be a closed high pressure helium loop. As stated previously, this loop operates with a hot leg temperature of 875°C and a cold leg temperature of 475°C. The loop operates with pressure essentially balanced across the NHS IHX, so the operating pressure is 5MPa.

5.3 System Analysis of High Temperature Steam Electrolysis

While AREVA's scope did not include the design of the hydrogen production plant, it was agreed that AREVA should have hydrogen producers on its team perform a separate side study of a hydrogen production system using high temperature electrolysis (HTE) supplied with high temperature steam and supplemental electrical heating instead of direct very high temperature heat. This chapter also presents the results of this evaluation. The HTE evaluation that was performed to examine the overall system performance of a high temperature steam electrolysis system supplied by steam and electricity from a steam cycle HTR (or the steam bottoming cycle of the CCGT plant).

This evaluation of high temperature steam electrolysis system performance is not a part of the AREVA reference NGNP preconceptual design concept development activity.

The evaluation focused on overall system performance assuming heat was supplied using extraction steam from an adjacent steam cycle HTR and electric energy. Over a range of electrolyzer operating temperature of 600°C to 800°C, the best performance was predicted at 600°C. This result is based on the overall system performance, not just the electrolyzer. The details of the evaluation are described in Appendix D.

6.0 REACTOR AND PRIMARY SYSTEMS

6.1 Reactor and Core Systems

The reactor and core systems are the high temperature nuclear heat source of the VHTR. It includes the core comprised of the graphite blocks, the non-metallic reactor internals (the replaceable and permanent reflector elements, the core support structure), and the surrounding metallic Reactor Internals (the core support, the core barrel, the upper core restraint elements, and the upper plenum shroud). The Reactor System also includes the neutron control assemblies and related equipment.

6.1.1 Reactor Core

In order to minimize the project risk, the design is based on similar components and technology successfully demonstrated at Fort Saint Vrain in the seventies and eighties. Moreover, as AREVA had been participating in the GT-MHR project until year 2000, the reference core is largely taken from the GT-MHR as a starting point.

6.1.1.1 Core Structure

The core is located within the permanent graphite reflector enclosed in a metallic shroud, the core barrel, inside the reactor vessel (cf. 6.1.2 reactor internals). As compared to past realizations, the real innovation lies in the configuration of the active core, which forms an oblong hollow cylinder as opposed to a solid cylinder as usually found in other reactor systems. Thus, the core structure basically consists of an annular arrangement of prismatic fuel and reflector blocks, both replaceable by means of a fuel handling machine (see Section 7.4). Figure 6-1 shows a view of the core structure. The overall arrangement of elements stacks is shown in Figure 6-2.

The annular core configuration is adopted to achieve a maximum power rating and still permit core decay heat removal while maintaining an acceptable fuel temperature distribution.

The subsequent main core dimensions are:

- outer equivalent diameter of annular core zone : 4.84 m
- inner equivalent diameter of annular core zone : 2.96 m
- height of fuel element stack : 8.00 m

These dimensions along with a power density of about 6.1 MW/m3, provides a power of 565 MW thermal.

The annular active core arrangement is built of 1020 Fuel Elements (FEs) stacked up in 102 columns of 10 FEs each, roughly making up 3 rings of 30, 36 and 36 columns, respectively (Figure 6-1 and Figure 6-2). Among them, 30 columns have a dedicated channel for the introduction of an absorber material, 12 start-up control rods and 18 for reserve shutdown system control material. The other 72 columns do not comprise such a channel.

The top and bottom reflectors are arranged above and below the FEs arrangement, respectively. Replaceable reflector columns are also arranged in the inner and outer zones of the core annulus, with 61 and 102 columns respectively. Among the 102 columns of the outer reflector arrangement, 36 have a dedicated channel for the introduction of a control rod.

The hexagonal pitch is 362 mm at room temperature (20°C) including a radial gap between stacks of 2 mm.

Reflector stacks are axially shifted by 400 mm (half an element height) with respect to fuel stacks.

A numbering system uniquely identifies each fuel or reflector element location in the core.

The main characteristics of the reference core design are summarized in Table 6-1: Main Characteristics of the Reference Core Design.

6.1.1.2 Prismatic Block

The basic core element is a hexagonal graphite block of 360 mm external flat to flat width. It is based on the Fort Saint Vrain design. Its main features are listed:

- There is a 35 mm diameter and 206 mm depth hole in the center of the top face for coupling with the fuel handling machine grip. The hole has a ledge where the grapple engages.
- There are guide sockets and guide dowels on the bottom face and top face, respectively. They are intended to provide alignment for refueling and coolant channels, and transfer of mechanical loads from fuel elements. Their number and location depend on the type of assembly. The details of the guide dowel and socket will be defined in the next project phase.
- The top and bottom edges of the elements are beveled to assist insertion into the core.
- Graphite candidates are the same as those envisioned for the reactor internals (see Section 6.1.2.3).

There are different types of blocks in the core structure depending on their function and location:

- The fueled ones which contain blind holes for fuel compacts and full length channels for helium coolant flow. They have an 800 mm height. There are 1020 such fueled blocks in the core among which 300 have a large channel for absorber motion. Sketches of the fuel blocks are shown in Figure 6-3.
- The non-fueled ones are graphite blocks which are located in the radial and axial reflector regions surrounding the core. They have different design features and height depending on their location.

6.1.1.3 Core Reactivity Control

The core reactivity is controlled by the core negative temperature coefficient and control rods, and possibly by lumped burnable poison located in the fuel assemblies. It is also complemented by the Reactor Reserve Shutdown System (RRSS). This system is used to shutdown the reactor and maintain it a sub-critical state if the rod system fails to trip the reactor.

Details on neutron control are provided in Section 6.1.3.

6.1.1.4 Reactivity Balance

The core reactivity balance is presented in Table 6-2 for Beginning of Cycle (BOC) and End of Cycle (EOC) and includes the following items:

- Reactivity due to equilibrium xenon.
- Temperature reactivity effect (Doppler, moderator, and reflector) – their sum represents the cold to hot transition.
- Reactivity due to burn-up, which is the excess reactivity required to achieve cycle lifetime.
- Control rod worths.

The sum of the xenon worth, the total temperature reactivity effect, and the burn-up reactivity yields a BOC required control rod worth of 19.2 %Δk/k. The total available worth is 24.9 %Δk/k, which is sufficient to cover stuck rod worth and shutdown margin.

6.1.1.5 Performance

6.1.1.5.1 Operating Parameters

As regard to the core performances, the most important parameters are the core inlet and outlet temperatures, the core radial power and flow distributions, and last but not least the total core power. These parameters are important because they affect the fuel temperature in normal operating and/or accidental decay heat removal conditions, and consequently the overall fuel performance. As a consequence, the definition of the operating parameters (mainly core inlet and outlet temperatures and core power) is performed through the verification of satisfactory fuel solicitations in normal and accident conditions.

6.1.1.5.2 Normal Operation

For normal operation, the important parameters are the core inlet and outlet temperatures. Their effects are such as: increasing the core outlet temperature or decreasing the core inlet temperature increases the peak fuel temperature. The core inlet temperature affects also the reactor vessel material and a trade-off has to be found between the peak fuel temperatures and the reactor vessel temperature. The magnitude of these effects is also dependent on the actual radial peaking factor and core bypass flow fraction: the larger these parameters, the larger the sensitivities to inlet and outlet temperatures.

6.1.1.5.3 Accident Conditions

For accident conditions, the selected reference transient has been the Depressurized Conduction Cool-down (DCC). The temperature limit of 1600°C, based on past HTR experience, has been considered as the maximum allowed peak fuel temperature during the transient. In addition, consideration has been given to the evolution of the fuel temperature distribution over the core volume during the transient.

For the accident decay heat removal condition (conduction cool-down), the total core power, or power density, is the key parameter that dictates, for a given core geometry, the peak fuel temperature reached during the transient. This is an important feature of the HTR design, because it means that the DCC transient finally determines the reactor power level.

Beyond the initial power density level, the fuel temperature response depends on the capability of the core structure to transfer the residual heat to the reactor vessel. Even if the DCC transient involves relatively simple physical thermal phenomena, conduction and radiation, the proper assessment of the fuel response requires knowing essential input data:

- First, the heat source: the precise time evolution and spatial distribution of the residual heat during the transient.
- Second, the heat path: the geometry and the thermal properties of the core structure and their possible dependence to temperature during the transient.
- Third, the initial conditions at the beginning of the transient, mainly the temperature field in the core structure.

Thus it is the knowledge and implementation of those input data that form the heart of the DCC assessment. Consequently, this transient has been investigated during the Pre-Conceptual Design Phase and numerous parametric studies have been performed to assess the sensitivity to parameters which play a role in the assessment:

- Initial power level
- Thermal conductivity of graphite reflectors

- Residual power
- Emissivity of the metallic internals

Calculations have been carried out with the following nominal parameters:

- Power level: 565 MWth
- Core inlet temperature: 500°C
- Core outlet temperature: 900°C
- Operating pressure: 5 MPa
- By-pass: 5%

Figure 6-4 gives, for the base case, the temperature evolution during the depressurized conduction cool-down accident. Table 6-3 provides a summary of peak fuel temperatures for the different cases studied (base case and 4 cases where parameters are set to the value corresponding to 95% confidence limit, according to an uncertainty study). It can be noted that calculated temperatures are below 1600°C.

6.1.1.6 Nuclear Design

The core nuclear design includes numerous design aspects: fuel assembly design, equilibrium cycle definition, reactivity coefficients, reactivity management and rod worth, fuel cycles, etc. The aspects that directly affect the selection of the main reactor features (core inlet and outlet temperatures, power level and core external dimensions) have been addressed.

6.1.1.6.1 Power Distributions

The core radial power distribution is important because it directly affects the fuel performance (temperature) in normal operation. Indeed the active core annulus is facing two lateral graphite regions, which generate important thermal fluxes at the core/reflector boundary causing local power peaks in the fuel compact rows close to the graphite reflectors. Subsequently the helium flowing in the neighboring channels is over heated as compared to that flowing deeper in the active core. This effect is expressed through the Radial Peaking Factor (RPF) which is the ratio between the highest power in the core and the average core power. A typical RPF value is 1.3 to 1.5. The dependence of the peak fuel temperature on the RPF is such that: (a) the higher the RPF, the higher the peak fuel temperature, (b) the higher the average core temperature rise, the higher the dependence on the RPF. To illustrate the dependence, reducing the RPF from 1.5 to 1.3 typically decreases the peak fuel temperature by more than 100°C.

As a result a main line of investigation in the core design optimization is the minimization of the RPF, by power flattening. At the fuel assembly level, a study has shown that acceptable radial power peaking can be obtained by the use of lumped burnable poison compacts at key points along the fuel/reflector interface. At the core level, additional benefits are realized from an optimization of the fuel loading scheme.

The core axial power distribution is also important because it determines the distribution of the decay heat power and thus affects the fuel temperature response in case of a conduction cool-down accident transient. Axial power distribution optimization is realized using similar techniques.

6.1.1.6.2 Power Stability

Spatial power oscillations are a concern for large power reactors. This was a concern in sizing the current generation of light water reactors (LWR). The damping of power oscillations depends on reactor characteristics such as the core composition, dimensions, and power density. For the graphite moderated reactor the neutron mean free path is much larger than for a LWR of the same power level. From a neutronic perspective the graphite moderated reactor is much smaller than an equivalent LWR. Analyses have shown that an 8 m prismatic reactor core with a power level lower or equal than 600 MWt is stable relative to xenon oscillations.

6.1.1.6.3 Reactivity Analysis

Another important aspect is the fuel cycle length with respect to the plant availability. Long fuel cycle lengths are favored to minimize the impact of refueling outages. A current target is to reach at least 18 months with as much flexibility as possible. Analysis has shown that sufficient reactivity is available to achieve this target with an initial uranium enrichment of about 14%.

Additional reactivity analyses were performed for xenon reactivity worth and control rod worth for the start-up and operating rods. It was established that sufficient rod worth is available to overcome the xenon reactivity, the cold-to-hot temperature reactivity effect, and the excess reactivity for cycle lifetime.

6.1.1.7 Thermal-Hydraulic Design

6.1.1.7.1 Core Bypass Flow

The major issue of the thermal-hydraulic design is the core bypass flow. It is directly related to the core thermal performance. In the core, the flow partitions itself among the coolant channels, the absorber element channels and the gaps between the columns of fuel and reflector blocks. The objective in the core flow design is to maximize flow through the coolant channels (which directly flow to where the power is being produced). This means minimizing flow through the gaps between columns, and limiting the flow in the absorber element channels to that needed to cool the absorbers when they are inserted.

Refined analyses will need to be performed in the frame of the Conceptual Design to assess the value of core bypass flow and propose design improvements to minimize the bypass.

6.1.1.7.2 Core Pressure Drop

It has been confirmed that the total pressure loss from the core inlet to the outlet collector is limited (lower than 0.5 bar). It does not seem that the expected value should be an issue for the circulators design.

6.1.1.7.3 Other Issues

Flow induced vibrations (FIV), fluid structure interactions (FSI) and acoustics are issues that will need to be addressed in the subsequent phases of the project.

Table 6-1: Main Characteristics of the Reference Core Design

Parameter	Value
Power Level	565 MW Thermal
Primary Pressure	5 MPa
Core fissile Height	8.00 m
Equivalent Inner Diameter	2.96 m
Equivalent Outer Diameter	4.84 m
Hexagonal Pitch	362 mm
Number of Fuel Block Columns	102
Fuel Blocks per Column	10
Core Inlet Temperature	500°C
Core Outlet Temperature (mixed)	900°C
Helium Flow Rate	272 kg/s
Fuel Type	Triso coated particles (UCO or UO_2)
Fuel Enrichment	< 20%
Fuel Burnup	< 20% FIMA
Equilibrium Cycle Length	18 months

Table 6-2: Reactivity Effects

Parameter	Reactivity (%Δk/k)	
	BOC	EOC
Xenon equilibrium	2.8	3.1
Temperature effect		
- Total	3.5	2.8
- Doppler	4.1	3.9
- Moderator	1.8	1.6
- Central reflector	-1.5	-1.6
- Side reflector	-0.9	-1.1
Margin due to burnup	12.9	2.0
Required rod worth	19.2	7.9
Rod worths		
- 12 start-up rods	7.8	8.8
- 36 operating rods	8.7	11.0
- all rods	24.9	30.2

Table 6-3: Peak Fuel Temperatures During Depressurized Conduction Cool-Down Accident

Case description	DCC Peak Fuel Temperature (°C) / Time (h)
Base case	1513 / 93.5
Power level 588 MW	1544 / 93.5
Reduced reflector conductivity	1563 / 101
Residual Power +10%	1588 / 94.5
Reduced steel emissivity	1528 / 98

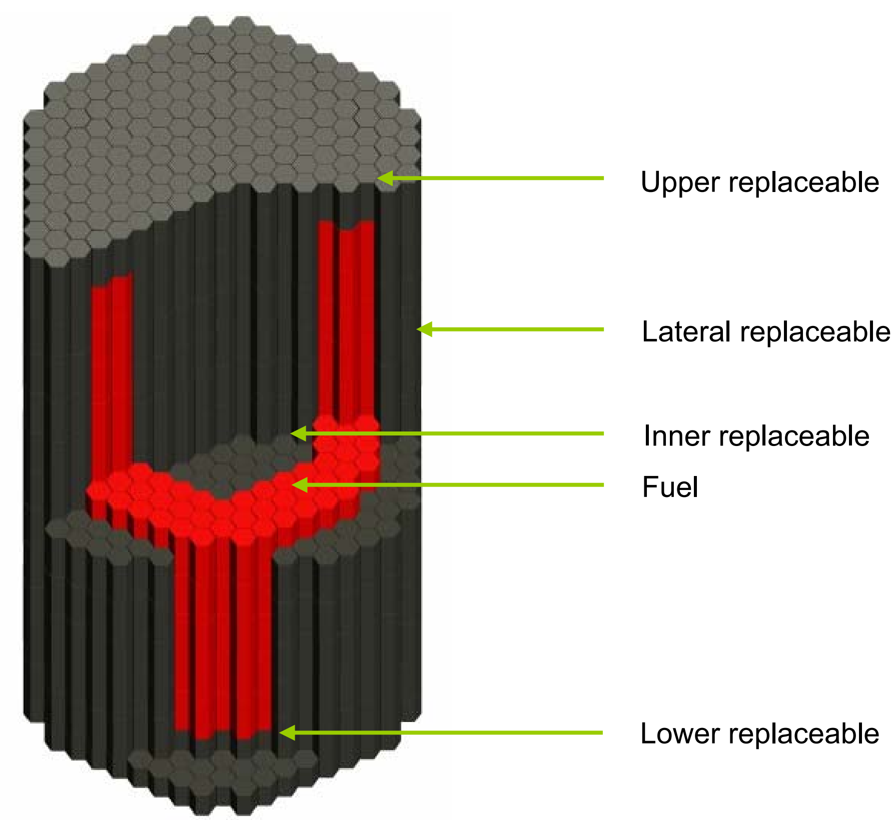


Figure 6-1: Core Structure

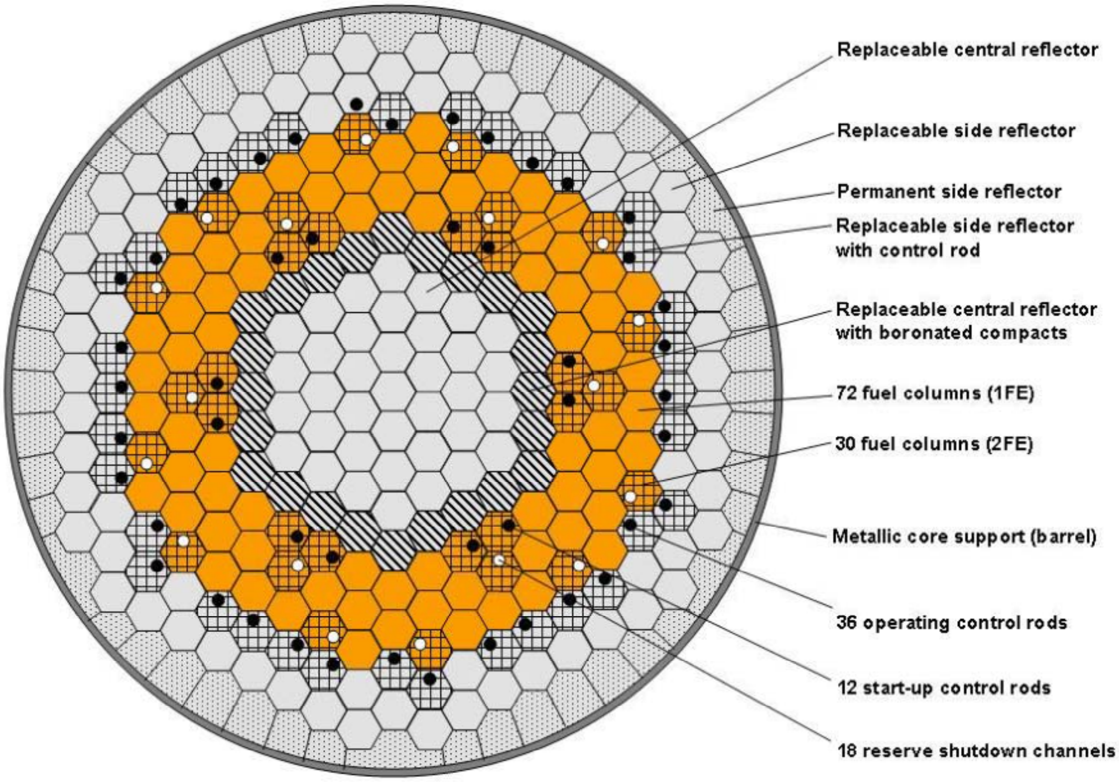


Figure 6-2: Core Layout

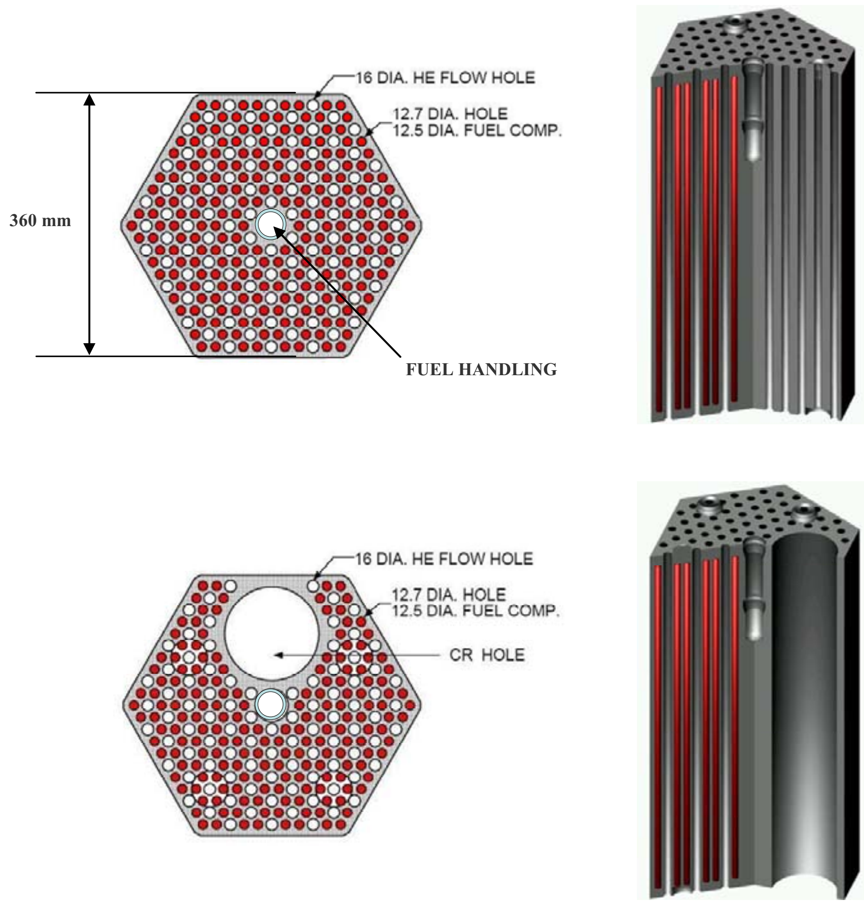


Figure 6-3: Main Features of the Fuel Assembly

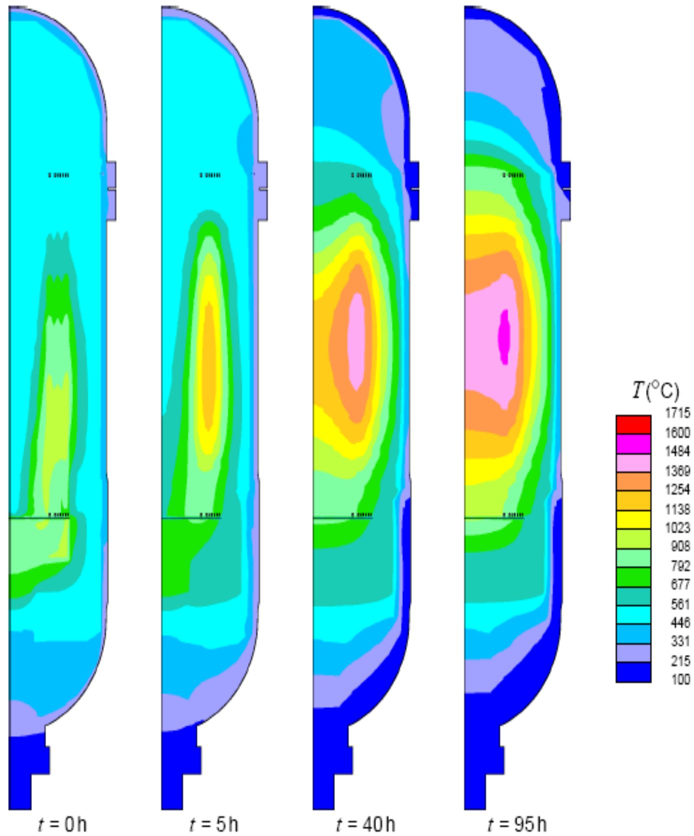


Figure 6-4: Temperature Evolution During Depressurized Conduction Cool-Down Accident

6.1.2 Reactor Internals System

Reactors Internals System (RIS) is part of the Reactor System. It includes those components which provide the structural interface between the Reactor Core System and the Vessel System and those components which provide for routing of helium between the Reactor Core System and the Primary Heat Transport and Shutdown Cooling Systems. The RIS further includes the primary coolant helium transiting the system.

The RIS comprises five major subsystems as follows:

- An Upper Core Restraint (UCR) assembly that limits the movement of the replaceable reflector/fuel columns at the upper end of the reactor core, while providing for the flow of primary coolant into the active core.
- A Permanent Side Reflector (PSR) that surrounds the lateral boundary of the core and provides for transfer of lateral loads, neutron reflection and shielding.
- A Graphite Core Support Structure (GCS) that supports the PSR and core/replaceable reflector assembly at their lower ends and provides for collection and channeling of primary coolant to the Hot Duct Assembly (HDA).
- A Metallic Core Support and Core Barrel (MCS) structure that contains and supports the core/replaceable reflector assembly, plus the UCR, PSR and GCS, above, and which transfers axial and lateral loads to the Vessel System.

- A Top Plenum Shroud (TPS) structure that, along with the MCS, forms an enclosed plenum at the top of the core.

The overall arrangement of the NGNP preliminary reactor design is shown in Figure 6-5.

The MCS contains and supports the GCS, PSR, UCR and reactor core. Along with the TPS, it forms the upper, inlet plenum to the reactor core.

The UCR, PSR and GCS bound the reactor core on the top, sides and bottom, respectively. In conjunction with the MCS, the UCR, PSR and GCS provide lateral and vertical restraint. They also establish the location of the core with respect to the reactor vessel and, thereby, with the reactor control and refueling components that are mounted on the reactor vessel top head.

The MCS and GCS are in interface with the hot gas duct (see section 6.3.4) to direct the flow of core outlet helium to the inlet of the intermediate heat exchangers.

In addition to providing structural support and directing the flow of helium, major functions of the RIS include thermal and radiological shielding of the reactor vessel, conservation of neutrons during power production and serving as a key link in the Reactor Cavity Cooling System heat transport path.

While designed to be removable in first approach with varying degrees of difficulty, the RIS components are designed for the full plant lifetime of 60 years.

6.1.2.1 Non-Metallic Internals

6.1.2.1.1 Upper Core Restraint

The upper core restraint (UCR) subsystem comprises an assembly of interlocking pieces. Their arrangement is such that each hexagonally shaped component of the UCR interfaces with the dowels provided on the upper surfaces of three adjacent reflector elements, effectively fixing their positions relative to each other at the top end. The hexagonal components also interlock with each other to form a semi-rigid structure over the top of the Reactor Core components.

The hexagonal UCR components include through-thickness holes to channel helium coolant flow from the upper core inlet plenum to the replaceable reflector elements above the active core.

The material of the UCR should have a similar thermal expansion coefficient as the graphite. This material could be a carbon composite in order to accommodate the thermal expansion and to withstand mechanical loadings.

In the proposed design, the UCR connects the graphite columns to the PSR graphite ring. However the UCR is not linked to the upper part of the core barrel. The mechanical restrain of the core will be performed by means of metallic inserts fixed to the core barrel.

6.1.2.1.2 Permanent Side Reflector

The permanent side reflector (PSR) subsystem serves as the radial transition interface between the exterior of the hexagonal Reactor Core System components (fuel elements and replaceable reflectors) and the interior of the circular core barrel. Axially, the PSR extends from the lower side of the UCR to the top of the underlying components of the graphite core support.

The PSR design comprises an array of graphite blocks of varying size and shape. The blocks are keyed vertically with dowels and are overlapped to minimize helium bypass flow.

The outer blocks are radially keyed to the core barrel inner surface by means of metallic inserts. This arrangement allows radial and axial growth, but prevents movement in the azimuthal directions. A functional gap shall be foreseen for the assembly with the graphite PSR and to take into account the differential expansion.

The PSR blocks are equipped with a handling hole and could be removed from the reactor in the same way as the replaceable reflectors and fuel assemblies.

Overall, the PSR limits movement of the reactor core and transfers structural loads (notably seismic) between the replaceable reflector elements that surround the active core and the core barrel.

As suggested by its name, another key function of the PSR is to reflect and conserve neutrons in order to facilitate the nuclear reaction for heat generation. The PSR in conjunction with the core barrel is designed to minimize helium bypass of the reactor core. It also provides radiological shielding. A subset of the graphite blocks contains B4C materials for shielding purposes to limit neutron fluence seen by the reactor vessel.

6.1.2.1.3 Graphite Core Support Structure

The Graphite Core Support (GCS) subsystem is a graphite and ceramic structure that is contained within the metallic core support and located below the PSR and the bottom replaceable reflector elements of the Reactor Core System. The GCS (Figure 6-6) incorporates the following major features:

- The permanent bottom reflectors constitute a transition region below the PSR and the bottom replaceable reflector elements that collects and channels the helium exiting the core to the core outlet plenum.
- The core outlet plenum collects the helium flow from the transition region and directs it to the hot duct that channels the flow to the IHX
- The lower floors function is to protect the metallic core support from the neutrons and from the high temperatures.
- Large outlet openings on the side of the GCS that interfaces with the hot gas ducts.
- Passages through the lower floor that provide for flow to the Shutdown Cooling System (SCS) interface.

Permanent Bottom Reflectors

The permanent bottom reflectors consist in 2 rows, one is devoted to mix the flow at the outlet of the core and the other is devoted to provide a closure of the gaps between blocks. The protection against neutrons will be ensured by inserting B4C rods inside the large blocks.

First row: the mixing blocks (360 mm across flats) are constituted, in their upper part, of channels of the same diameter as the flow channels of the core assemblies and, in their lower part of 3 large channels, symmetrically distributed in the section. The relative heights of those 2 parts depend on the machining feasibility and on the flow hydraulics.

Second row: it is made up of larger blocks (624 mm across flats), in order to reduce the by-pass flow and increase the stability of the core. The flow coming from the mixing blocks is directed towards a corner of a block. The channels are therefore connected to the coolant channels of the mixing blocks at the upper end and the channels of the core posts at the lower end. The diameter of the large channels for the coolant passage is defined in order to take into account the pressure drop and mixing considerations.

An engagement hole can be provided for handling purpose. Dowels/sockets ensure the vertical alignment of the blocks.

Core Outlet Plenum

The structure of the outlet plenum is an important consideration in the design of the GCS, which must transfer axial loads from the PSR and core assemblies above to the metallic core support structure below. The plenum is formed using an array of support posts between the upper transition floor and the lower insulating layer. The arrangement of the posts and transition blocks is such that the axial load from any given fuel/replaceable reflector column is shared by multiple posts.

The support of the core is therefore achieved by cylindrical solid posts for which the upper and lower ends have a hexahedral section with a flat-to-flat size of 360 mm. The central part of the columns is defined as a preliminary result as a 160 mm diameter cylinder. The length of the posts depends on the hot duct diameter. The flowing into the outlet plenum is achieved through holes drilled at the angles of the hexahedral section of the support posts.

Lower Floor

The lower floor protects the metallic plate located below against neutron and high temperature. The height of the first row shall be enough to protect the metallic parts against neutron. 400 mm is retained as a preliminary data. A smaller height is favorable as it will allow reduction in the size of the vessel bottom head. Large blocks, 624 mm across flats, are used. They are linked to each other by means of rectangular keys, the objective being to make the core more stable and to give to the whole GCS an overall cohesion.

The thermal protection of the MCS is provided by the ceramic plates located below the graphite. Its thickness shall be defined according to the efficiency of the material selected. A 100 mm thick plate is assumed in the preliminary design. Dowels/sockets ensure the vertical alignment of the lower floor blocks with the ceramic layer and of the lower floor blocks with the core support posts above.

6.1.2.2 Metallic Internals

6.1.2.2.1 Metallic Core Support and Core Barrel

The metallic core support and core barrel (MCS) subsystem establishes the proximate interface between the reactor vessel and the components of the RIS and Reactor Core System contained therein. The MCS is a complex cylindrical metallic structure that, in the present design, is manufactured in four packages and assembled on site. The upper part of the structure (core barrel), which comprises three of the four sections, is open at the top (where it supports the top plenum structure) and is attached to the metallic core support structure at the bottom. The core barrel section of the MCS is of double wall construction. The inner cylinder serves as the principal structural member of the core barrel assembly.

The outer cylinder is provided to establish an annular passage for helium flow. Flanges are incorporated at either end, which provide for stiffening, mounting of the outer cylinder shell and bolting of the four sections together in the field. The maximum length of the sections is based on materials supply (available length of ring forgings) and transportation considerations. The three core barrel sections are designed to be bolted to each other and to the metallic core support structure below.

The metallic core support structure at the lower end of the MCS serves both as an axial support structure and flow channeling device. The vertical shell of the support structure incorporates two concentric annular flow channels. The outer channel communicates with return (cold) helium flow from the cross vessel annulus surrounding the hot duct. This flow is directed to the lower plenum of the support structure. From there, the flow enters the upper plenum of the support structure via holes in the metal plate that separates the two plena and through an annular opening at the exterior of the vertical cylindrical shell that encloses the interface with the SCS Heat Exchanger. From the upper plenum, the flow enters the inner annular passage of the vertical shell, which communicates with the core barrel flow annulus above. The multi-plenum structure of the metallic core support is designed to provide uniform flow distribution of helium around the circumference of the structure at its interface with the core barrel at the upper end.

6.1.2.2.2 Top Plenum Shroud Structure (TPS)

The top plenum shroud (TPS) structure is connected to the top end of the MCS to form the inlet plenum of the reactor. The TPS comprises an inner structural shell with external stiffening ribs. An outer shell is provided to contain the combined thermal and radiation shielding material that is included in the enclosed space within.

Penetrations are provided for the neutron control assemblies (NCAs) during normal operation and/or refueling and fuel manipulation equipment and other maintenance equipment that is utilized during outages.

In normal production, the main helium flow exits the core barrel annular channel into the plenum formed by the MCS and the TPS. Flow outside the TPS is essentially stagnant to reduce parasitic heat losses. When the SCS is in operation, helium flow is on the outside of the MCS and enters the upper plenum via the clearances between the TPS penetrations and the NCAs.

6.1.2.2.3 Stagnant Areas and Core Support Cooling

Three zones are concerned, the bottom area delimited by the core support line, the external area around the core barrel and finally the top zone above the upper plenum shroud. The two last zones are directly communicating.

The feeding of the peripheral reactor vessel bottom is ensured by 3 holes in the lower plate of the core support structure. These holes allows to cooling the core support during SCS start-up (cold shock).

Forty holes in the core barrel upper part ensures the feeding of the area above the top plenum shroud.

In order to ensure the communication between the vessel bottom and the peripheral stagnant zone above the supporting line area, openings are proposed at the level of the MCS support ring.

6.1.2.2.4 Material Selection

Material envisioned for the metallic components is alloy 800H. This material is already codified at high temperature and provides higher flexibility to cope with high temperature transients.

Graphite grades selected so far are the following:

- PCEA from GrafTech
- NBG17 or 18 from SGL

The selection was made taking into account different fabrication processes and different coke types.

NBG17 (Vibromolded, pitch coke) is thought to have more uniform properties and should have better behavior under irradiation compared to PCEA. Both have similar grain sizes and could be used for similar application (fuel elements, reflectors, etc).

NBG18 is close to NBG17, except that the grain size is larger. NBG18 should enable fabricating larger graphite components (in particular larger blocks of the core support structure).

For Ceramic materials, the preference would go to C/C composites which are considered as more mature technology.

6.1.2.3 Design Code

The metallic components of the Reactors Internals System have to be designed according to the ASME Code Subsection NG and Code Case N-201-4 for elevated temperature service.

Rules for nonmetallic materials are presently under preparation in the context of the ASME Subgroup on Graphite Core Components

Figure 6-5: Reactor and Core System Arrangement

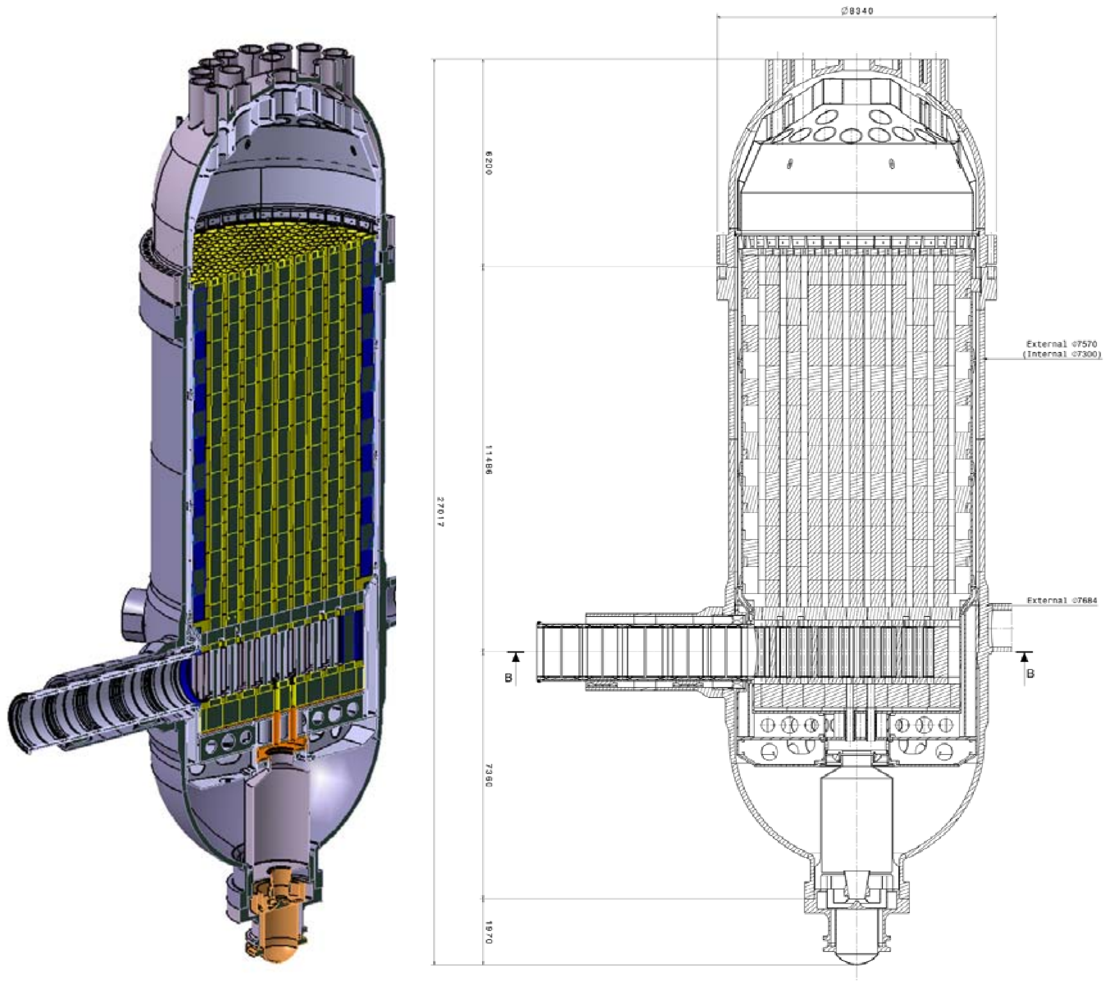
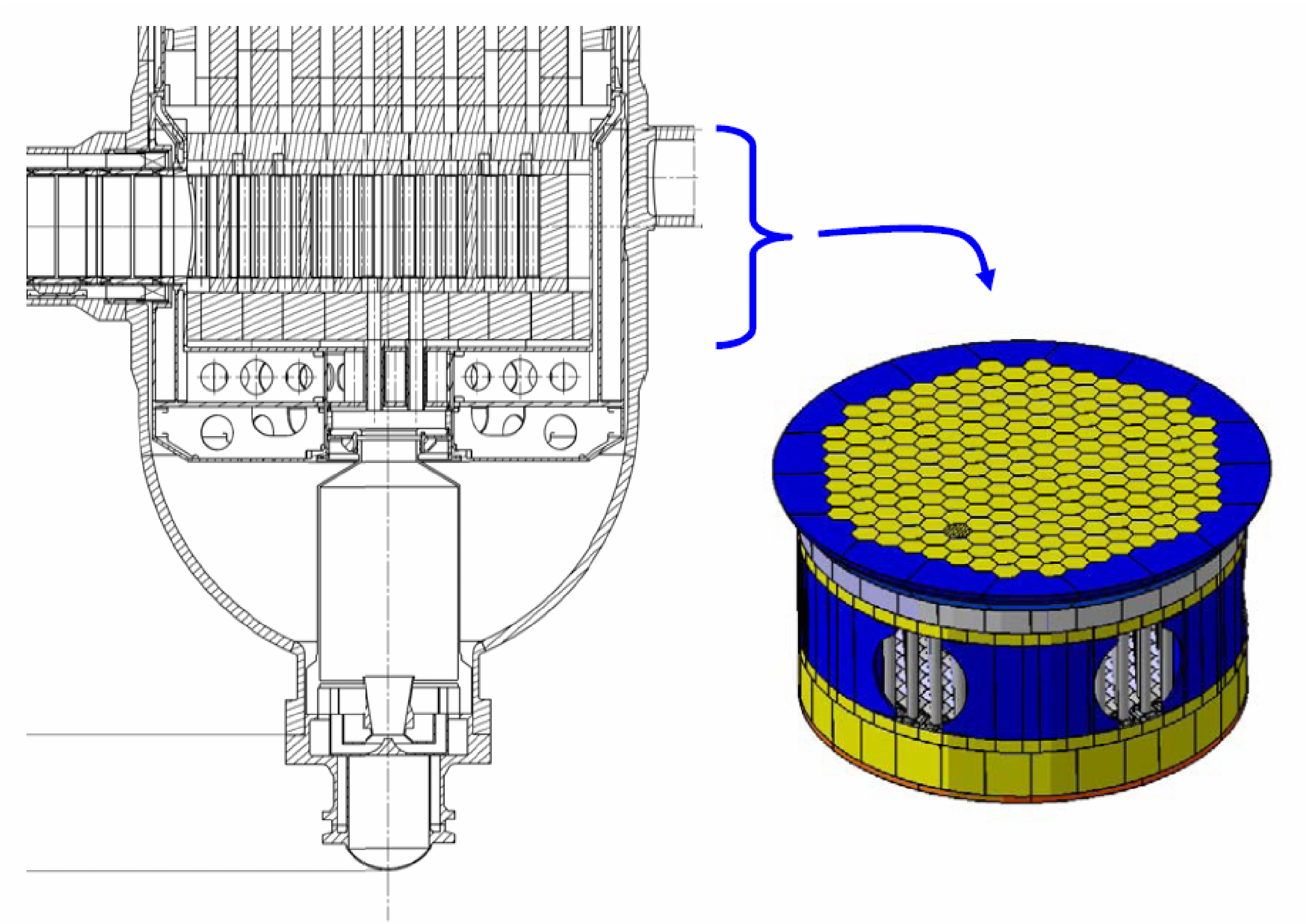


Figure 6-6: Graphite Core Support Structure



6.1.3 Neutron Control

The core reactivity is primarily controlled by the core negative temperature coefficient and control rods. In addition, the placement of fuel blocks having known reactivity based on burn-up (for irradiated fuel), initial enrichment levels and the possible inclusion of burnable poisons provide for further control of reactivity. Core reactivity control is complemented by the Reactor Reserve Shutdown System (RRSS) that will safely shutdown the reactor and maintain a subcritical state in the event that the control rods fail to operate during accident conditions.

6.1.3.1 Control Rods

The control rods are split into two categories: a) 36 operating control rods located in the inner ring of the outer reflector, and b) 12 start-up control rods in the inner ring of fuel columns (see Figure 6-2). The control rods are clustered in groups of two, with each group sharing a Neutron Control Assembly (NCA). The operating control rods are inserted to varying heights for control during reactor operation and fully inserted for protection. The 12 start-up control rods are only inserted during shutdown and refueling operations; they are fully withdrawn whenever the core is critical. With the start-up and operating control rods inserted, a 1% $\Delta\rho$ shutdown margin can be maintained under cold conditions.

The control and start-up rods have the same design. While the design details and materials have not been finalized, the selected design concept consists of annular compacts containing a neutron absorber material, enclosed in carbon-carbon (C/C) composite canisters connected to each other. A control rod element, illustrated in Figure 6-7, is comprised of 18 canisters assembled into a flexible string that is designed to facilitate full insertion of the control rod into the core for all normal and accident conditions, including postulated fuel block offsets that might occur during seismic events. Heat removal from the annular compacts is accomplished by flowing cold helium through the central hole within the compacts and through the annulus between the canisters and the control rod channel within the graphite blocks.

The reference design uses annular absorber elements fabricated from a mixture of B4C granules and graphite matrix. However, alternative absorber materials may be considered in the Conceptual Design Phase. The use of C/C composite canisters, as opposed to Fe/Ni alloys typically used for control rods (such as 800H), is dictated by the high temperatures to which the control rods will be exposed. Temperatures during normal operating conditions could range between 700ºC to 1100ºC depending on the control rod channel cooling strategy. A reasonable assumption for the preconceptual design would be an operating temperature of 1000ºC. Under accident conditions, control rod temperatures could reach 1600ºC in the fuel blocks. C/C composites are relatively mature materials and are widely used across many industries. However, their use as a control rod is not proven, particularly with respect to the impact of irradiation on the mechanical and physical properties. As a result, a significant development program will be required to fully develop and qualify C/C composites for control rods. It should be noted that AREVA, in collaboration with leading composite suppliers and the CEA have already initiated a development program that should allow selection of the final material (with a complete database) for control rods by the end of 2012. This program will also consider C/SiC and SiC/SiC composites, but it is unlikely that these materials will be ready in the required timeframe.

6.1.3.2 Neutron Control Equipment

Neutron control is effected using equipment for positioning the control rods and nuclear instrumentation. The primary components include Neutron Control Assemblies (NCA) and nuclear instrumentation. There are 24 NCAs, of which 18 are used for the 36 operating control rods in the outer reflector and 6 are used for the 12 start-up control rods in the inner core. Each NCA contains 2 independent chain, wheel, gear and motor type control rod drives – one per control rod. A friction clutch between each motor and the drive mechanism is included to prevent overload.

The nuclear instrumentation consists of ex-vessel neutron detectors, source range detectors, and in-core flux mapping units. During normal operation, the neutron flux levels are monitored by the ex-vessel neutron detectors, whose range overlaps with that of the source-range detectors. During startup and shutdown, the neutron flux levels are monitored using the source-range detectors. The in-core flux mapping units are used to verify axial flux profiles and confirm power stability.

6.1.3.3 Reactor Reserve Shutdown System (RRSS)

The Reactor Reserve Shutdown System (RRSS) provides for shutdown of the reactor independently and diversely from any other installed safety system and maintains the core in a subcritical state if the control rod actuation system fails to operate properly. The RRSS is actuated by the operator either by an electrically driven device or manually. The reactor is shutdown by means of spherical absorber elements that are dropped into the core and into the RRSS channels of a designated group of fuel block under gravity driving force.

The arrangement of the RRSS channels in the core is shown in Figure 6-2. There are 18 RRSS channels, 12 located in the outer ring of core blocks and 6 located in the central ring of core blocks. Each channel is serviced by a RRSS feed mechanism that is associated with the NCA units. The mechanism includes a loading tank (hopper), absorber elements, electromagnetic and manual gate drives, gate position sensors, and a guide tube. The RRSS loading tanks are located in stand-pipes within the NCA on the reactor vessel upper head. Absorber elements are stored inside the loading tanks. A gate located at the bottom of the tank is closed during reactor operation, and is opened when the RRSS is deployed, thereby allowing the absorber elements to drop into the core under gravity. At the bottom of each tank, the nozzle and a guide tube attached to the top reflector insures that absorber elements will be inserted into the core under all conditions.

Each loading tank contains about 105 kg of 6 to 7 mm diameter spherical absorber elements. The absorber elements contain B4C and graphite, and are formed into spheres onto which a protective coating is applied. The coating insures corrosion resistance of the elements in the coolant environment for the entire service life and prevents loss of boron during high-temperature abnormal events.

6.1.3.4 Neutron Control During Accident Conditions

The detection of reactivity insertion events leads to reactor shutdown by automatic insertion of the control rods by the Reactor Protection System (RPS). In cases where the events are coupled with a loss of electrical power, the controls rods will drop into the core by gravity. The RRSS can be manually actuated to achieve a diverse method of reactor shutdown, should control rod insertion not be accomplished. The two neutron absorbing systems are designed so that the insertion of either one of these systems ensures and maintains a subcritical state in all conditions. This includes, in particular, the reactivity due to the core cooling down to the coldest shutdown state combined with the xenon effect and the reactivity insertion due to the initiating event.

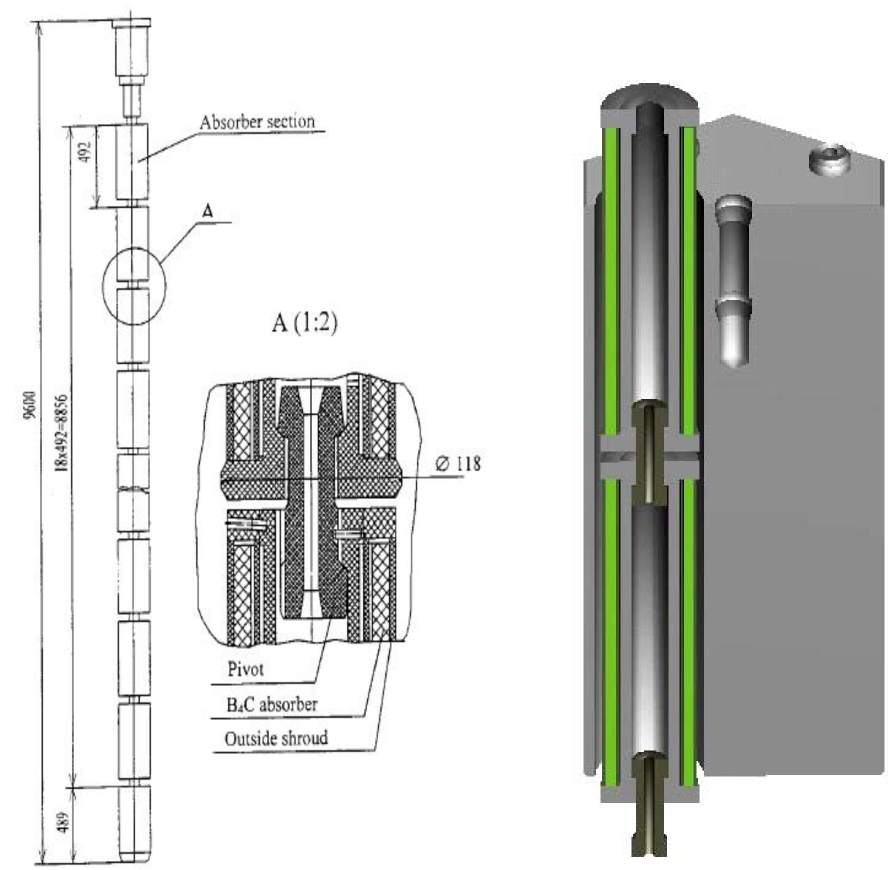


Figure 6-7: Control Rod Element

6.2 Vessel System

The Vessel System is composed of the vessels and supporting devices of the primary pressure boundary. This system is divided into the following subsystem:

- The Reactor Vessel
- The intermediate heat exchanger vessels (3 tubular IHX vessels and 1 compact IHX vessel).
- The cross-vessels (one for each IHX vessel)

The vessels are designed to contain the heat transport medium (helium) inventory within a leak tight pressure boundary and to maintain the integrity of this pressure boundary.

These vessels house and support the components of the Reactor Core, Reactor Internals, and the components of the Primary Heat Transfer System.

The Reactor Vessel and the IHX vessels are located in separate underground silo-type containment buildings and are interconnected by the cross-vessels, also located underground.

The following sections give a description of the different vessel types. The arrangement of the Reactor Vessel, cross vessels and IHX Vessels is shown in Figure 6-8.

The material of the vessels is mod 9Cr1Mo steel. The selection of this material is discussed in Section 6.2.4.

6.2.1 Reactor Vessel

6.2.1.1 Description

The Reactor Vessel (see Figure 6-9) is approximately 25 meters high, 7.5 meters in diameter and 150 mm thick in the core belt line region.

The upper closure head provides penetrations for the neutron control rod drives and fuel handling system. The closure head sealing device is ensured by means of 80 studs and a principle of metallic gaskets based on PWRs experience. For that, two concentric gaskets are fastened inside grooves machined into the top head flange.

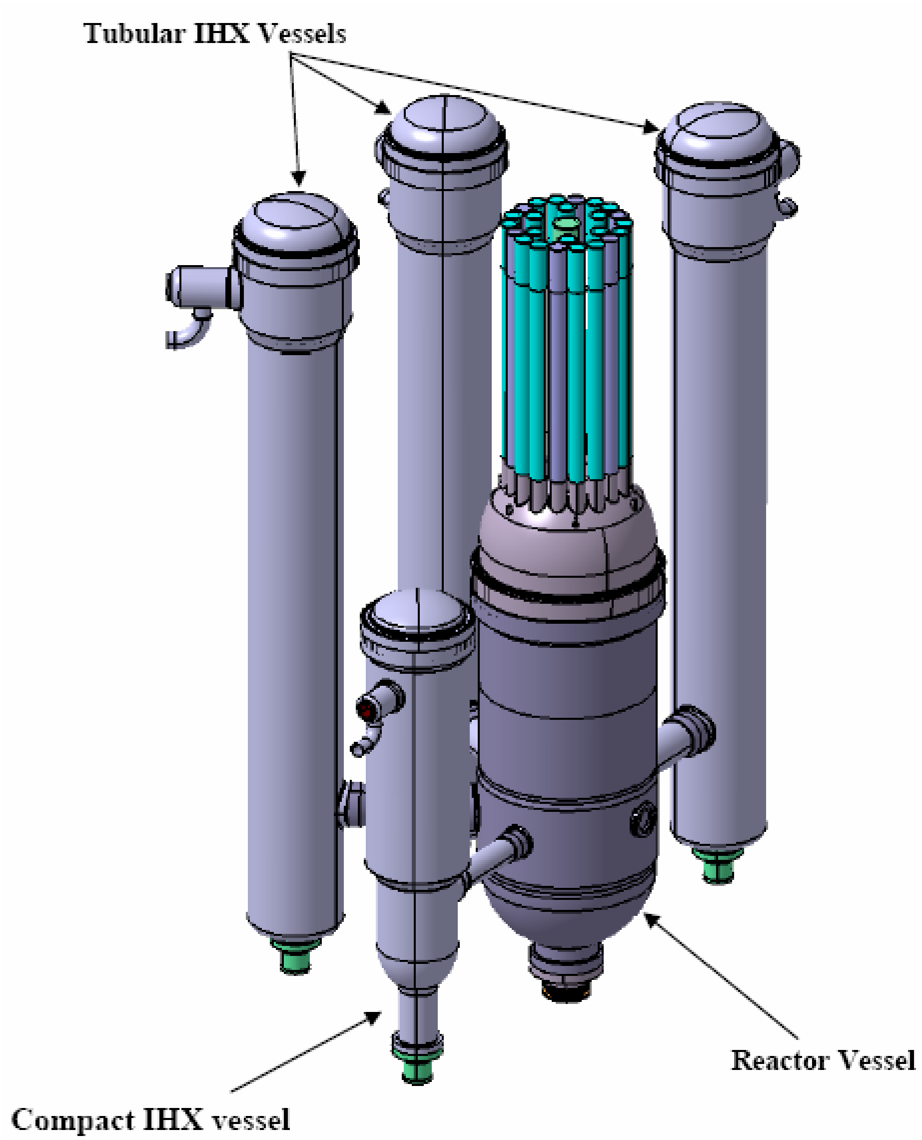
The bottom head provides a single large opening for the shutdown cooling system blower and heat exchanger components.

The lower portion of the cylindrical vessel includes a local reinforcement because of the presence of the cross vessel nozzles and one lug welded at the level of the cross vessel axis which is used, together with two cross vessels, to support the reactor vessel.

Due to transportation limitations to INL site, the size of the Reactor Vessel will likely require that the vessel be assembled on the construction site. The current concept is that the vessel will be delivered on site in 4 packages + 1 for the cover head. Three circular welds will be required for final assembly of the Reactor Vessel on site (see Figure 6-10). The site welding could be performed in a dedicated on site workshop including the corresponding heat treatment, the final machining, the non destructive examination, the hydrotest and the cleaning facilities.

The total weight of the Reactor Vessel is 825 T (including 225 T for the cover head).

Figure 6-8: Vessels Arrangement



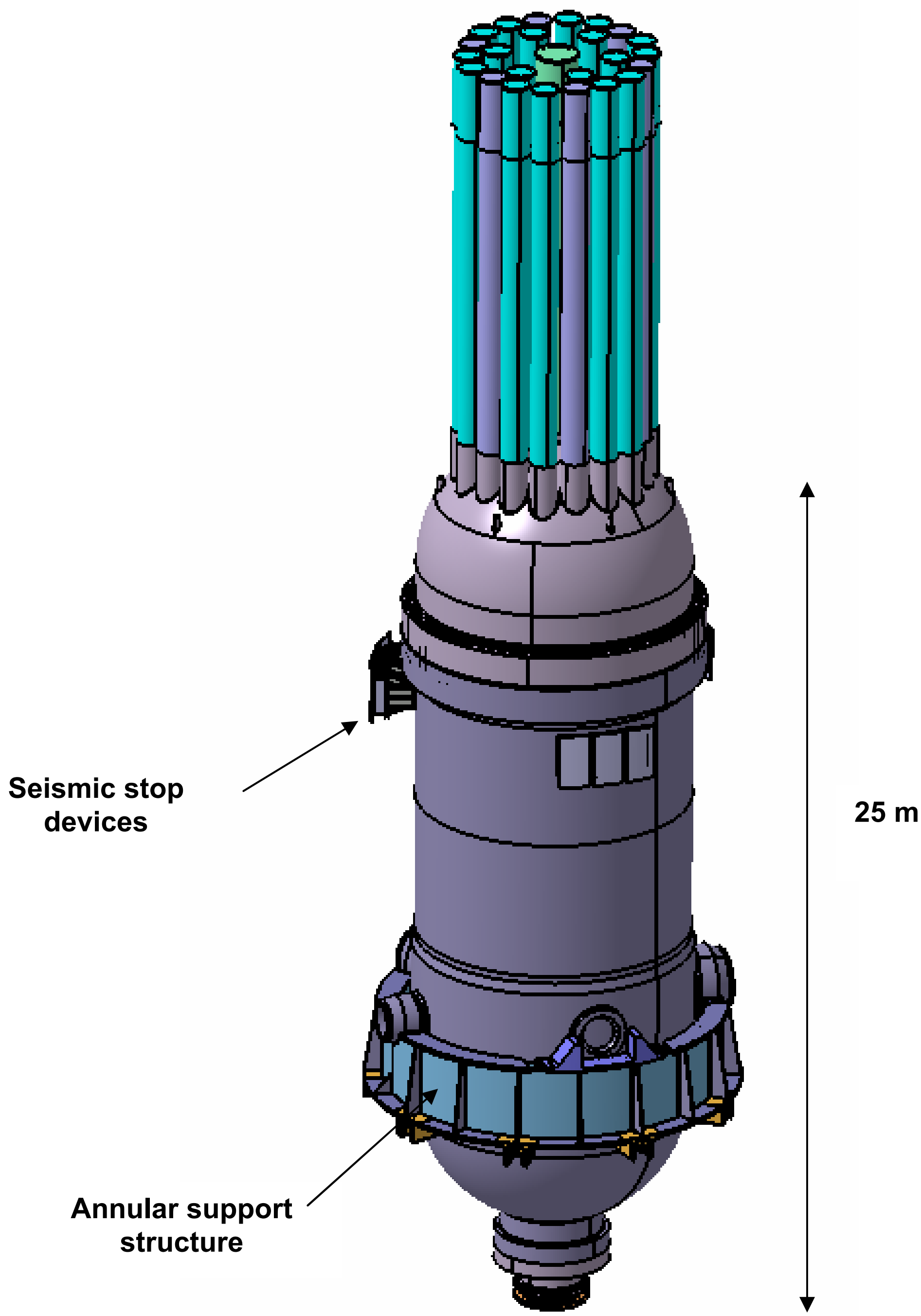


Figure 6-9: Reactor Vessel and Support System)

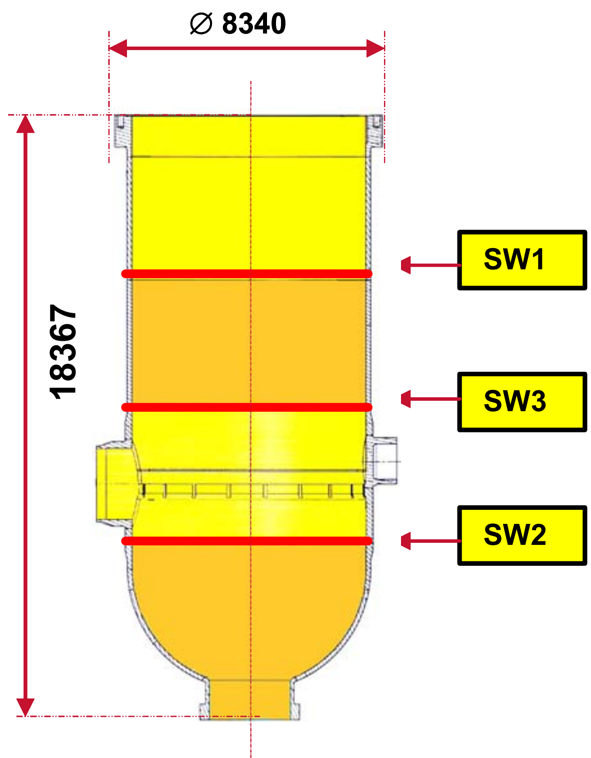


Figure 6-10: Reactor Vessel on Site Weld Locations

6.2.1.2 Nominal Operating Conditions

During normal operation, the Reactor Vessel is protected from the return Helium at 500°C by stagnant areas at the bottom, periphery and top of the vessel (see Section 6.1.2.2.3). A thermal hydraulic calculation of the Reactor Vessel has been performed under NGNP operating conditions. The calculation showed a peak vessel temperature of 425°C and an average through wall temperature of 363°C. These temperatures indicate that the requirement to operate the Reactor Vessel in the negligible creep regime should be satisfied with the present design and with mod 9Cr1 Mo as reference material (see Section 6.2.4).

The allowable stress for mod 9 Cr 1Mo vessel materials at 400ºC is 178.3 MPa, which gives a margin greater than 30% in the core beltline region during normal operating conditions.

6.2.1.3 Accident Conditions

A Depressurized Conduction Cooldown calculation has been carried out with a set of parameters conservative for the reactor vessel:

- Power level increased to 588 MW
- Residual power increased by 10%
- Inlet and outlet temperatures increased to 535°C and 918°C
- Graphite reflectors with properties of non-irradiated material

- Outer surface of the Reactor Vessel with an emissivity of 0.7.

Figure 6-11 indicates that the maximum temperature reached during DCC is 520°C which is fully acceptable for the vessel material.

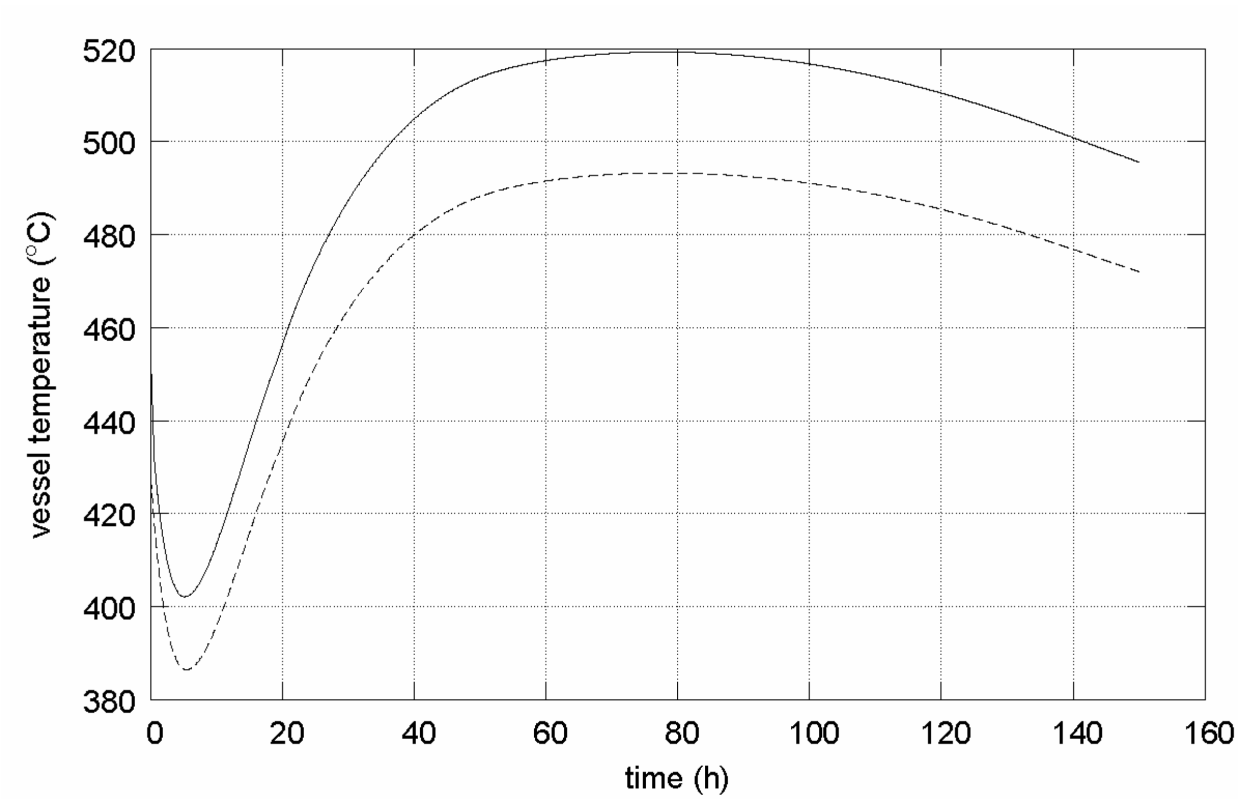


Figure 6-11: Evolution of Reactor Vessel Temperatures During DCC (Conservative Case) – Solid Line for Maximum Temperature and Dashed Line for Average Through-Wall

6.2.2 Cross-Vessels

The cross-vessels connect the lower portion of the Reactor Vessel to the lower portion of the intermediate heat exchanger vessels. The cross-vessels include a concentric duct (primary hot gas duct) that separates the hot (core exit) and the cold (core inlet) gas flow streams. The hot gas duct (see Section 6.3.4) is insulated to reduce regenerative heat losses to the outer flow stream (core inlet cold gas).

The cross vessels are spread around the Reactor Vessel with an angle of 60°. Cross vessels and IHX vessels are clustered on one side of the Reactor Vessel to minimize the footprint impact.

The welding of the cross-vessels to the Reactor Vessel and IHX vessels will be performed in the reactor cavity.

6.2.3 IHX Vessels

The sizes of the tubular and compact IHX vessels differ essentially by their height (about 32 m for the tubular vessel and about 21 m for the compact IHX vessel).

The outer diameter in the flange region is about 5 meters for both designs and, in contrast to the Reactor Vessel, it should be possible to fabricate these vessels in the workshop and transport them in one piece at INL site.

The IHX vessels should be thermally insulated in order to limit the heat losses and therefore increase the plant efficiency. As a result, the temperature should be very close to 500°C (except if active cooling were used or if thermal insulation was implemented inside instead of outside).

From a thermomechanical standpoint, the transition between “cold” and “hot” parts of the vessel system should be outside discontinuities (in particular outside the nozzles regions). The proposal is to use the cross vessel for this temperature transition, which means that the cross vessel would be partially insulated

6.2.4 Selection of Vessel Material

The reference material for the vessel system is modified 9-Chrome-1-Molybdenum (mod 9Cr-1Mo), also known as grade 91. This material was selected following a review of various candidates. The following sections describe the rationale for selecting mod 9Cr1Mo as the reference material for the Reactor Pressure Vessel (RPV) and the present status of its development.

6.2.4.1 Material Selection

The choice of the RPV material is to be made on the basis of the following considerations:

- Industrial experience,
- Research &Development information available,
- Operating conditions.

Three families of candidates can be envisaged for RPV applications:

- The first one is the grade used in PWRs, for which there is the greatest amount of industrial experience. This material is known in France as 16MND5 and is covered by the RCC-M code. It is also covered by American Codes and Standards and corresponds to ASME/ASTM A508 grade 3 class 1 for forgings and A533 grade B class 1 for plates.
- The second family covers 2.25Cr1Mo steel grades. A large number of grades can be considered with differences in the required tensile strength at room temperature. The different levels of yield and ultimate strength are obtained by different heat treatments and particularly by different tempering temperatures:
  - The ASME/ASTM 2.25Cr1Mo grade for use at elevated temperature in accordance with ASME Section III (nuclear application) is a fully annealed grade known as grade 22 class 1 (ASME SA 336 for forgings and SA 387 for plates). The level of its tensile properties is far below that of other material candidates.
  - Other grades are also approved in the ASME Code. ASME SA 541 grade 22 class 3 (forging) has tensile properties similar to those of the PWR steel but is not permitted in ASME Section III. ASME SA 336 grade 22 class 3 is permitted in Section III up to 371°C but its tensile properties are intermediate between those of ASME SA 336 grade 22 class 1 and SA 541 grade 22 class 3.
  - A particular grade of 2.25Cr1Mo had been developed in France as an alternative to 16MND5 in French PWRs. A strength similar to that of 16MND5 was obtained by steel making practice and heat treatment, but the final decision was to keep 16MND5 as the PWR material.
  - The RCC-MR code refers to other grades of 2.25Cr1Mo (10CD9 10 and 7CD9 10) which correspond to normalised and tempered or quenched and tempered grades. These grades are more resistant

compared to the annealed one, at least in tensile and short term creep conditions but they have not been characterized in heavy sections.

- The last family concerns high chromium alloyed steels of the type 9Cr1MoVNb or modified 9Cr1Mo (mod 9Cr1Mo). This material is known as ASME/ASTM grade 91 (ASME SA 336 for forgings and SA 387 for plates) and is also covered in the RCC-MR code.

Figure 6-12 provides a comparison of allowable stresses for the different material candidates. Notice that PWR steel and mod 9Cr1Mo have similar allowable stress around 370°C. 2.25Cr1Mo grades with high allowables show a significant drop in properties beyond 430°C. Other grades would require a significant increase of thickness compared to other candidates to compensate for the reduced tensile properties.

Table 6-4 describes the pros and cons of the different candidates. Mod 9Cr1Mo was selected taking into account the following points:

- Keep the RPV material in the negligible creep regime in order to avoid considering long term creep during design and avoid the implementation of creep damage monitoring in the surveillance program,
- Limit the selection to materials already considered by American Codes and Standards and permitted in Section III of the ASME Code.

It is believed that the development still required to qualify mod 9Cr1Mo is feasible in a time frame compatible with the NGNP schedule. Section 6.2.4.2 gives the status of development for mod 9Cr1Mo.

2.25Cr1Mo grades were not selected because the penalty on the vessel thickness is too large for the grades already permitted in Section III or the duration required to qualify the most resistant grades (not presently permitted in Section III or for temperatures higher than 370°C) is expected to be of the same order as or even larger than that required to qualify mod 9Cr1Mo.

The size of the RPV will require a final assembly of the vessel at INL site, but this is independent of the material selected.

A 100% forging design does not seem to be compatible with operation by 2018, and it is proposed to base the design on plates with a limited number of forgings (to be used for the flanges and nozzles). Japan Steel Works (JSW) is the only supplier for the large forgings that will be required. To ensure timely delivery of needed RPV components, the NGNP project must be positioned to move quickly on initiating work with JSW.

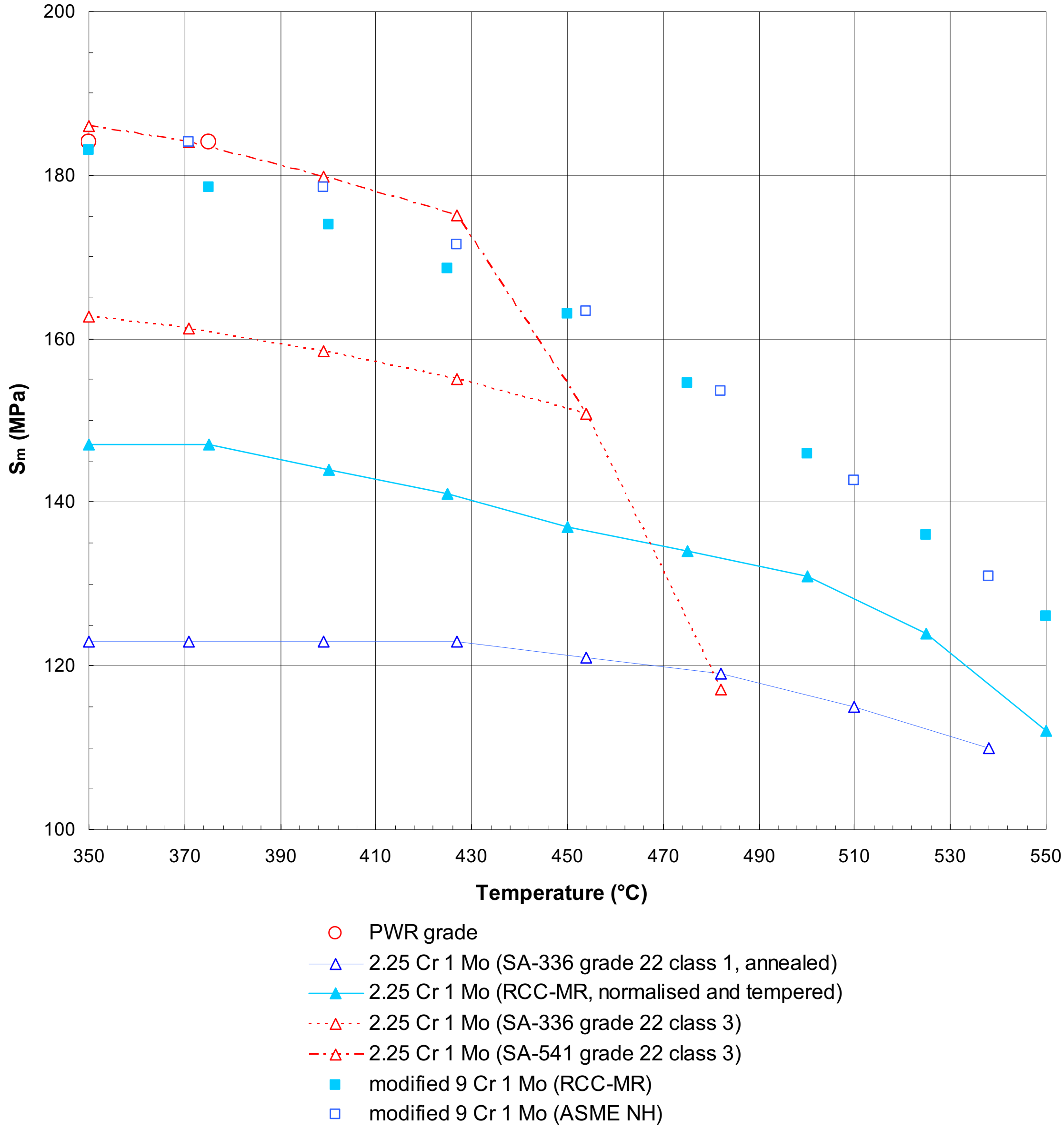


Figure 6-12: Comparison of Allowable Stress for Candidate RPV Materials

Table 6-4: Pros and Cons of RPV Candidates

Material	Pros	Cons
PWR grade	Material for which feedback from experience in nuclear industry is largest Material data file is already available Technological developments are limited R&D work needed is limited	Resistance to creep is limited Resistance to corrosion in He environment is likely to be lower compared to Cr-Mo grades Presently limited to maximum temperature of 1000°F (538°C) according to ASME
2.25Cr1Mo grades	Resistance to creep is better compared to PWR grade	Feedback from experience limited Tensile properties are low compared to those of PWR grade (at least for the grades permitted in Section III) which would necessitate larger thickness compared to other candidates (for the same loading conditions) Material data file to be written Technological developments needed (including definition of welding parameters and products)
Mod 9Cr1Mo	Resistance to creep is better compared to other candidates Tensile strength is similar to that of PWR grade	Feedback from experience mainly for non-nuclear industry with lower thickness products More difficult to procure large size ingots compared to other candidates Material data file to be written Technological developments needed (including definition of welding parameters and products) Post weld heat treatment temperatures higher than for PWR grade

6.2.4.2 Mod 9Cr1Mo Development Status

Mod 9Cr1Mo has been extensively studied in the past in the context of the Fast Reactor programs and data bases developed at that time can be used as a basis for defining R&D tasks which remain to be performed. Mod 9Cr1Mo is also commonly used in the non-nuclear industry. Two new issues have to be addressed in the context of HTRs:

- Heavy section products (including their welding),
- 60-year design life.

The following sections give a summary of actions carried out by AREVA NP to improve the knowledge of this material.

6.2.4.2.1 Mechanical Properties of Heavy Section Products

Significant efforts have been made to perform characterizations on representative heavy section products. Metallographic evaluations performed so far indicate a good homogeneity throughout the thickness. R&D actions presently underway are based on two products recently purchased:

- a forged plate, 200 mm thick, supplied by Japan Steel Work
- a rolled plate, 140 mm thick, supplied by Industeel.

Tensile tests performed on the 200 mm forged plate in the temperature range 20°C-600°C indicated that yield strengths are higher than the ASME minimum values. Concerning ultimate tensile strength, the data obtained from the 200 mm forged plate are slightly lower than ASME values but further evaluation should be performed to clarify if the difference should be attributed to a product effect or to the definition of ultimate tensile strength (as a reminder, ASME design values should not be considered as true minima).

It is also to be mentioned that actions are underway in the context of the ASME/DOE Gen IV material project (actions led by the University of Dayton Research Institute). Activities concern the update of stress allowables for mod 9 Cr1Mo, covering the effect of product form and extension of stress allowables to a 60-year design life.

6.2.4.2.2 Impact and Toughness Properties

Impact tests have been performed at - 20°C and 0°C on products purchased in Europe. Impact tests were also performed at other temperatures in order to determine the Charpy V transition curves. For the Charpy V at -20°C, the target in Europe was 40 J minimum for the average value of the three specimens and 28 J minimum for individual test results. These values were met for rolled and forged plates with thicknesses from 20 to 200 mm.

6.2.4.2.3 Creep

Creep test programs underway are mainly dedicated to defining the negligible creep domain. They are aimed at improving the knowledge of creep properties at moderate temperatures (< 500°C), including the effect of the post weld heat treatment. Negligible creep is also a topic which has been studied in the context of the ASME/DOE Gen IV material project (AREVA NP as the lead contractor). This work also covers creep-fatigue of mod 9Cr1Mo.

6.2.4.2.4 Effect of Aging

From available data on modified 9Cr1Mo, it can be expected that there should not be any significant aging effect below 480°C. Nevertheless thermal treatments with increasing duration up to more than 25,000 hrs at 450°C, 475°C and 500°C have been started to confirm this conclusion. Base material, heat affected zone, and weld metal samples are included in the test program. The present status after 10,000 hrs at 500°C indicates no shift in the ductile brittle transition temperature (DBTT).

6.2.4.2.5 Effect of Irradiation

Irradiations have been carried at Joint Research Center in Petten on both base metal and weld metal (150 mm thick welded joint). No significant shifts in mechanical properties and ductile-brittle transition temperatures have been observed for the expected end-of-life fluence of the reactor pressure vessel.

6.2.4.2.6 Corrosion in Helium Environment

For temperatures below 450°C, expected carburization in impure helium environment will be a very slow process affecting only the surface layers of the vessel wall. For temperatures expected during off-normal situations (about 550°C), no problems are expected due to their short durations. A test program is however necessary to confirm the performance of mod 9Cr1Mo in representative HTR conditions.

6.2.4.2.7 Weldability

A significant R&D program has been launched to demonstrate the weldability of heavy section products. Initial tests carried out with GTAW (Gas Tungsten Arc Welding) process had shown hot cracking. It was shown later on that proper selection of filler material could eliminate this problem. The welding program covered the main welding processes likely to be used, namely SAW (Submerged Arc Welding), GTAW and SMAW (Shielded Metal Arc Welding) processes. The available results are very encouraging, showing acceptable mechanical properties and no cracks. Optimization is still necessary to achieve the required impact test values for the range of post weld heat treatment temperatures investigated, in particular for SAW and SMAW processes. Further activities are envisioned on GMAW (Gas Metal Arc Welding or MIG) process, which uses a filler material similar to that used for GTAW and is suitable for automatic on-site welding in horizontal position, with a larger deposit rate compared to GTAW.

6.2.4.2.8 Emissivity

Understanding of radiative heat transfer is of prime importance in the evaluation of the temperatures of the fuel, reactor vessel, and metallic internals, particularly during conduction cooldown situations. Measurements have been carried out to define emissivity values, not only for the Reactor Pressure Vessel material but also for the metallic and graphite internals. Tests have been carried out for the range of temperatures covering normal to off-normal situations and taking into account specimens with surface conditions representative of the RPV at the beginning and end of life.

6.2.4.2.9 Codes and Standards

Mod 9Cr1Mo is presently covered by ASME Section III Subsection NB for temperatures below 700°F (371°C). Rules have been introduced in Subsection NH (2004 edition) to include mod 9Cr1Mo for higher temperatures. Rules are presently limited to plates and small size forgings, and revision is necessary to extend the rules to heavy section plates and forgings and extend the stress allowables to cover a 60-year design life. Other necessary Code improvements concern the definition of negligible creep conditions and the improvement of creep-fatigue design rules.

6.2.5 Vessel Support Arrangement

The support concept of the primary circuit vessels is based on AREVA experience feedback on PWRs and consists of:

- An annular support structure for the reactor vessel, anchored in the concrete in the prolongation of the cross vessels axis.
- Seismic stop devices at the upper part of the reactor vessel.
- An annular support structure for the IHX vessels ensuring both, vertical supporting and free horizontal thermal expansion of the component in the cross vessels axis direction.
- An annular structure at the upper part of the IHX vessel ensuring both, the horizontal free thermal expansion of the component in the cross duct axis direction and the horizontal locking in case of seism (stop device dashpot type + lateral stops).

Figure 6-9 shows the detail of the Reactor Vessel support system.

6.3 Primary Heat Transfer System

The main function of the Primary Heat Transfer System (PHTS) is to transfer the heat from the reactor core to the secondary circuit. Mixture Helium/Nitrogen is proposed for the secondary gas as the reference gas for electricity production. Pure helium is the option proposed for the heat transport loop to H2 plant.

This system is designed with the following components:

- The Main Primary Gas Circulators (MPGC) which blow the primary helium. Four circulators are proposed for the NGNP design.
- The circulator shutoff valve.
- The Intermediate Heat EXchanger (IHX). The tubular IHX concept is the reference for the IHXs to Power Conversion System. Compact IHX technology is proposed for the IHX to H2 plant.
- The primary and secondary hot gas ducts.
- The isolation valves.

All these components (except the isolation valves) are fully immersed in the primary helium inside the pressure vessels (cross and IHX vessels).

6.3.1 Circulators

For the NGNP, two types of circulators are envisaged:

- Circulators for the heat transport to electric plant,
- Circulator for the heat transport to H2 plant.

It is proposed for the NGNP to use 3 circulators for the heat transport to electric plant (one for each tubular IHX vessel). Those circulators have a power level around 5 MW and remain in the gas circulators design feasibility. The use of two circulators instead of three should remain acceptable and would need no significant R&D or qualification needs. On the opposite, using one single gas circulator is considered as feasible but is beyond present state of the art and would require significant development.

The gas circulator for the 60 MWth process heat loop is a smaller gas circulator (about 1.5 MW) which does not present any feasibility issue.

The following description is applicable to both types of circulators:

- The gas circulators are encapsulated centrifugal, vertically mounted, and with the impeller mounted directly on the motor shaft. The circulator may hang from the bottom (reference option) or may rest on top of the IHX vessel without significant consequences on feasibility. Impellers are high efficiency backward curved plate bladed and are matched aerodynamically to a radial diffuser, radial to axial bend and a short annular diffuser.
- The gas circulator flowrate is controlled by speed control through an electrical motor/inverter.
- The motor is vertically mounted within the pressure vessel. Therefore it is a submerged design and is separated from the IHX vessel by a thermal barrier plate devoted to minimize the ingress of heat from the primary side into the motor compartment. There is a small clearance into this barrier at the impeller level to balance the pressure between main primary vessel and motor compartment.

- The motor compartment is cooled by a water/helium heat exchanger, a fan within the compartment ensures fluid circulation for an efficient heat transfer. Also helium injection into the cavity allows maintaining a slightly higher pressure into the compartment and can be used for cavity cooling at motor standby if required. It could also be considered as a redundant cooling system in case of water/helium heat exchange cooling circuit failure.
- The gas circulator is supported by radial and axial electro-magnetic bearings (EMB). Mechanical catcher bearings are set to support the machine in case of EMBs failures.

There is no material issue concerning the gas circulators. High stresses expected at the impeller level should be solved by the selection of Alloy 718 as reference material. This material is identified as a suitable material due to its high strength and performance at elevated temperatures.

6.3.2 Circulator Shutoff Valve

Each primary gas circulator is equipped with a self-actuating valve set at impeller inlet. This valve is requested to prevent possible reverse flow (potentially hot) from the reactor core onto cold primary structures in case of conduction cooldown situations and also to avoid the core by-pass in shutdown conditions when the SCS is in operation. This valve closes in the event of a malfunction or failure of the indirect cycle system or during a maintenance operation.

6.3.3 Heat Exchangers

The following sections describe the selected technologies for the NGNP and provide the rationale for this selection.

6.3.3.1 Types and Sizes

Table 6-5 lists the pros and cons of IHX concepts. Taking into account the feedback from experience on tubular IHX and the challenge associated to designing a compact IHX by 2018, it is recommended to base the design of the IHX to PCS on tubular technology. Operation at 900°C for 20-year design life seems feasible with such a technology. The past experience indicates however that this temperature would necessitate a lower pressure difference at the hot end (requiring a pressure on the secondary side larger than that on the primary side). It could be also envisioned with such a design to operate at higher temperature (e.g., 950°C) for a short duration.

It is recommended for the NGNP to use three 193 MWth tubular IHXs. The power level is slightly higher than what is considered as state of the art today (about 150 MWth) but this is considered as achievable. Such an increase of power will be obtained by a combination of increase of the number and length of the exchange tubes.

The design of a compact IHX for PCS application is considered as too challenging for a startup by 2018. It is however recommended to use such a concept for the 60 MWth IHX to H2 plant in order to use the NGNP as a test bed for new technologies. This should be all the more feasible that conditions are more favorable compared to those for IHX to PCS:

- He on the heat transport loop side
- Pressure transients less severe.

It is, however, likely that the design life of the compact IHX will be significantly reduced compared to that of the tubular IHX (expected design life of the compact IHX is 5 years). As technology evolves, this IHX could be replaced in the future by IHX with more advanced materials (for instance ceramics).

Table 6-5: Pros and Cons of IHX Concepts

	Tubular	Compact	Comments
Feedback from experience	++	0	Compact IHX design not demonstrated
Robustness	++	-	1) Tubular IHX considered as robust which may have consequences on other systems (e.g., non-safety isolation valves may be envisioned) 2) Compact IHX design may be challenging for pressure transients (loss of pressure has to be considered on either side)
Thermomechanical design	+	- -	Tubular design is more flexible and thermal gradient is over a longer distance (tube length around 30 meters)
Welding	+	- -	Most compact IHX designs rely on bonding techniques that remain to be developed
Corrosion / Nitriding	-	- -	Less an issue for tubular IHX as tube thickness is larger than the plate thickness used in most compact IHX
ISI&R	++	- -	ISI&R is possible for tubular concept (including tube plugging if need be) whereas ISI&R of compact IHX modules is impossible (even the identification of the leaking IHX module may be challenging)
Control of radionuclide	+	-	
Acceptability by the Regulator	+	-	May be challenging for the compact IHX due to the difficulty to inspect it.
Compactness	- -	++	3 or 4 IHX vessels required for the tubular concept whereas one single vessel required for the compact IHX (even though the latter is larger than the tubular IHX vessels)
Economics	- -	++	Clear advantage at lower temperatures. For temperatures which require a life reduction of the compact IHX design life, a detailed cost evaluation is needed
Risks	0	--	Compact IHX design not demonstrated

6.3.3.1.1 IHX for Heat Transport To Electric Plant

The proposed tubular IHX is based on the helical coil concept, with tube diameter of 21 mm (2.2 mm wall thickness).

The proposed concept represents a first-of-a kind design, because of its size and of the stringent requirements on thermal effectiveness, leak-tightness, thermo-mechanical resistance, materials, reliability, and cost.

The main components of the tubular IHX are the following (see Figure 6-13):

- Tube plate
- Central Tube
- Hot Header
- Helix Tube Bundle

The bottom of the tube plate is protected by a special insulation system to minimize the stresses inside the tube plate during the normal conditions.

A removable insulation system is arranged inside the Central Tube. This insulation system minimizes the stresses inside the Central Tube and minimizes the thermal losses of the secondary coolant flowing from the Hot Header at the lower region of the IHX to the top of the IHX.

The Helix Tube Bundle is separated from the cold gas back flow to the bottom of the IHX by an insulated outer shroud. This shroud is aimed at minimizing the heat loss from the hot flow areas inside the bundle to the cold gas area.

The Hot Header is insulated with a special fiber insulation system and covered by a liner. Together with the outer shroud of the tube bundle, these systems build a special formed flow channel to get a nearly constant flow velocity inside the tube bundle.

The proposed IHX bundle is composed of 31 rows with the smallest diameter of 1,248 mm and the biggest diameter of 3,253 mm. The tube bundle is designed with a helix-angle of 25.38° to minimize forces resulting from thermal expansion. Exchanger tubes are supported by tube supports which are designed to allow free radial and axial motion of the tubes.

The Hot Header is made out of a forging. The connection studs from the Hot Header to each tube are machined. With this design each tube welding is a normal circumferential welding and therefore easy to control with Ultrasonic Testing.

With regards to its mechanical resistance, the IHX has to take no part in the primary boundary function. This function is ensured by the pressure vessel where the IHX is inserted and the IHX has only to withstand the pressure difference between the primary loop pressure and the secondary loop pressure. The IHX will have however to withstand the full primary pressure (1 bar in the secondary loop) for a specified time duration.

A calculation has been carried out to evaluate the parameters of such an IHX with NGNP conditions and these parameters are summarized in Table 6-6. The number and length of tubes has been adjusted so as to give a pressure drop on the secondary side lower than 3 bars, to avoid penalizing the efficiency of the PCS system.

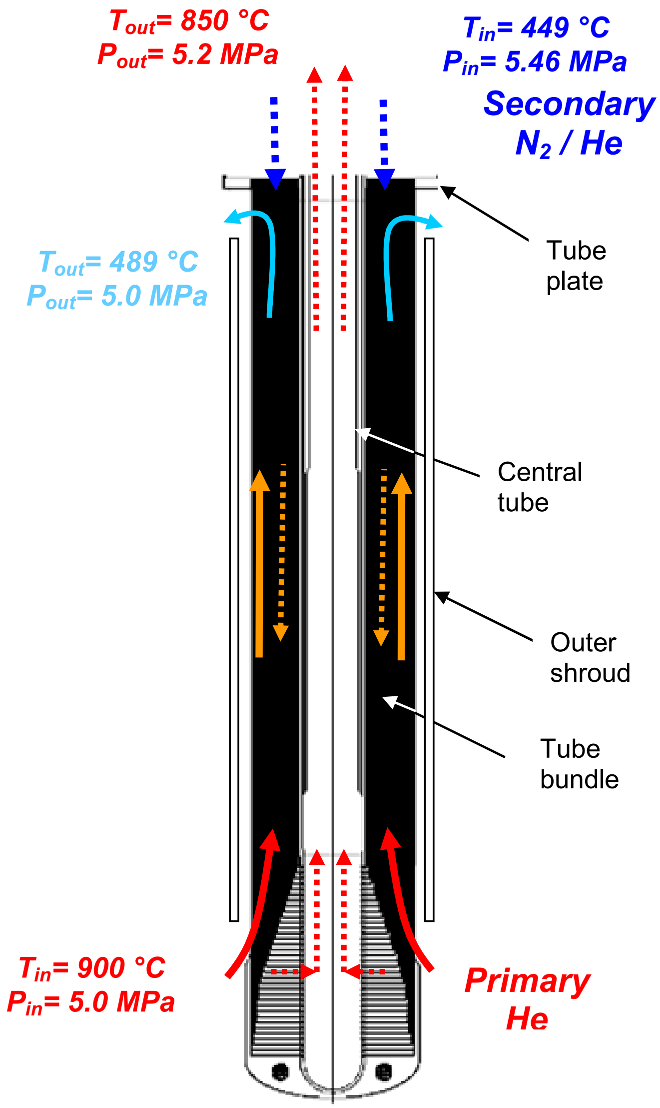


Figure 6-13: Tubular IHX Concept

Table 6-6: Tubular IHX Parameters

		NGNP Tubular IHX
Desired Thermal Power	MW	193
Primary fluid		He
Secondary fluid		$He\text{-}N_2$
Primary side (shell side)		
Inlet Temperature	°C	900
Outlet Temperature	°C	489.5
Inlet pressure	MPa	5
Pressure loss	bar	0.1133
Mass flow	kg/s	90.5
Secondary side (tube side)		
Inlet Temperature	°C	449.1
Outlet Temperature	°C	850
Inlet pressure	MPa	5.46
Pressure loss (only for the bundle area)	bar	2.6
Mass flow	kg/s	245
Geometry		
Outer Tube diameter	mm	21
Inner Tube diameter	mm	16.6
Tube wall thickness	mm	2.2
Conductivity of tube	W/(m*K)	23
Number of tubes		3021
Helix angle	°	25.38
Diameter of first coil	m	1.248
Helix inner diameter (inner wall)	m	1.181
Helix outer diameter (outer wall)	m	3.39
Length of tubes (developed)	m	28.6
Tube bundle height	m	12.25
Heat exchange area	m^2	5697

6.3.3.1.2 IHX for Heat Transport to H2 Plant

Compact IHX technology is proposed for the IHX for heat transport to H2 plant.

Different types of compact IHX could be envisioned for such an application:

- Plate Machined Heat Exchanger
- Plate Fin Heat Exchanger

- Plate Stamped Heat Exchanger

The different concepts will have to be evaluated in the Conceptual Design phase. The Plate Stamped concept could be considered as the best candidate as less demanding in terms of bonding technology.

The parameters of the compact IHX are given in Table 6-7. An approach temperature of 25ºC is selected to provide the higher temperature on the H2 plant side. Taking into account this approach temperature, it is evaluated that the volume necessary for compact IHX modules should be about 4 to 5 m3 (module volume excluding pipes or headers). Therefore, 6 compact IHX modules would be required for such an application.

It is proposed to base the design on the "hot centre" concept which has been developed by AREVA for commercial application. Figure 6-14 shows how this concept could be implemented for a 6-module IHX.

Table 6-7: Compact IHX parameters

		NGNP Compact IHX
Desired Thermal Power	MW	60
Primary fluid		He
Secondary fluid		He
Primary side (shell side)		
Inlet Temperature	°C	900
Outlet Temperature	°C	500
Inlet pressure	MPa	5
Mass flow	kg/s	29
Secondary side (tube side)		
Inlet Temperature	°C	475
Outlet Temperature	°C	875
Inlet pressure	MPa	5.2
Mass flow	kg/s	29

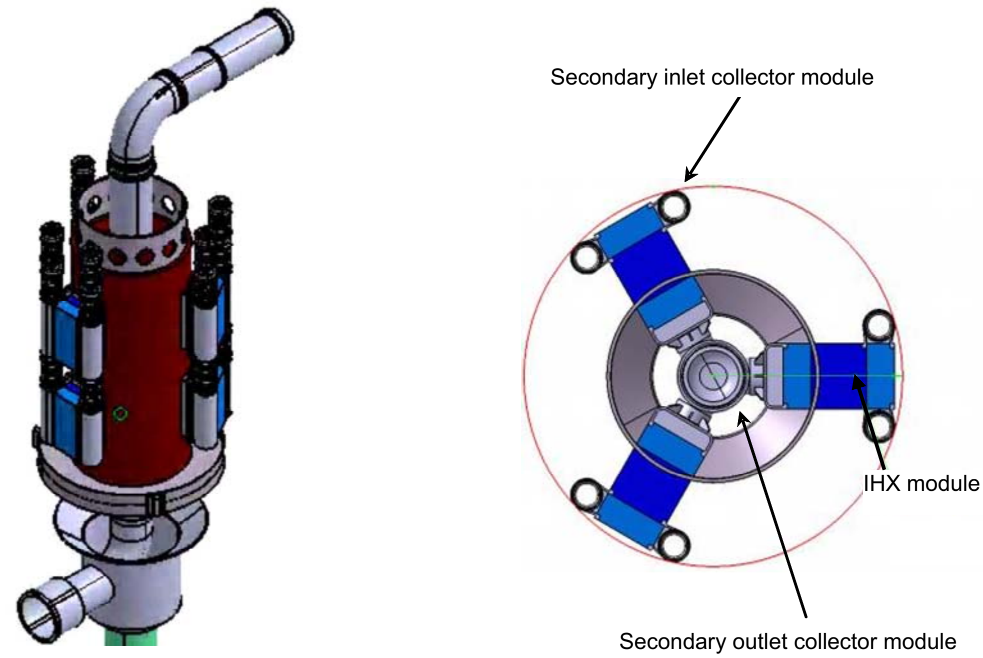


Figure 6-14: Compact IHX Concept

6.3.3.2 Working Fluids

Helium is the environment on the primary side and in the heat transport loop to H2 plant. Helium alone is an inert atmosphere but He in HTR environment will contain impurities coming from various sources.

Corrosion in helium environment is caused by the reaction of the metal with helium impurities (H_2O, CO, CH_4, H_2, and CO_2). Key parameters that influence reaction rates are the temperature, the impurity concentration partial pressures and the alloy composition (chromium activity).

The main corrosion phenomena observed and their consequences are:

- Oxidation: loss of thickness (superficial) or ductility loss and cracks formation (internal)
- Carburization: loss of protection against oxidation (external layers) or ductility loss and crack formation (internal)
- Decarburization: strength loss.

All these interactions between helium impurities and alloys can be described by the stability diagrams which link the material properties and the environmental conditions, considering that for Ni-base alloys with high Cr and low Al, Cr is the main element governing the corrosion phenomena (Figure 6-15). This diagram is divided into several regions:

- Region I : the chromium is theoretically stable,
- Region II : the alloy is decarburized through the oxide layer, this area is considered as the most deleterious for the mechanical behavior,
- Region IIIa : the carburization through the oxide layer is low if this layer is protective,
- Region IIIb : the carbon activity is lower than Region IIIa and the area is considered as safer

- Region IV : the oxide film may be found under the carbide surface layer, the oxide is not very protective and internal carburization will occur,
- Region V : carburization occurs without formation of an oxide layer.

The impurity level in He environment has to be controlled to ensure that components operate in Region IIIb (benign domain).

The environment on the PCS side is a mixture of 80% N2 and 20 % He. This mixture has been selected to enable the use of air-breathing turbine. Nitriding may be an issue on the secondary side.

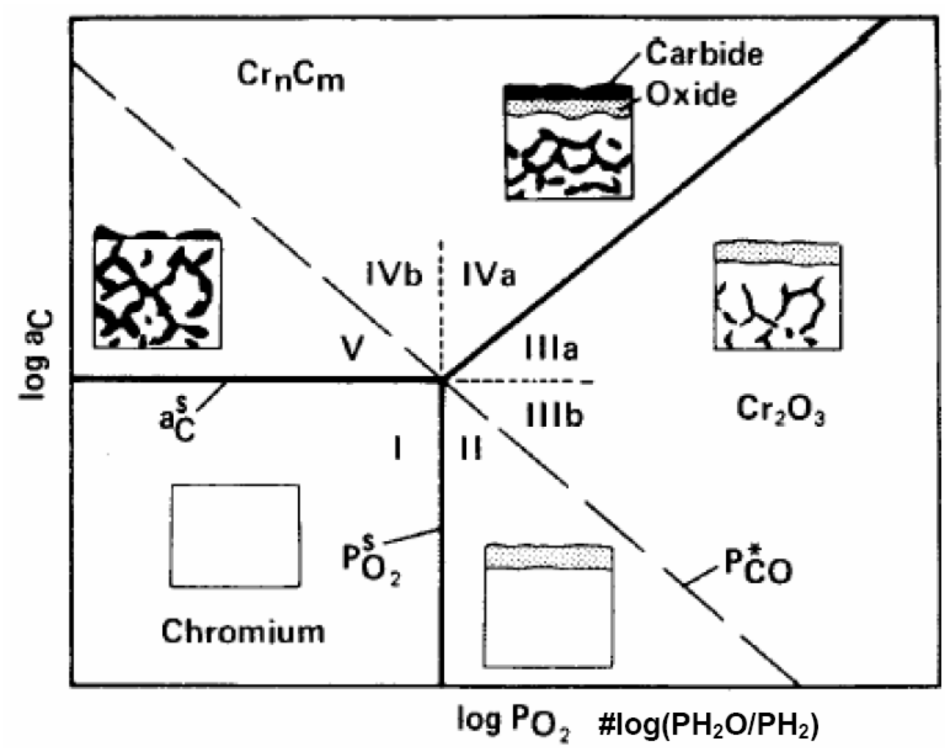


Figure 6-15: Chromium Stability Diagram

6.3.3.3 Materials

Interaction between the material and the environment is a serious issue at high temperature (850ºC and above) for the Ni base superalloys envisioned for IHX applications.

As discussed in Section 6.3.3.2, a control of impurities is required to limit the effects on the materials. A "benign" He environment will have however the following consequences on the material:

- Creation of an oxide layer
- Internal oxidation
- Carbide depleted zone under the oxide layer

After 100,000 hr of operation and more, a significant fraction of the tube or (thin) plate thickness could be affected by those phenomena.

Nitriding on the secondary side is also an issue and it does not seem that a control of the He-N2 mixture chemical composition (for instance creating favorable conditions for the formation of an oxide layer) is sufficient to prevent nitriding.

It is therefore recommended investigating the possibility of adding a coating on both primary and secondary side (mainly on thin products). Aluminide coatings have been developed in the other sectors of industry and could be envisioned for such an application.

It is also to be mentioned that the behavior under HTR environment can be very dependent on the material selected. Haynes 230 seems to have a slightly better behavior than In 617 in He environment but does not seem to be the ideal candidate from this point of view. The advantage is due to reduced Al and Ti content as indicated in Table 6-8 On the other side, the stability of Haynes 230 under thermal aging does not seem to be as good as that of Inconel 617.

Table 6-8: Chemical Compositions of Haynes 230 and In 617

Alloy	Fe	Cr	Co	Mo	W	C	Si	Mn	P	S	Al	Ti	La	B
230	*- 3.0*	*20.0 24.0*	*- 5.0*	*1.0 3.0*	*13.0 15.0*	*0.05 0.15*	*0.25 0.75*	*0.3 1.0*	*- 0.030*	*- 0.015*	*0.20 0.50*		*0.005 0.050*	*- 0.015*
617	*- 3.0*	*20.0 24.0*	*10.0 15.0*	*8.0 10.0*		*0.05 0.15*	*- 1.0*	*- 1.0*		*- 0.015*	*0.8 1.5*	*- 0.6*		*- 0.006*

Other material candidates would be materials developed in Japan for similar applications: Hastelloy XR and Ni-Cr-W superalloys.

Hastelloy XR is the material used for the IHX of the HTTR. This material is not as creep resistant as Haynes 230 and In 617 but its behavior in He environment seems to be significantly better. Hastelloy XR is the result of several years of development starting from Hastelloy X. Hastelloy XR has been developed to improve the behavior under He environment. The modifications consisted in:

Restriction in Mn and Si content to form stable and adherent oxide layers

Reduction of Al and Ti content to suppress internal oxidation and intergranular attack.

Hastelloy XR improvements were made for He environment only and it is not expected that those improvements would have any effect in nitriding environment.

Other materials developed in Japan were Ni-Cr-W superalloys. These materials seem to be very promising in terms of creep behavior but seem to be challenging in terms of welding.

The preferred material is In 617 due to the availability of the database. This material can only be used if solutions can be found to efficiently protect it against corrosion in He and nitriding in He/N2 mixture. It is therefore recommended to study those solutions with high priority. If such solutions would appear not feasible by 2018, another material will have to be selected. Hastelloy XR seems appropriate with He environment but would not solve the problem on the secondary side. In this case, other environments would need to be investigated on the secondary side (e.g., mixture He+ Argon, He+ Neon, etc).

6.3.3.4 Design Code

There are currently no rules in the ASME Code to cover the design of IHX made of high temperature Ni base superalloys. A draft Code Case had been developed in the 90's to cover In 617 but this Code Case would need to be updated to incorporate the progresses made since then. Actions have to be initiated rapidly with ASME to ensure that validated design rules will be available in due time.

6.3.4 Hot Gas Duct

The reference design for the primary and secondary hot gas duct is based on the concept presented in Figure 6-16. The structure is composed of the following:

- Liner
- Depressurization gap for controlled discharge volume of gas volume from fibrous fill in case of rapid decompression
- Perforated tube (baffle ring)
- Wrapped fiber mat insulation
- Support tube serving as pressure boundary between the hot and cold channels.

This arrangement is interrupted approximately every 1200 mm by V-shaped spacers. The insulation is packed in the region of the V-shaped spacers.

This type of arrangement is foreseen from graphite core support structure to the IHXs and from the IHXs to the isolation valves. It consists of either straight parts or elbows. Differential thermal expansions between the hot gas ducts and the vessel system can be accommodated either by bellows or metallic seals.

No definitive decision has been taken in terms of metallic material selection. Material candidates are alloy 800H for the support tubes and either alloy 800H or Ni based superalloys for the other internal materials. It is evaluated at this stage that this design should be appropriate even at 900°C core outlet temperature. It will be however needed to check that hot streaks coming from the hot gas plenum will be acceptable for the liner material of the primary hot gas duct. If temperatures were assessed to be too challenging for the material, another design based on ceramic liner and spacers could be used.

The primary hot gas duct is located at the proximity of the core and Ni-base superalloys will have to be selected with a preference on low-activation materials, such as Haynes 230, to facilitate maintenance of such a component.

As far as the secondary hot gas duct is concerned, the temperature is 50°C lower than that of the primary hot gas duct. In addition, there is no hot streak issues and no risk of activation. Its design and qualification can be therefore considered as enveloped by that of the primary hot gas duct, except that environment is the mixture He/N2 instead of He and issues discussed in Section 6.3.3.4 will have to be addressed.

6.3.5 Other Ducts and Piping

Other ducts and piping concern essentially the internals of the compact IHX vessel. Those ducts will have to be thermally isolated with a technology similar to what is envisioned for the primary and secondary hot gas duct.

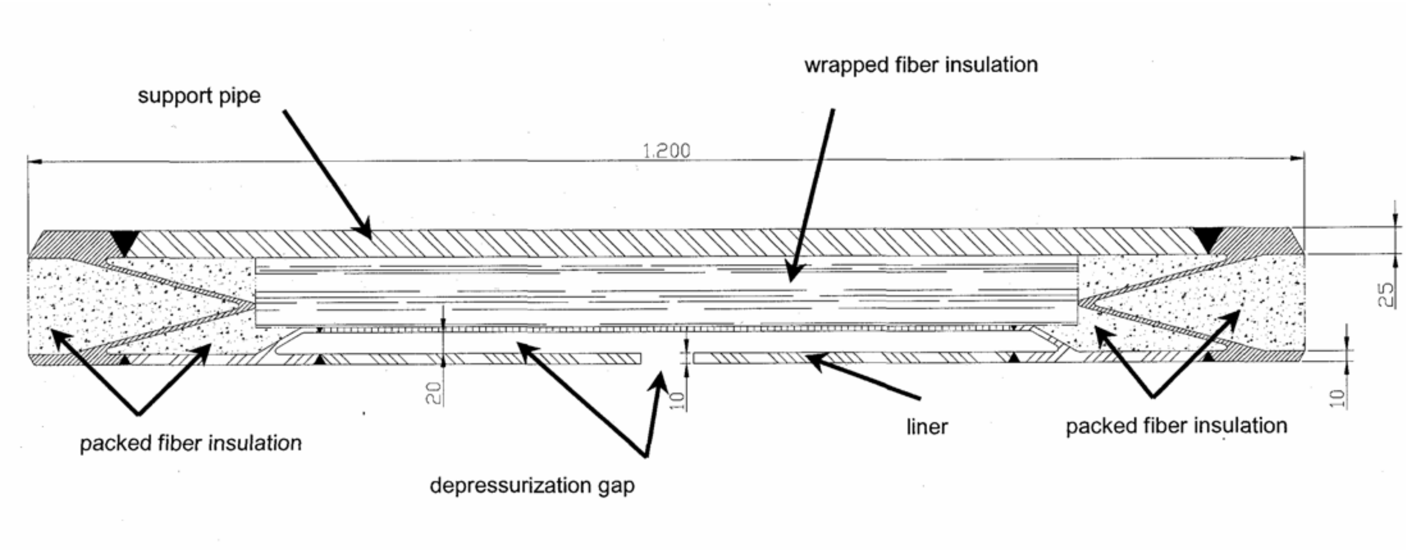


Figure 6-16: Hot Gas Duct Concept

6.3.6 Isolation valves

It is envisioned using isolation valves at the inlet and outlet of each IHX.

The hot gas isolation valve is of the double-acting axial type which is closed and held closed pneumatically by gas injected into the space between the two piston cones. This type of technology has been qualified up to 900 C in the context of the former German HTR development project and led to testing at KVK facilities.

The radial layout of the hot gas isolation valve is as follows:

- Liner
- Depressurization gaps for controlled discharge of gas volume from fibrous fill in case of rapid decompression
- Stuffed fibre insulation
- Valve seat on both sides for the seal cones
- Three radial struts for positioning the piston cones
- Two piston systems with special gas-bearings
- Special fiber brick insulation for the cones and the struts
- Special formed sealing elements with cooling system
- Pressure housing.

The valve can be closed at any time by the injection of pneumatic gas at a pressure of about 20 bars above the secondary pressure level, i.e., it is closed even against the full static pressure at nominal flow. With such a design, when the valve is closed, only pneumatic gas can escape across the valve seats at the specified leakage rate. This prevents the release of potentially contaminated gas in the event of leakage of an intermediate heat exchanger.

As far as metallic materials are concerned, material selection and issues are the same as those for the secondary hot gas duct.

Conventional valves can be used for the cold gas isolation valves. Contacts will need to be made with suppliers such as KSB in Germany to define the most appropriate products for nuclear application.

6.4 Reactor System Feasibility Issues

The concept studied in the context of the NGNP preconceptual design was specifically selected to maximize feasibility and to minimize development risk. The design based on indirect combine cycle relies on conventional gas turbine technology on the Power Conversion System side. This concentrates the needed development effort on the reactor system, in particular on the IHXs required to couple these two systems, and avoids dealing with complex interfaces between these systems.

Feasibility issues could be further reduced by adopting an alternative design for which heat would be supplied to the H2 plant at a lower temperature and the required temperatures would be obtained by electric heating. This is in particular the case for the steam cycle HTR concept which couples the reactor to a Rankine power conversion system through a steam generator. Such a concept has the fewest feasibility issues, because the corresponding operating conditions tend to be less demanding. More importantly, several such systems have been built and operated around the world, and the basic technology is well established.

The following sections describe the feasibility issues of the reactor system described in Sections 6.1 to 6.3, in accordance with AREVA scope of work which had to be focused on the combine cycle concept.

Fuel Feasibility

Requirements on fuel are based on ambitious goals with operation at high temperature, high burnup and very low defects. The present goals push the fuel performance beyond proven demonstrated capabilities. An extensive demonstration program will be necessary before the claimed goal performances are met.

Fuel feasibility issues are further discussed in Section 15.4

IHX Feasibility

The tubular IHX selected for the heat transfer to Power Conversion system is considered as feasible, even if the proposed parameters (in particular the power) goes beyond the present state of the art for this kind of components.

Based on past experience in Germany (full scale mock up tested in the KVK helium loop) and Japan (HTTR), a high temperature tubular IHX is deemed feasible at the following conditions:

- Helium/helium heat exchanger
- Effectiveness 90 %
- T = 850°C and with some limited periods in operation up to 950°C
- Limited pressure difference in operation < 3 bars
- Lifetime 20 to 30 years
- IHX module power around 150 MWth.

The proposed 193 MWth tubular IHXs will require an increase of the number and length of tubes which should be achievable through design improvements. The extension to 900°C design temperature should be obtained by a reduction of design life to 20 years. This design life will however require a perfect knowledge of interaction between the metallic material and the VHTR environment. As discussed in Section 6.3.3.3, corrosion and nitriding are a concern at such high temperatures and it is recommended investigating the possibility of protecting the hottest parts of the IHX with a coating. Further R&D will be also required to confirm the material behavior at such temperatures and provide necessary information in the context the material and component qualification program.

For the compact IHX proposed for the heat transport to H2 plant, significant R&D and design work is still required to obtain a design able to operate at 900°C (or above). Operating conditions are however less demanding (reduced pressure transients and He environment on the secondary side) and it is currently considered that such a concept can be implemented, subject to limiting the design life to 5 years. This life reduction is acceptable due to limited cost impact on the overall plant and due to the fact that the availability required on the H2 plant side should not be as large as that required on the Power Conversion side.

Vessel Material Feasibility

Mod 9Cr-1Mo is the reference material for the Vessel System. Feasibility issues have been minimized by proposing a combination of plate and forging design. As discussed in Section 6.2.4, timely delivery of heavy section forgings is an issue and NGNP project will have to move quickly on initiating work with JSW.

Welding of mod 9Cr-1Mo is also an issue but weldability actions carried out by AREVA in the past few years indicate that welding of heavy section products should be fully achievable (even though optimization of welding products and welding parameters is still required).

Graphite Feasibility

There is no feasibility issue associated to the mechanical design of graphite core components. Feasibility lies more on the availability of material properties of the new grades envisioned for VHTR design (in particular properties of irradiated material) and on the availability of design rules approved by ASME Code Committee and by the Regulator.

6.5 Critical SSCs

The following sections lists, in addition to the feasibility issues already identified in Section 6.4, the Structures, Systems and components for which a particular attention will need to be paid in the context of the design and qualification work.

Control Rod

The selection of composite materials as control rod cladding requires significant R&D actions to qualify this component and facilitate its approval by the Regulator.

Gas Circulators

The design of 5 MWe circulators is slightly above state of the art technology (about 4 MWe) and a few developments will be required to achieve a design suitable for the VHTR. The option taken of multiloop design is however favorable for the feasibility of this component.

Hot Ducts

Technology is already available for transporting gas at high temperature. It will need to be verified for the Primary Hot Gas Duct that the feasibility of the metallic concept could not be challenged by the presence of hot streaks coming from the hot gas plenum.

Hot Isolation Valves

Such type of component has already been qualified in the context of the former German HTR program but it will need to be checked that the environment proposed on the secondary side will not justify significant design adaptations.

Fuel Handling System

Contrary to the LWRs, the Reactor Vessel remains closed during refueling of a prismatic HTR in order to avoid the oxidation of the graphite components. This requires the implementation of an original and relatively complex system for handling the fuel blocks in and out of the vessel. The performances and the reliability of this system are of prime importance to the plant availability and more analyses are still necessary to confirm that the reference system design selected during the preconceptual phase can meet the corresponding requirements.

7.0 REACTOR SUPPORT SYSTEMS

7.1 Shutdown Cooling System

7.1.1 Proposed Design

The primary function of the Shutdown Cooling System (SCS) is to transport heat from the core to the environment during shutdown conditions when the Primary Heat Transfer System (PHTS) is not operational and during refueling operations. The SCS system may also be used to support cooling of the IHX, and potentially other components, during recovery from conduction cooldown events. In order to fulfill these functions, the SCS must be able to function with the primary system in either the pressurized or not-pressurized condition.

Because this system interacts directly with the primary coolant within the reactor vessel, there are several design considerations imposed. During normal PHTS operation, the SCS system should not significantly impact core bypass flow. In addition, the SCS should be designed to retain helium and radionuclides within primary pressure boundary and to limit ingress of contaminants into the primary circuit.

The SCS is based on the GT-MHR design and consists of three heat transport circuits in series designed to remove heat from the Reactor Core System and transfer that heat to the ambient air. The general configuration and physical relation to the reactor is shown in Figure 7-1. The first circuit is in parallel with the plant PHTS across the Reactor Core System and consists of a helium-to-water heat exchanger, an electrically powered gas circulator and a shutoff valve. The second circuit is a closed pressurized water heat transport loop that runs from the helium-to-water heat exchanger to a water-to-air heat exchanger. The water is circulated by conventional electrically powered pumps, and the ultimate heat sink is an air-blast type heat exchanger with electric fans.

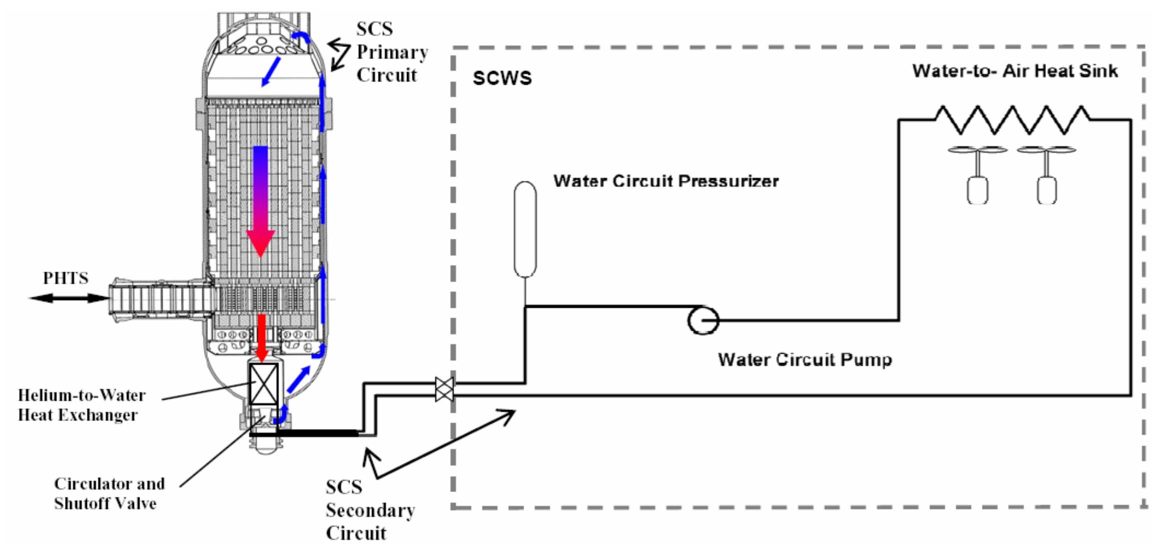


Figure 7-1: Shutdown Cooling System Configuration

When SCS is providing forced flow, hot helium from the reactor core outlet plenum flows through the center of the core support structure and into the Shutdown Cooling System Heat Exchanger (SCSHX). Once cooled, the helium is drawn downward through the Shutdown Cooling Helium Shut-off Valve (SCHSV) to the Shutdown Cooling Helium Circulator (SCHC). The SCHC circulates the helium and discharges flow to the reactor vessel bottom head cavity. From there, the pressure head forces the flow through the internal passage formed by the core support structure. The flow reaches the core inlet plenum, and the circuit is completed as the helium follows downward through the reactor core and reflector.

The Shutdown Cooling Water Subsystem (SCWS) is the second circuit of the three heat transport circuits in series. Water flows in a pressurized, closed loop from the SCSHX to the Shutdown Cooling Water Heat Sink (SCWHS), where the heat is transferred to the ambient air. The SCWHS is a water-to-air heat exchanger with forced air flow over finned tubes. The air flow is the third heat transport circuits in series. Flow is driven by a bank of electrically powered fans. The cooled water is then returned to the SCSHX by the SCWS pump. The SCSHX can be isolated from the SCWS at valves which define the boundary of the SCWS.

The SCWS operates in the liquid phase under all design conditions. When SCS is operating in forced cooling, overall system control is maintained by varying the motor speed of the SCHC to assure a setpoint value of the water exiting the SCSHX with sufficient margin to prevent boiling in the SCWS. When SCS is not operating, the water-to-air heat exchanger heat removal rate is controlled by louvers on its enclosure and by its fans to limit parasitic heat loss.

All of the components of the system are conventional and well understood, including pumps, valves, heat exchangers, circulators.

7.1.2 Alternate System Possibilities

The proposed NGNP reactor, while based on the ANTARES preconceptual design, does have some differences. These differences are primarily driven by the slightly different operating conditions specified for the NGNP, particularly required operating temperatures. One key difference is the configuration of the IHX systems. The proposed NGNP design employs multiple, tubular IHX trains to transfer heat to the PCS rather than the single compact IHX of ANTARES.

Based on the use of multiple, tubular IHXs for the NGNP design a question can be posed: Can the required SCS functions be preformed by the Startup and Decay Heat Removal System, thereby eliminating the need for the SCS as a separate system?

Such a configuration may be possible because:

- The tubular heat exchanger design is significantly more robust than the compact heat exchanger design utilized for ANTARES. That is, the wall thickness of the tubes is much greater than the web thickness within the compact heat exchanger, resulting in much lower operational stresses. Such a design should be able support operation at higher temperatures without unworkable design constraints being imposed due to high temperature creep.
- The multi-loop IHX design presents the possibility of operational configurations whereby selected modules can be isolated while maintaining required cooling flow through other IHX trains.

Figure 7-2 presents a schematic representation of the relationship between SDHRS and SCS.

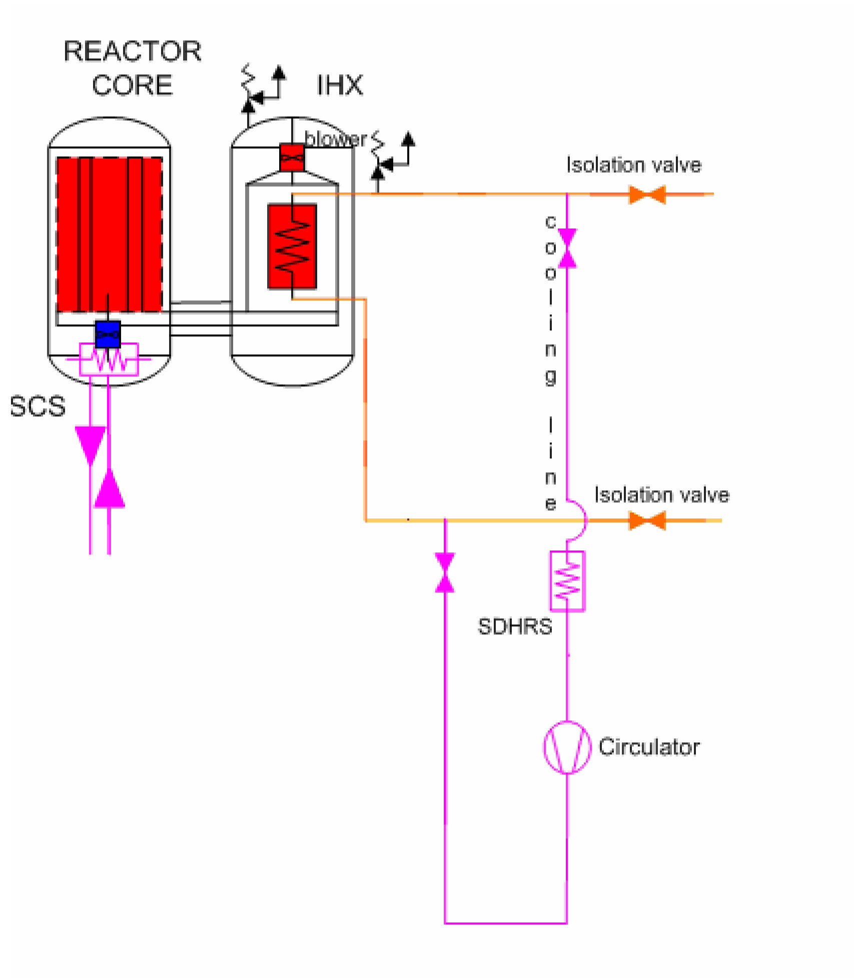


Figure 7-2: SCS and SDHRS Configurations

Implementation of this design change would have several advantages, including simplification of the reactor design by removing a potentially redundant system and elimination of the helium-to-water heat exchanger within the reactor vessel, which may reduce the impact of certain accident sequences.

It is recommended that this proposed system configuration be considered during the Conceptual Design phase of the NGNP project.

7.2 Reactor Cavity Cooling System (RCCS)

The primary functions of the Reactor Cavity Cooling System (RCCS) are to protect reactor cavity concrete, including RPV supports, from overheating during normal operation and to provide an alternate means of heat removal from the reactor system to the environment when neither the PHTS nor the SCS are available.

The proposed NGNP reactor will use a passive approach to fulfilling these functions through the use of a steel reactor pressure vessel (RPV), which is not insulated in its cylindrical part, to provide a decay heat removal path. During normal operation, heat lost through the uninsulated vessel is extracted by the ventilation systems and by the cavity cooler of the RCCS, depicted in Figure 7-3. During normal shutdown, decay heat is removed through the Primary Heat Transport System or through the Shutdown Cooling System. However, in the event that these paths are unavailable, decay heat is transferred by passive means through the RPV to RCCS panels. For example, during a depressurized conduction cooldown event, decay heat is removed by conduction through the graphite reflector and by radiation and natural convection from the uninsulated part of the RPV.

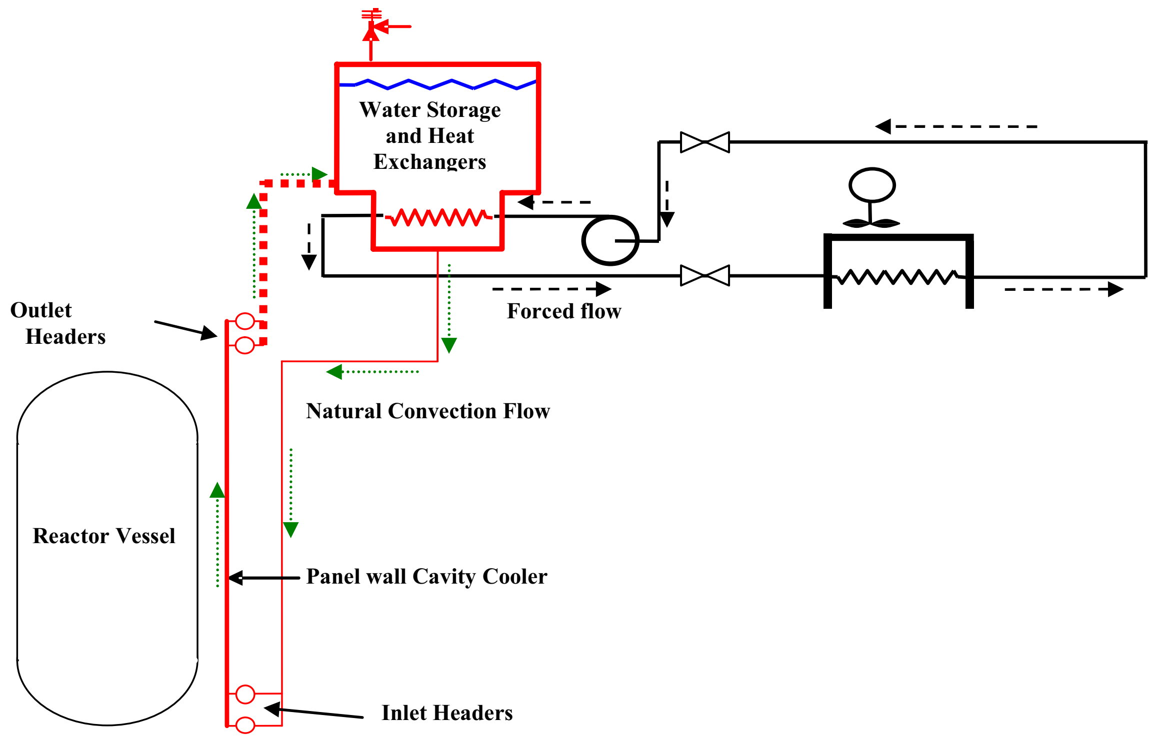


Figure 7-3: Reactor Cavity Cooling System Schematic Representation

The cooling systems for the Reactor Cavity, which receive the heat transferred from the vessel, include the cavity cooler panels. This cavity cooler surrounds the reactor pressure vessel at a distance of approximately 1.5 m to enable access for vessel ISI. This panel wall functions as a compact air-to-water heat exchanger. It takes the form of a closed tube wall about 16 m long and is arranged at a distance of approximately 10 cm in front of the concrete wall of the reactor cavity. The space between cavity cooler and concrete is sealed to prevent ingress of hot air streams. The tubes are installed vertically and joined by welding to form a welded membrane panel wall. Two inlet headers (for two independent trains) are located in the lower section of the reactor cavity and, accordingly, two outlet headers in the upper section. The inlet and outlet piping is routed separately through the wall of the reactor cavity. It is therefore possible to isolate each of those two trains from a safely accessible point. The eight segments of the cavity cooler are placed in the reactor cavity and are bolted down, and the header segments are installed by welding. Each segment is suspended from two points at the top of the reactor cavity. The headers are axially fixed, circumferentially guided and free to move radially. Expansion loops compensate for thermal axial expansion of the cooling tubes. This solution was chosen because it needs little room in the reactor cavity and has a very good heat extraction capability.

The process flow for the RCCS system is shown schematically in Figure 7-3. When the system is operating normally, water is circulated by natural circulation through inlet headers to the panel wall heat exchangers in the reactor cavity and returned via outlet headers to water storage tanks. Heat is removed from the water storage tanks by a dedicated circulating water loop with active forced flow that picks up heat in a water-to-water heat exchanger submerged in the storage tanks and rejects the heat to the atmosphere in a forced draft, wet, water-to-air heat exchanger. This system contains both safety and non-safety components. The panels, loop piping, tanks and heat circulation within the tank are all safety related. The forced flow loop, pumps and air blast heat exchanger are non-safety related. If the non-safety pumped circuit heat rejection path is not available during conduction cooldown events, then water stored in tanks above the reactor vessel continues to flow by natural circulation to the panel wall heat exchangers, and heat is rejected by vaporizing the water inventory and venting steam through the water tanks to the atmosphere.

The RCCS is required to operate continuously in all plant states, including shutdown following loss of forced reactor cooling by the PHTS and SCS with simultaneous loss of pumped circulation of RCCS cooling water and a safe shutdown earthquake. All safety-related components and pipes of the RCCS are designed against seismic loads. All components and pipes inside the reactor building, including the connections for emergency water supply are designed against external events, e.g., aircraft crash or pressure waves.

Accordingly, the RCCS decay heat removal function ensures both investment and safety protection. Since the RCCS is expected to be relied upon to meet safety acceptance criteria, the components of the system shown in red in Figure 7-3 are classified as “safety-related”. Note that while the function of water pressure boundary integrity through the water-to-water heat exchangers submerged in the storage tanks is “safety-related”, the functions of heat transfer and powered circulation of water are not “safety-related”.

Use of an uninsulated reactor vessel coupled with a water-cooled panel heat exchanger as a core cooling mechanism for accident conditions has not been demonstrated on a commercial scale. Nonetheless, the operating principle and the controlling physics are very simple and well understood. The basic components of the system are also convensitonal and well understood. Proper design and sizing of the system will require a demonstrated understanding of key heat transfer parameters for the vessel wall and panel surfaces.

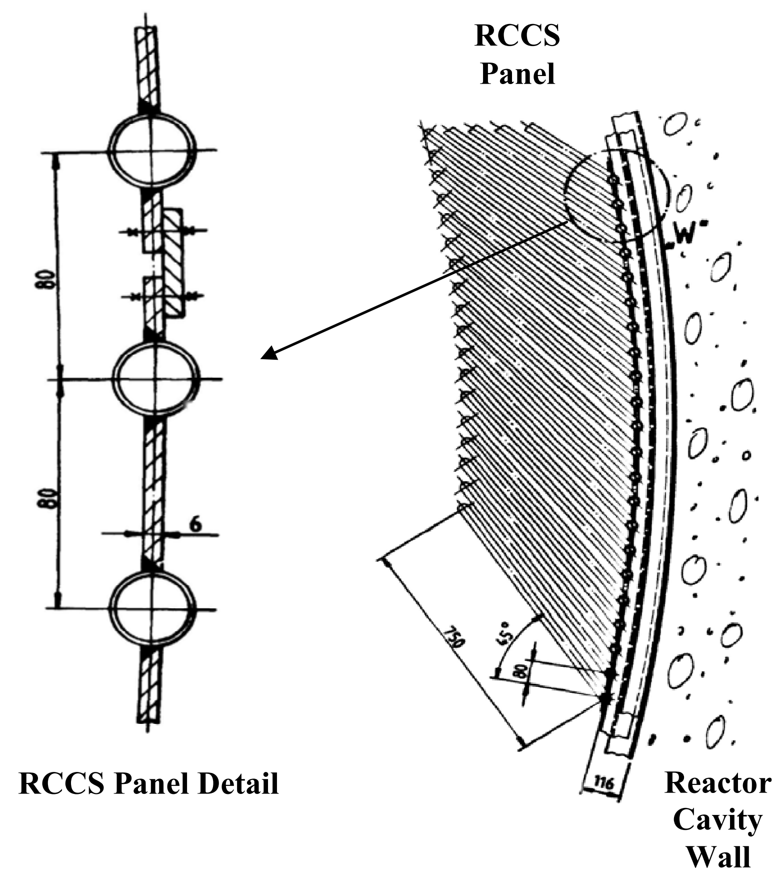


Figure 7-4: RCCS Cooling Panel Configuration

7.3 Startup and Decay Heat Removal System

The Startup and Decay Heat Removal System (SDHRS) is connected to the secondary coolant system between an IHX and the associated Main Isolation Valves. This system provides a heat removal path for conditioning during plant startup and shutdown. In addition, this system can provide heat removal during normal shutdown modes when the PHTS and secondary circuit are available up to the main isolation valves. Operation of this system requires forced convection in the primary system via the Main Circulator associated with the IHX to which the system is attached.

The components and key interfaces of the SDHRS are shown in Figure 7-5. For the NGNP, it is envisioned that this system will be installed with the option to be connected to the secondary system via the piping adjacent to each of the IHX modules, allowing alternate cooling paths to be implemented. This flexibility will allow maintenance on each IHX loop while maintaining operation of the SDHRS system for decay heat removal during shutdown conditions.

The main components of the SDHRS system are a secondary working fluid-to-atmosphere heat exchanger , a circulator, a series of valves for each IHX module, and the required piping and control systems.

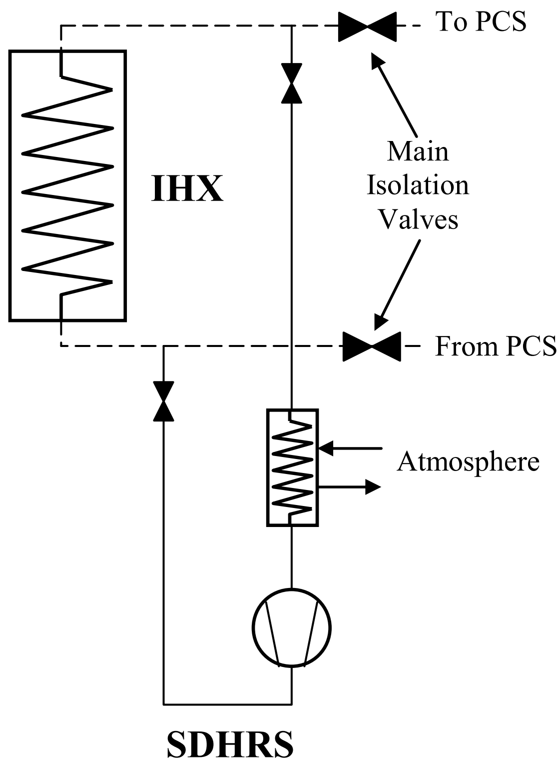


Figure 7-5: SDHRS Components and Interfaces

7.4 Fuel Handling and Storage Systems

The primary functions of the Fuel Handling System are to:

- During refueling shutdown conditions, remove irradiated fuel, reflector blocks, and other core components from the reactor vessel and transfer them to the Spent Fuel Storage System (SFSS).

- During refueling shutdown conditions, receive new and irradiated fuel, reflector blocks, and other core components from the SFSS and place them in the reactor vessel.
- Maintain positive control and traceability of all Special Nuclear Material (SNM) during fuel handling activities, including verification of core configuration.

In performing these functions, the system must be able to:

- Provide shielding to protect workers from radiation
- Limit ingress of potential contaminants into the Primary Helium System
- Accomplish refueling within planned outage time allocations

The Fuel Server portion of the Fuel Handling System has been described only as a design concept at this point. The remainder of the Fuel Handling System components, including the Fuel Elevator, Adaptor Plate, and Fuel Handling Machine, and based on the fuel handling systems in use at the Fort St. Vrain reactor. In addition, the HTTR reactor utilizes a similar set of components.

7.4.1 Operational Conditions

Irradiated fuel will be a strong source of gamma radiation, with surface dose rates on the order of $10x10^7$ R/hr. Unlike light water reactors that handle irradiated fuel under water, the NGNP fuel will be handled in helium or air. To protect workers during the refueling operation, all Fuel Handling equipment will be equipped with shielding material to attenuate the dose rate in the maintenance hall to below 2.5 mR/h. The FHS is configured to minimize the time required to accomplish refueling and to thereby minimize the risk of worker exposure.

The FHS equipment must also protect workers and the public in the event of an external hazard event during refueling operations. Such an event might be an earthquake, fire, aircraft crash, or extreme weather event. Generally, this functional requirement falls only on the Fuel Storage Server (FSS). The Fuel Handling machine (FHM) only handles fuel inside the reactor pressure vessel, and the Fuel Elevator (FE) handles fuel when it is below the floor of the Reactor Auxiliary Building or inside the Fuel Storage Server. Failures or breaches of the FHM or FE induced by external hazards might cause a loss of helium, excessive air ingress, or the dispersion of some plate-out and dust activity. However, such failures are not likely to result in the overheating or dispersion of fuel or fuel particles. The Fuel Storage Server addresses these risks through several design selections.

7.4.2 Summary Description of the Fuel Handling Process

The FHS consists of a series of machines and devices that are capable of carrying out the functions of the system. The system is based on the design of Fort St. Vrain and GT-MHR with the exception of the Fuel Storage Server (FSS) instead of Fuel Transfer Casks. The FSS reduces the estimated refueling time.

The HTR core is assembled inside the reactor by stacking up fuel and reflector blocks. Fuel and reflector blocks are in the form of hexagonal prisms, including a central handling hole which allows gripping the element, chamfered corners at the top and bottom surfaces, dowels for alignment and means for identification of each block. Each position in the core can be identified by a unique address.

A functional block diagram of the Fuel Handling System is given in Figure 7-6.

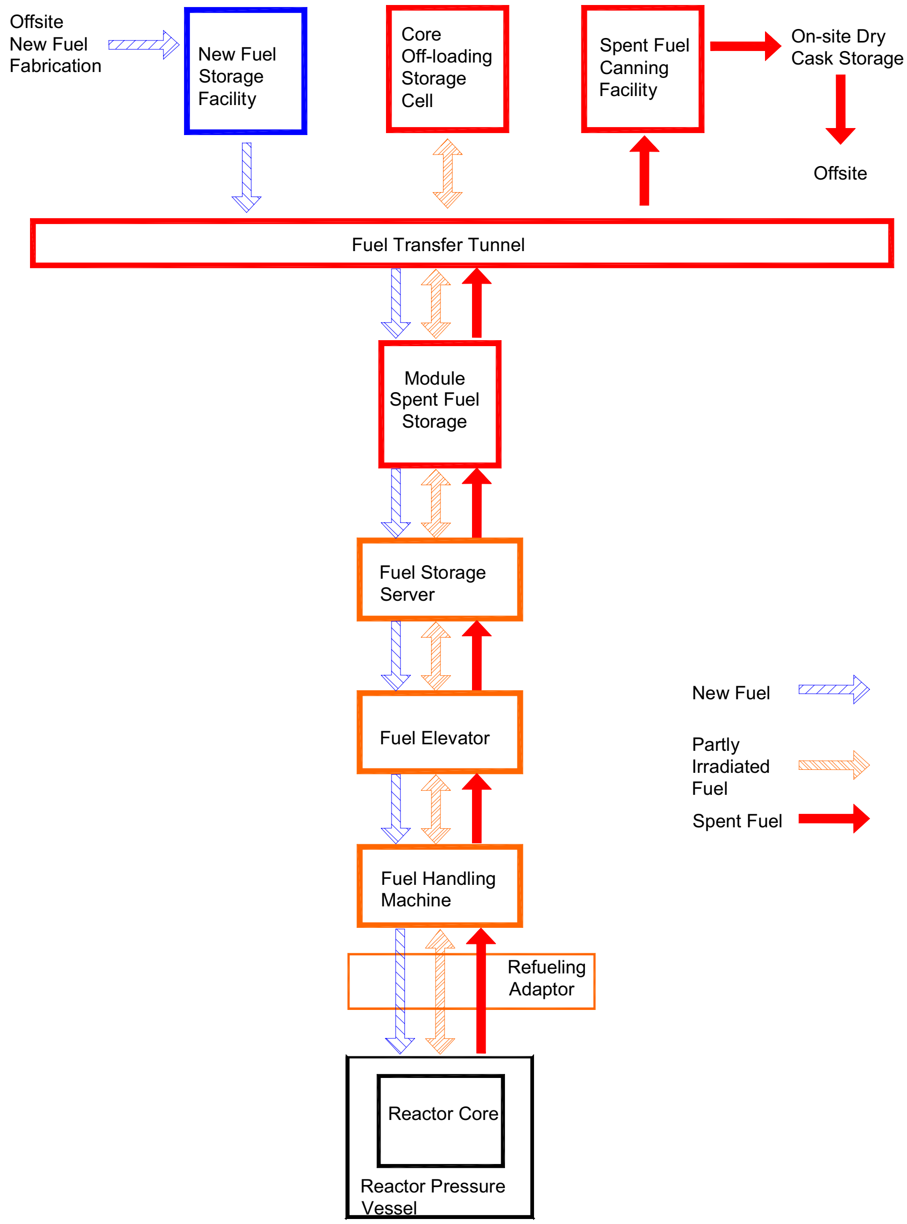


Figure 7-6: Fuel Handling System Block Diagram

Refueling is accomplished using robotic machines that remove core blocks and transfer them to the adjacent Local Spent Fuel Storage cell (LSFS), which is part of the Spent Fuel Storage System (SFSS). Subsequently, new and irradiated fuel blocks are delivered to the FHS which then places them into the reactor vessel. Access to the core region is provided by a fuel elevator which is mounted at the central position. All fuel elements and reflector blocks go through this central position. A fuel handling machine for the in-vessel fuel handling is installed at the position of an inner control rod. From there the fuel handling machine can unload and reload one sixth sector of the core. The steps are listed below.

1. New fuel and reflectors are placed into the LSFS.
2. The reactor is shutdown and depressurized. The reactor cavity shield plug is removed.
3. Seal welds and covers are removed from all control rod drive nozzles, and the fueling adaptor (FA) is installed. Its weight and the weight and loads from the Fuel Elevator (FE) and Fuel Handling Machine

(FHM) are carried by a support skirt mounted on the reactor vessel. The weight of the FA, FE, and FHM are transmitted to the reactor building through the reactor vessel and vessel supports.

4. The Fuel Storage Server and Fuel Elevator are installed, providing a shielded pathway connecting the reactor vessel and the MSFS.
5. The Inner Control Rod Drive Mechanism (CRDM) for Sector 1 is removed and the Fuel Handling Machine (FHM) installed. Outer CRDMs and guide tubes in Sector 1 are raised and jacked clear of the core, to provide access to the reflector columns to be replaced (usually 25% of replaceable reflectors will be replaced each outage)
6. Sector 1 is completely un-stacked, layer by layer. Blocks are first grappled and lifted by the FHM, then placed into the FE, which raises them to the Maintenance Enclosure operating floor level. They are then transferred by the Fuel Storage Server (FSS) to a floor valve where they are transferred to the control of the Spent Fuel Storage System (SFSS).
7. Sector 1 is then completely re-stacked, using a mix of new and partly-irradiated fuel elements.
8. Outer control rods for sector 1 are lowered and inserted into the core
9. The FHM is removed from the inner CRDM nozzle for sector 1, and the inner CRDs are installed and inserted into the core
10. Steps 5 through 9 are repeated for sectors 2, 3, 4, 5, and 6
11. The FE is withdrawn and the FSS moved to its storage location
12. The Fueling Adaptor is removed
13. CRD nozzle covers and seal welds are replaced
14. The reactor is re-pressurized and checked for leakage. The Helium Purification System is operated to remove coolant impurities, and the reactor can be restarted.

7.4.3 System Configuration

The components of the FHS are mobile, and may be shared among all reactor modules in multi-module plants. The FHS also interacts closely with the control rod removal cask. This cask, part of the Reactor Service System, mates with the Fueling Adaptor, and is used to remove and reinstall central column instrumentation and both inner and outer control rod and reserve shutdown assemblies. These in-core components are strong sources of gamma radiation, and are stored in wells in the Reactor Building during refueling. The Spent Fuel Storage System also interfaces closely with facilities for storage of graphite elements and waste treatment systems.

In particular the FHS consists of the following devices and facilities:

7.4.3.1 Fueling Adaptor

The Fueling Adaptor (FA) is a mechanical device that extends the CRDM nozzles. The FA is a welded steel structure that fits over the CRDM nozzles and creates a soft seal around each nozzle, to permit the removal of CRDMs and insertion of the FE and FHM without opening the reactor coolant boundary to the RAB maintenance hall. The weight of the FA is carried on the FHS support skirt, which is in turn supported by the reactor vessel.

Before the FA is installed, the reactor must be shutdown and depressurized, and the reactor cavity shield plug removed. It is also necessary to cut seal welds, if they are used on the CRDM nozzles, and remove mechanical closures (bolts). This strategy places some requirements on the design of the control rod drive mechanisms. The CRDMs must be designed so that a reasonable seal is created between the CRDM nozzle and the drive mechanism. It is also necessary that the rod controls and instruments be connected and operable with the nozzle covers removed and the FA in place. For the outer rods, it must be possible to raise the rods to a full up position and then physically lift the rods, CRDM, and guide tubes clear of the core, using a jacking mechanism built into the FA. It must be possible to re-make outer rod instrument and control connections after they have been lowered back into place.

The FA will be designed with soft or inflatable seals which operate on the outside surface of the CRDM nozzle. On the inner CRDM nozzles and the central column nozzle, the FA will also have horizontal valves. Conceptual valves have been located at three separate levels, so the gate enclosures and operators do not interfere with each other. Inner control rod drive mechanisms must also be designed to create a soft seal with the CRDM nozzle, and to be operable with the hard nozzle covers removed. As each sector is refueled, the inner CRD is raised to its topmost position, and the rod and CRDM are then physically lifted out of the core and into a handling cask, so that the FHM can be inserted. During this process, there will necessarily be a short period when the reactor vessel is open to the CRDM handling cask.

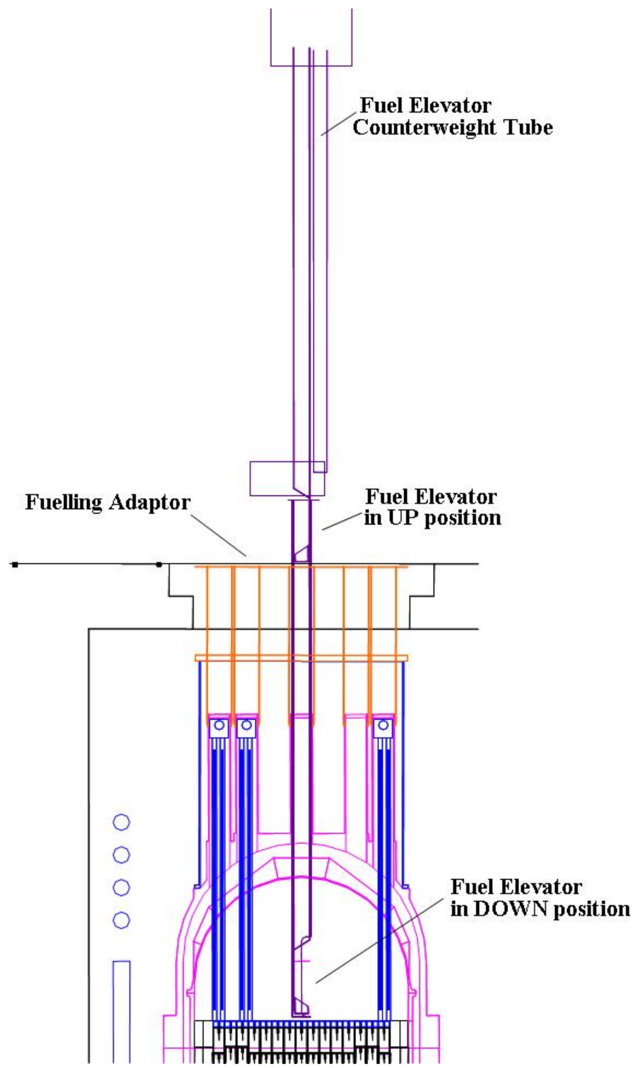


Figure 7-7: Fueling Adaptor and Fuel Elevator

7.4.3.2 Fuel Elevator

The Fuel Elevator (FE) is a machine that can be inserted through the Fueling Adaptor into the reactor vessel. The central penetration on the reactor head, which normally contains in-core instruments, is used for the FE. The FE supports blocks from the bottom, and has a vertical range of motion. It can also rotate, so that the elevator car can face the FHM at any sector, and the car can face the Fuel Server in the raised position. Because fuel in the fuel elevator is always inside the reactor vessel or the Fuel Storage Server, the FE does not need to provide any shielding or specific ventilation capability. A soft seal between the fuel elevator and the Reactor Vessel is sufficient to maintain the reactor coolant boundary during refueling.

7.4.3.3 Fuel Handling Machine (FHM)

The Fuel Handling Machine (FHM) is a robotic manipulator that can be inserted into an inner control rod drive penetration. It is equipped with a grapple probe that can be inserted into the handling hole in the top of any hexagonal block. The probe can be expanded to engage the block so the machine can lift it. The FHM is equipped with a pantograph-like mechanism giving it the capability to extend the grapple out to a radius sufficient to reach all the blocks within a sector, including all hexagonal reflector blocks (Figure 7-8). The FHM can rotate, extend, and raise/lower, so that it can access all the hexagonal blocks in a sector from an inner CRDM nozzle. The FE must be designed so that it can rotate and always face the FHM when lowered, and always face the FSS when raised.

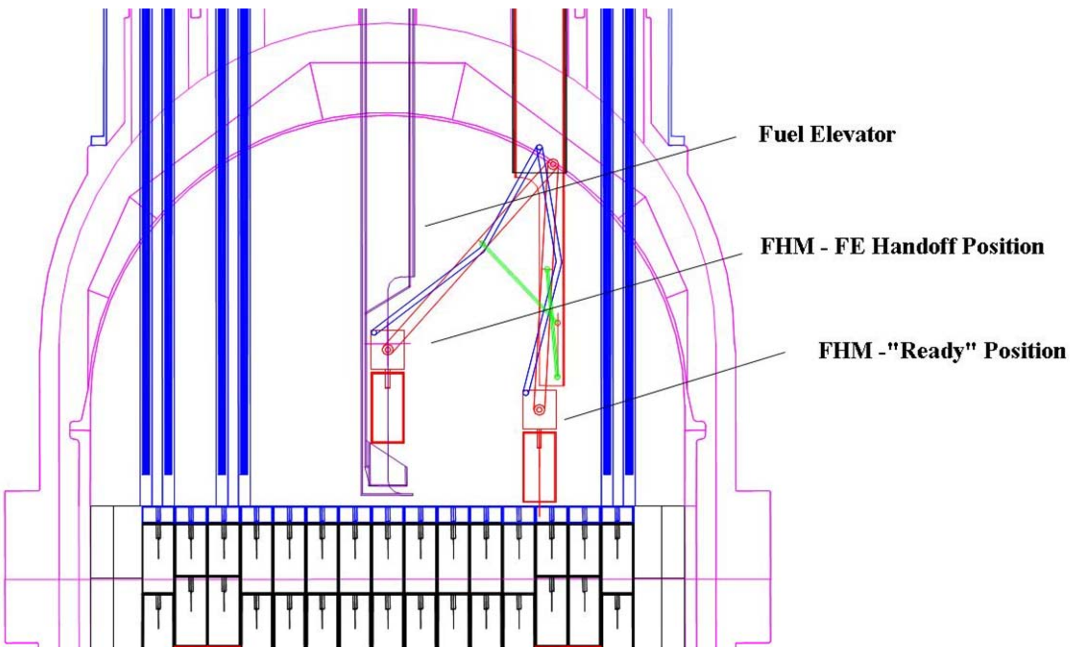


Figure 7-8: Fuel Handling Machine

Blocks are grappled and raised one at a time. When the FHM is in its uppermost operating position and oriented toward the center of the core, it can extend and place the block in the Fuel Elevator. When returning blocks to the core, the process is reversed, with the FHM grappling a block from the Fuel Elevator, locating the block over the proper column, and lowering it. The rotational position of the block is maintained so that CRDM holes and dowels are correctly aligned. To accomplish this, the FHM also has the ability to rotate the block on the grapple axis, and to move the grapple probe over small horizontal distances.

The FHM can move in Z, R, theta (around the inner CRDM nozzle axis) and phi (around the grapple head axis) as well as fine motion at the grapple head. The FHM concept is expected to include a rigid, non-rotating tubular mast. Inside this mast, a rotating tubular mast can be lowered from a "ready" position above the top of the core to the elevation necessary to grapple the lowermost element. Raising or lowering the inner mast accomplishes changes in Z, and rotation of the inner mast accomplishes changes in theta.

The complete machine, when closed at R=0, must fit through the CRDM nozzle. Relative vertical motion of bearings supporting the grapple arms will change the angle of the supporting arms, and change R. Phi and grapple head X and Y motions are accomplished by electromechanical drivers located on the grapple head, as is the expansion and contraction of the grapple probe.

7.4.3.4 Fuel Storage Server (FSS)

The Fuel Storage Server (FSS) is a proprietary AREVA fuel element handling device designed to replace the cask-based handling systems envisioned for most past prismatic HTR systems. Fuel elements are transferred between the FE and the module Spent Fuel Storage area utilizing the FSS The FSS was developed to decrease refueling times and to reduce worker exposure during refueling operations.

7.4.4 FHS Component Performance and Refueling Time

Refueling outages will occur after approximately 417 full-power days of operation; however, design burn up of NGNP fuel is achieved only after fuel has been irradiated for two cycles, so half of the 1020 fuel blocks are replaced in each outage. The core must be completely un-stacked and re-stacked using a mix of new and partly spent fuel.

Many options are possible, including placing new fuel in alternating columns, moving fuel from the inner part of the core to the outer part, and others. However, the fuel management scheme selected has no impact on the estimated duration of a refueling operation.

The estimated refueling time is 20.9 days and includes approximately 25% contingency. This estimate assumes that 25% of the replaceable reflector blocks are replaced during each refueling outage. This refueling duration satisifies the required availability allocation. In addition, AREVA studies have also demonstrated that the average refueling outage duration can be reduced to 18.4 days if a wholesale reflector replacement every fourth refueling is implemented instead. The overall performance of the FHM and therefore the ANTARES refueling time is based on a number of assumed performance values, all of which must be further developed during the design process. Table 7-1 presents a summary of these performance values used in the refueling time estimate study. The overall performance is most sensitive to the assumption that the FHM can perform multiple motions (R, Z and theta) at the same time. Since a complete one-sixth sector of the core is un-stacked before any re-stacking starts, the estimated refueling time is not dependent on the fuel management scheme selected.

Table 7-1: FHS Performance

	Capacity, Blocks	Degree of Freedom (DOF)	Range of Motion m (ft)	Maximum Velocity m(ft)/s
Fuel Handling Machine	1	Z	12.7 (41.7)	0.25 (.82)
Fuel Handling Machine	1	R	1.67 (5.48)	.05 (.16)
Fuel Handling Machine	1	Θ	390°	15 deg./s
FHM Grapple Head	1	Φ (local)	340°	15 deg./s
FHM Grapple Head – Confirmation of operation	1		Engage, disengage	30 sec
Fuel Elevator	1	Z	13.9 (45.6)	0.4 (1.3)
Fuel Elevator	1	Θ	390°	15 deg./s
Fuel Storage Server				
Fuel Storage Server Grapple Head	1	Z	Engage, disengage and 0.2 (.65)	60 sec cycle time

7.4.5 Design Issues and Subsequent Steps

In general, the development of the overall NGNP refueling concept is not completed additional studies and detailed design work will have to be done in the Conceptual Design Phase. These should address, in particular, the following items:

- Consequences of the reference FHS solution on the CRDM and the CRD nozzles with the particular concern of testing the leak tightness of the new welds after refueling
- Definition of the necessary design and administrative measures to cope with plant security issues

These studies will also confirm the current estimate for the refueling duration.

7.4.6 Spent Fuel Storage System

There are three Spent Fuel Storage modules that support the NGNP reactor.

The Local Spent Fuel Storage system provided near-reactor storage for fuel during refueling activities and is directly accessed by the FSS. This system consists of 200 wells, deep enough for 5 full sized fuel or reflector elements, designed to hold all of the fuel and reflector elements required for normal refueling activities. It also includes a fuel manipulator and elevator system which accesses these wells and interfaces with the FSS.

The Core Offload Storage system provides sufficient space for offload of the entire reactor core to support in-vessel inspection activities and other operational events which may require emptying of the reactor vessel. This system has a storage capacity of 1050 fuel elements and 2700 reflector elements. The Intermediate Term Spent Fuel Storage system provides a space within the reactor services building for storage of fuel discharged from up to ten cycles of operation of the NGNP plant. This system has a capacity of 9000 fuel or reflector elements. Each of these systems includes appropriate fuel manipulators for servicing the fuel storage wells.

The Spent Fuel Storage system also includes a cask loading and preparation area to support eventual movement of the fuel to either long term storage on-site or off-site repository storage. This area will also include fuel inspection stations and equipment to support post-irradiation evaluation of fuel elements as necessary.

7.4.7 New Fuel Handling and Storage

The new fuel storage and handling system consists of equipment for acceptance of new fuel from the fabrication vendor, including necessary receipt inspection stations and temporary storage locations for fuel awaiting transfer to the Local Spent Fuel Storage module for staging in support of refueling activities.

7.5 Spent Fuel Cooling System

The primary function of the Spent Fuel Cooling System (SFCS) is to accept decay heat from spent fuel elements and reject it to the environment.

Fuel elements removed from the reactor after irradiation, either for re-insertion in subsequent reactor cycles or for ultimate disposal in an off-site facility, are stored in dry tubes immersed in a circulating water cooling system. This system consists of a pool in which the tubes are immersed, a circulating pump, a water-to-air heat exchanger which rejects the decay heat to the atmosphere, and required piping and control systems.

7.6 Nuclear Island Cooling System

The primary function of the Nuclear Island Cooling System (NICS) is to remove heat from the following systems:

- Helium Purification System
- Sampling Systems
- Helium Transfer and Storage System

In performing this function it also provides a barrier between these potentially radioactive systems and the environment.

This system transfers the waste heat from the above systems to the atmospheric and is comprised of circulating pumps, water-to-air heat exchangers, and associated piping and control equipment.

7.7 Helium Services System

The configuration and major components of the proposed NGNP Helium Services System are effectively identical to those currently in use or used in various other helium cooled reactors, including AVR, THTR-300, HTTR, and HTR-10. Use of this system for the NGNP will require sizing of the various components for the desired flow rates.

7.7.1 Helium Purification Train

The primary functions of the Purification Train are:

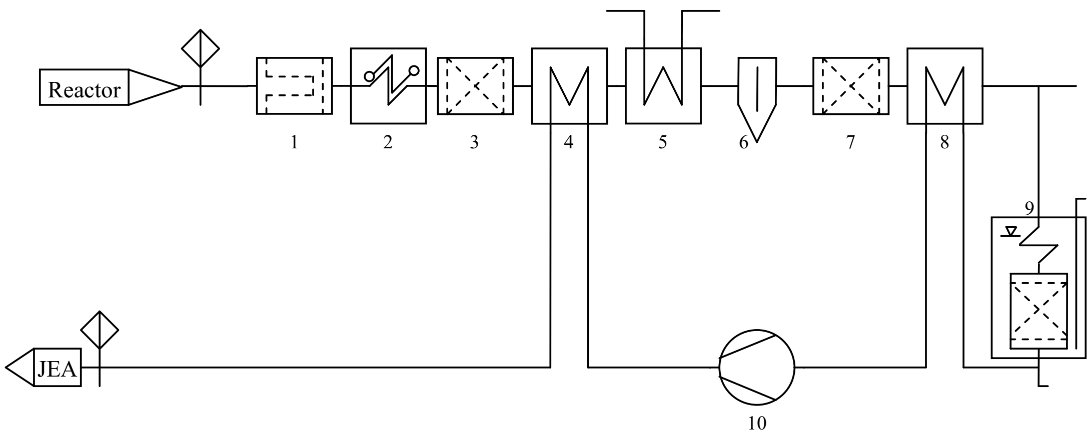
- Removal of chemical and particulate contaminants from the primary coolant
- Supply of purified helium to appropriate systems

Since helium is used as the primary coolant, a helium purification system is required to provide the necessary degree of helium purity. Oxidizing contaminants, in particular, may not exceed predetermined limits established in the specification. In detail, the helium purification system has the following functions:

- Removal of particulate and gaseous contaminants from the primary coolant to maintain design values, in particular for H_2O, CO, CO_2, N_2, H_2, CH_4

- Removal of tritium
- Removal of other radioactive contaminants from the helium, especially before transfer to the purified gas store (Xe, Kr, Ar)
- Start up purification of the primary system before initial start up and after inspections and maintenance
- Purification of newly delivered helium

A flow diagram for the Helium Purification System is shown in Figure 7-9



1	Dust removal filter	6	Water separator
2	Heater	7	Molecular sieve
3	Copper oxide bed	8	Recuperative heat exchanger
4	Recuperative heat exchanger	9	Low temperature adsorber
5	Helium-to-Water cooler	10	Blower

Figure 7-9: Helium Purification System Flow Diagram

The purification train is designed to handle approx. 5 % (150 kg/h) of the primary system inventory per hour during normal operation. The purification system operates at pressures between 1 and 70 bar. The system always purifies the same volumetric flow, the mass flow changing according to inlet pressure and inlet temperature. Inlet temperature can range from 30 to 500°C.

Anticipated inlet contamination levels allow operation of a purification train for approx. 8000 hours before a regeneration cycle is necessary. This typically will occur during refueling activities. Regeneration of a train takes approx. 20 hours. Assuming an equilibrium state of helium contaminants as the initial condition, the reactor can be run for several days in normal operation without the helium purification system and pressurizing functions.

7.7.2 Helium Transfer and Storage Train

The primary functions of the Helium Transfer and Storage Train are:

- Accepting helium from filled auxiliary systems during depressurization activities
- Storage of radioactively contaminated helium
- Evacuation of primary and supporting systems
- Removal of helium from the primary and supporting systems and storage in a purified gas store

The most important components of the system are a purified gas storage tank, a purified gas compressor, and a purified gas receiver.

When helium from the primary system is dumped into the purified gas store, pressure in the primary system is first relieved until it has been equalized. Afterwards the compressor draws the remaining helium out of the primary system down to a pressure of 1 bar.

The purified gas store is filled with helium from tanker trucks in similar fashion. The process demands that some tanks are always kept at high pressure. The compressor circulates helium in store for this purpose.

The various helium loads are supplied with purified helium from the purified gas receiver. Pressurization of the primary system is performed by the pressure control and pressure relief system.

7.8 Radioactive Waste and Decontamination System

The Radioactive Waste and Decontamination system is designed to process all plant radioactive waste streams so that waste quantities are minimized, activity levels are managed, and worker exposure is kept ALARA

7.8.1 Radioactive Waste Management

Radioactive Waste Management consists of three individual Systems based on the form of the waste streams. These are:

1. Gaseous Radioactive Waste System (GRWS)
2. Liquid Radioactive Waste System (LRWS), and
3. Solid Radioactive Waste System (SRWS).

The SRWS provides for holdup, solidification, volume reduction and packaging of radioactive material prior to their shipment offsite for ultimate disposal. High level waste is packaged separately from low level waste and stored in shielded containers. Compactable waste is collected, sorted and where possible the volume is reduced (such as by compaction or incineration) before packaging in 55 gallon drums. Slurry from highly tritiated liquid waste or from high conductivity liquid waste is solidified in 55 gallon drums.

The LRWS collects and processes all potentially radioactive liquid waste generated within the reactor plant, before monitored discharge via the Plant Cooling Water System to the environment.

The LRWS contains two processing paths. The low tritiated, low conductivity waste stream is filtered/de-mineralized before monitored release via Plant Circulating Water. The liquid waste streams that are likely to be high in tritium or has high conductivity are collected in a tank where it is neutralized and the concentrated effluent is subsequently transferred as slurry to the SRWS for solidification.

The Gaseous Radioactive Waste System (GRWS) separates liquids and solids from the gas stream, then pressurizes and stores remaining gases until safe to release. The GRWS contains two processing paths. The waste stream from sources that are likely not radioactive or low in radioactivity is filtered for particulates and radioiodine before monitored release via process blowers to the Building HVAC exhaust. The gaseous waste streams that are likely to be radioactive such as from Gas Sampling System or the Helium Purification System, and low level waste streams not meeting discharge limits are collected in a vacuum collection tank or in surge tanks for an extended period of time to allow for radioactive decay prior to monitored release to the HVAC. If the activity levels in the discharge stream, after allowing it to decay, are still too high for atmospheric discharge, it will be routed back to the HPS for further treatment. Moisture condensed out in the vacuum collection tank or the surge tanks is collected and transferred to the LRW System. The plant gaseous effluent is documented via the HVAC radioactivity measurement sensors.

7.8.2 Decontamination System

During operation of nuclear power plant, parts and components have to be maintained and repaired. To conform to ALARA principles, the components and parts need to be decontaminated before repair or maintenance. To perform the necessary decontamination functions, high-pressure devices, a degreasing unit, and an ultrasonic unit will be used.

Two chambers with high-pressure cleaning devices are planned. One high-pressure device is operated at 130 bar with a liquid flow rate of 10 l/min and is used for parts and components that are easier to cleaned. The second high-pressure device is operated at 250 bar with a flow rate of 200 l/min and is used for parts and components which are heavily contaminated or foreseen for conventional discharge. This device is operated automatically because of the high pressure during operation to protect the operator. The high-pressure cleaning devices can be operated with hot water and/or with chemical additions. The high-pressure units are mainly used for larger parts and for components and parts to be discharged (conventional or as radioactive waste).

The degreasing immersion unit and ultrasonic unit are heated water baths, to which chemicals can be added to support the cleaning effect to the water in the bath. The ultrasonic bath and the immersion bath are mainly used for components and parts which have to be treated with care.

7.9 Plant Control System

Overall control of the NGNP facility is provided by the Plant Supervisory Control System. This system provides coordination between the Nuclear Heat Source, the power generating facility, and the Hydrogen Production Facility. It oversees overall load control for the plant, and it coordinates startup and shutdown activities. Actual control and protection functions within the nuclear heat source and within the hydrogen production facility are handled by separate dedicated systems. These systems receive load instructions from the supervisory control system.

Since the AREVA NGNP team's preconceptual design scope includes only the Nuclear Island and the Power Conversion System, detailed development of the Plant Supervisory Control System and the dedicated control and protection systems for the Hydrogen Production Facility and the High Temperature Heat Transport Loop has not been performed. The assumed top level plant control configuration is stated simply to provide the context for the remaining discussion of plant control and protection systems.

The anticipated top-level plant control configuration is shown in Figure 7-10. The PCS is controlled from the NHS control room and PCS control is integrated with the NHS control systems. This configuration assumes that the High Temperature Heat Transport Loop is controlled from the Hydrogen Production Facility. Information on the status of each system must be passed back to the supervisory control in order to coordinate plant operation. The development of the NHS/PCS control system and the H2/HTHTL control systems will have to be tightly integrated in order to coordinate which system has control authority over which process variables at the system interfaces.

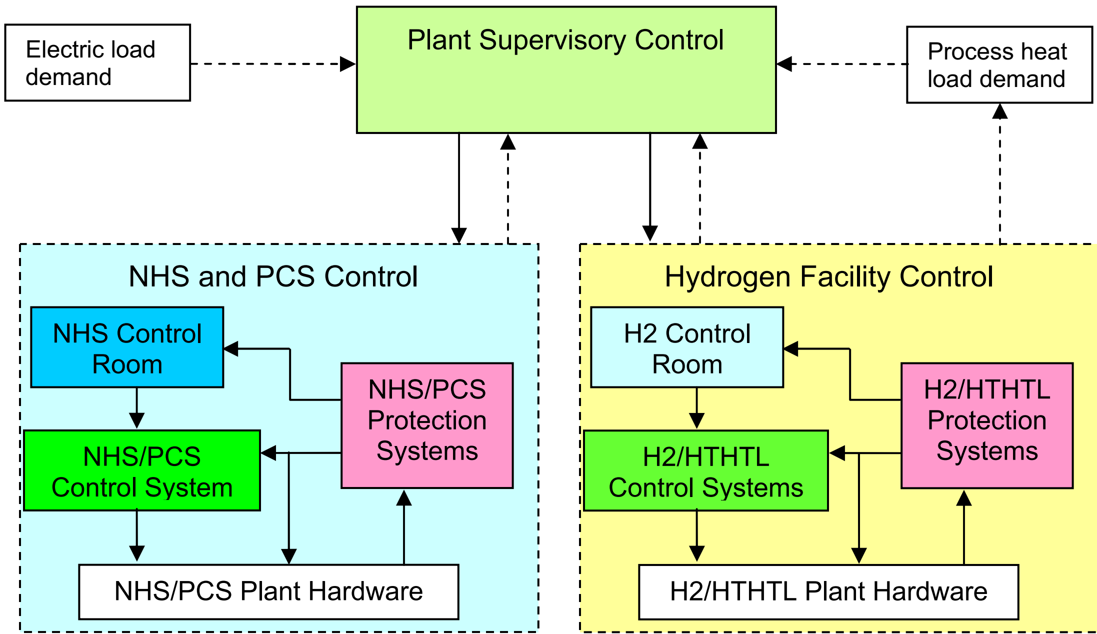


Figure 7-10: Simplified Overall Plant Control Configuration

The remainder of the plant control discussion is focused on AREVA's scope which includes the Nuclear Heat Source and the PCS. This discussion is largely based on the ANTARES control configuration.

The plant is monitored and controlled during normal operations by the Process Automation System (PAS). The PAS monitors and operates the mechanical systems of the plant in all conditions, as long as the PAS and the mechanical systems are operable, and insures that the plant is operated within its defined operating envelope.

The plant control systems are devised to support the principle of defense in depth. The PAS provides normal monitoring and control of the plant. In the event of PAS failure, or failure of equipment controlled by the PAS, the non-safety portions of the plant protection systems come into play, protecting the financial investment. Actions to protect key components and systems from damage are initiated. These actions also protect health and safety of the public by preventing damage to the reactor and radiological release. In the event that the non-safety portions of the plant protection system do not function as desired, the safety related Reactor Protection System (RPS) and supporting systems protect the health and safety of plant personnel and the public by tripping the reactor to reduce reactor heat generation. The supporting systems include the Reactor Control & Limitation System (RCLS), the Priority Actuator Control System (PACS), and the Post-Accident Monitoring System (PAMS). All of these systems supplement the inherent safety features of the plant. In particular, reactor power generation is reduced as core temperature rises, thereby allowing the cavity cooling system to remove all heat generated. An overview of the interrelated plant instrumentation and control systems is shown in Figure 7-11.

The PAS is functionally, physically, and electrically independent of the safety related systems, except for information that flows unidirectionally from the protection systems through appropriate isolation to the PAS so that control adjustments can be made in the portions of the plant that remain in operation following a reactor trip. This insures that the appropriate information is available to the operator at all times. All protection actions are fully automated and do not depend on the PCS or the operator.

The PAS uses industrially proven, distributed, microprocessor-based control platforms, a hierarchical data information network and human factored operator interfaces. The system integrates major plant instrumentation using highly reliable, multiple redundant data highways connecting control signals and data from sensors via remote field control stations located throughout the plant. All data and control functions can be monitored at the central control room, providing plant operators and managers with real-time information on the status of the plant and all of its systems.

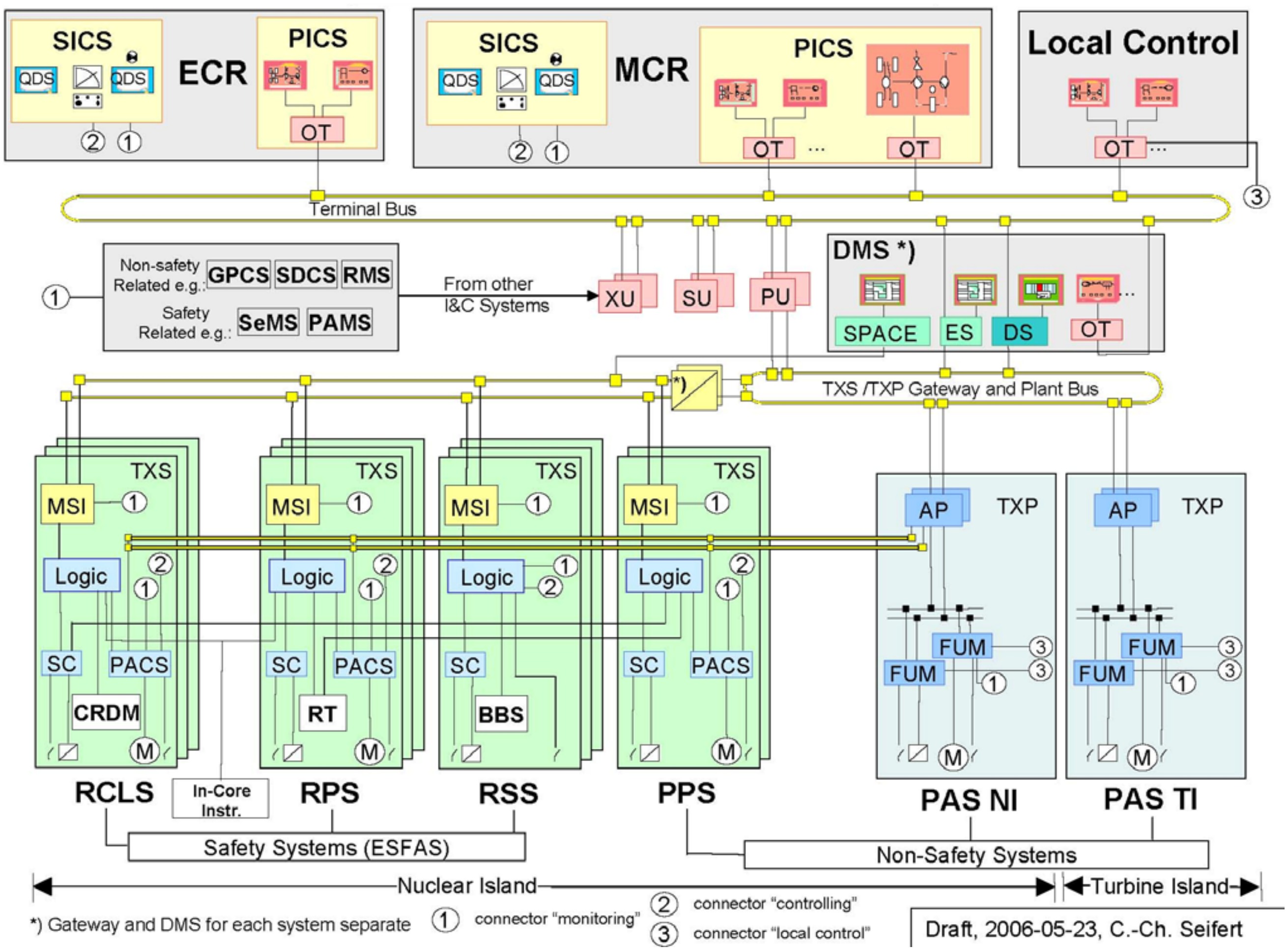


Figure 7-11: NGNP Preliminary I&C Architecture

7.10 Plant Protection System

The Plant Protection System (PPS) is a collection of safety and non-safety systems. It includes the following safety related systems:

- Reactor Protection System (RPS),
- Reactor Control & Limitation System (RCLS),
- Priority and Actuator Control System (PACS),
- Post-Accident Monitoring System (PAMS), and the
- Seismic Monitoring System (SeMS)

It also includes the following non-safety systems:

- Investment Protection System (IPS), and the

- Diagnostic & Maintenance System for Safety I&C (DMS)

7.10.1 Safety Related Systems

The Reactor Protection System (RPS) is provided to insure that the reactor is brought to a subcritical condition and that its heat generation is minimized in the event a significant failure within the plant occurs that challenges the ability to remove reactor heat. This is done to protect the health and safety of the public and the personnel operating the plant. The system also provides assurance of long-term subcriticality following such a plant failure. The RPS meets the requirements of IEE Std. 603, and its computers and software meet the requirements of IEE Std. 7-4.3.2. The RPS hardware is Class 1E quality.

The RPS is a digital safety system provided to detect plant failures that challenge reactor heat removal. It is physically and functionally separate and independent of the plant control systems and the IPS. The system is composed of three redundant channels. Commands from the redundant channels are combined in a two-out-of-three coincidence voting logic (to be confirmed). The RPS automatically performs reactor trip using two independent and diverse mechanisms:

- Reactor control rods
- Reactor Reserve Shutdown System (RRSS) (to be clarified if RRSS actuation is automated or operator actuated)

Both systems release neutron poison into the reactor core upon the loss of electrical power to the respective systems. These systems compliment the inherent negative coefficient of reactivity of the core that will automatically lower reactor power as core temperature increases. Table 7-2 shows the primary reactor trip parameters for the RPS.

Table 7-2: Plant Protection System Trip Actions

RPS Action	Trip Parameter	Measured Parameters
Trip Control Rods	High neutron flux to He mass flow	Neutron flux Coolant flow rate
	Low primary coolant pressure	Primary coolant pressure
	High primary coolant pressure	Primary coolant pressure
	High reactor outlet temperature	Primary coolant temperature at outlet
	High reactor inlet temperature	Primary coolant temperature at inlet
	Low cooling water flow	Cooling water flow
	Turbine trip	Turbine trip logic
Trip RRSS	High neutron flux to He mass flow	Neutron flux Coolant flow rate

The Reactor Control & Limitation System (RCLS) is provided to insure reliable reactor cooling and decay heat removal through:

- Core and reactor control
- Monitoring of Limiting Conditions of Operation (LCO) and automatic limitation functions for core and RCS (?) parameters

- Manual control and LCO functions for core and RCS

The RCLS initiated control and limitation (vis. CRDM) actions are of higher priority than plant control systems and, with the aid of the PACS, will override those signals.

The Priority and Actuator Control System (PACS) allows safety related systems to take control of safety related actuators and initiate plant shutdown contrary to possible signals from the plant control system. Priority and Actuator Control (PAC) modules are provided for all control rod actuators. These modules perform the priority selection, actuator control and monitoring functions for the safety classified actuators.

The Post-Accident Monitoring System (PAMS) includes sensors and processors to monitor, record and display parameters important to plant safety in the control room. They will permit the plant operating personnel to:

- Perform preplanned manual actions
- Verify that safety systems are performing their intended functions
- Provide information on plant status to allow appropriate operator decisions, provide warning of the release of radioactive materials or potential for occurrence of gross breach of barriers

The Seismic Monitoring System (SeMS) is a set of seismic instrumentation that provides information about the seismic history of the site for normal, upset and accident conditions. The information is used to analyze the seismic event (e.g., earthquake or induced vibrations) and helps making decisions on inspection and maintenance activities. In the event of a significant seismic event, it informs the operator of the event and its magnitude and characteristics (e.g., acceleration vectors, frequency and duration). This information will allow identification of components that may have been put through excessive stresses and require inspection.

7.10.2 Non-Safety Related Systems

The Investment Protection System (IPS) primary functions are to sense process variables and detect abnormal plant conditions and to actuate equipment to maintain plant parameters within acceptable limits to ensure plant component damage limits and/or technical specification limits are note exceeded. This protects the plant investment. The IPS is separate and independent of plant control systems. Major IPS functions are:

- Brayton engine over speed protection
- Steam turbine over speed protection
- Brayton engine trip
- Steam turbine trip (if present)
- Initiation of shutdown cooling system
- Cooling water isolation

The Diagnostic & Maintenance System for Safety I&C (DMS) provides diagnosis and troubleshooting for the systems in the safety-related initiation path and in the plant data processing system, allowing fault diagnostic, parameterization, periodic test and software loading. The DMS is connected to the terminal bus as well as to the plant bus and operates via gateway and the Monitoring and Service Interface System (MSI) on the safety-related I&C systems.

7.11 Plant Monitoring System

The plant monitoring system is composed of two independent systems, the Process Information and Control System (PICS) and the Safety Information and Control System (SICS). These operating and monitoring systems enable the operators to monitor plant conditions during normal and abnormal operations. Instrumentation employed for plant monitoring includes:

- Analytical instrumentation
- Environmental instrumentation
- Meteorological instrumentation
- Seismic instrumentation

PICS provides information and control means to operate and monitor the plant in all plant situations as long as PICS is operable. Information interchange and operations by PICS are possible via plant bus and gateways to the safety-related systems and via terminal bus to operator terminals.

On the reactor primary coolant circuit, this system will monitor plant parameters such as neutron flux around the perimeter of the reactor reflector, reactor coolant inlet and outlet temperatures and pressures, coolant flow rate from each of the four primary heat exchangers, reactor vessel and support structure temperatures, and control rod positions. On the secondary side, the system will monitor plant parameters such as secondary coolant flow rate and temperatures at key locations of the Brayton engine, including compressor inlet and outlet, turbine inlet and outlet and bypass line. Steam and condensate temperatures and pressures, condensate flow rate and Brayton engine and turbine rotation speeds will also be provided. Gas flow rates, pressures and temperatures to the hydrogen generation plant will also be shown. Secondary system process parameters such as equipment coolant temperatures and flows, and gas inventory system pressures, flows and temperatures will also be monitored. In addition to process parameters, the system will show the status of all important pieces of equipment in the plant including major pumps, valves and electrical switches.

In addition to monitoring key plant process variables and component status, the PICS will automatically monitor the chemical and radiological impurities in the primary and secondary coolant circuits during all modes of plant operation. To maintain safe plant operation and to limit allowed contamination levels, the system monitors the primary coolant that circulates through the reactor and the helium purification/regeneration train, as well as releases from the primary circulating inventory to the radioactive waste management system. The secondary coolant loop is also monitored to identify fission products that may leak from the primary system across the heat exchangers.

The SICS is the central location for alarm annunciations from the reactor protection system in the control room. It displays the conditions of the reactor protection system in normal operation, during anticipated operational occurrences and in the event of accidents. The system provides key plant process parameters, status indications from instrumentation strings and indicators. In addition to providing information to the operators, protection functions can be manually initiated from the SICS.

7.12 Electrical Systems

The Plant Electrical System and its associated subsystems support the process and balance of plant facility functions and services during all modes of plant operation, including start-up and testing, normal operation, shutdown and plant outages.

The Plant Electrical System includes the following subsystems:

- Plant Main AC Power Supply
- DC and Uninterruptible Power Supply
- Plant Standby Power Supply

Refer to the Key One Line Diagram shown in Figure 7-12 (Drawing No. SK E0001]) for the overall power distribution scheme. The conceptual depiction of electrical system layout is indicated on the Plot Plan contained in Section 9.

Hydrogen facilities are not included in this system, except for a provision of medium voltage underground cable and step down transformers (4.16 kV secondary) for power requirements of the hydrogen facilities is included.

On-site HV switchyard is connected to the offsite utility transmission system through a 4- mile long single circuit 138 kV overhead transmission line. A short single circuit 138 kV overhead line connects the common single step-up transformer to the on-site HV switchyard. The transmission lines provide sufficient capacity to export the generated power to the grid as well as to provide for the power requirements of the plant during startup, shutdown and outages.

The output of the Steam Turbine Generator (STG) is connected to the common step-up transformer through an overhead Iso-phase bus, to which its unit auxiliary transformer is connected. A generator circuit breaker in the circuit provides for fault isolation. The arrangement for the Combustion Turbine Generator (CTG) is similar. Its output is connected to the common step-up transformer.

Normal plant operation utilizes two (2) 4.16 kV (nominal) buses, each rated for 100% plant load, as the normal power source. Each 4.16 kV bus is in a single bus double breaker arrangement. The normal supply to a 4.16 kV bus is from its own unit auxiliary transformer. The alternate power source to the bus is provided from the other unit auxiliary transformer of equal capacity. Provision of the alternate power source allows power restoration, should the normal source be out of commission. The 4.16 kV buses supply power to 4 kV motor loads and the station service transformers.

The plant power distribution is provided at 480 volts via four (4) 480Vac unit substations (USS), two (2) on each bus. The equipment loads affiliated with the individual processes are aligned with the individual USS, wherever possible. This design approach serves to minimize the pieces of equipment to be tagged and locked out during partial plant maintenance outages.

The function of the plant DC electrical system is to supply 125 V DC power to the plant normal DC loads and equipment, and DC control power for the operation of AC and DC systems. It consists of two batteries, two DC switchboards and DC panel boards as required. Each battery is equipped with two battery chargers. The chargers are powered from separate AC buses.

The function of Uninterruptible Power Supply (UPS) system is to support the critical loads that require an uninterruptible power source for reliability of operations. Two separate UPS buses are provided.

The standby power supply subsystem is comprised of an on-site standby power source available within 60 seconds, to support shutdown process operations and achieve orderly shutdown in the event of total off-site power source loss. The design of the power system is such that the standby power source can provide power to both the 4.16 kV buses and their associated 480 V buses.

The plant electrical equipment consisting of medium and low voltage switchgears, motor control centers, ac distribution panels, DC system and UPS system is located in Electrical Building. Additionally there is a provision for locating motor control centers in various buildings to supply power to the local loads.

The Standby Power System Building houses the standby generator.

7.13 Component Handling System

The primary function of the Component Handling System is to provide a means of handling and storing large components during disassembly of key nuclear heat source components. This is not an integrated system, rather it is a series of fixtures and devices, each designed specifically to handle one reactor component.

As the Conceptual Design process unfolds, these handling devices will need to be designed based on the final configuration of the components involved and the overall layout of the reactor building. Laydown/storage areas will also need to be defined, as well as the required support from various building cranes.

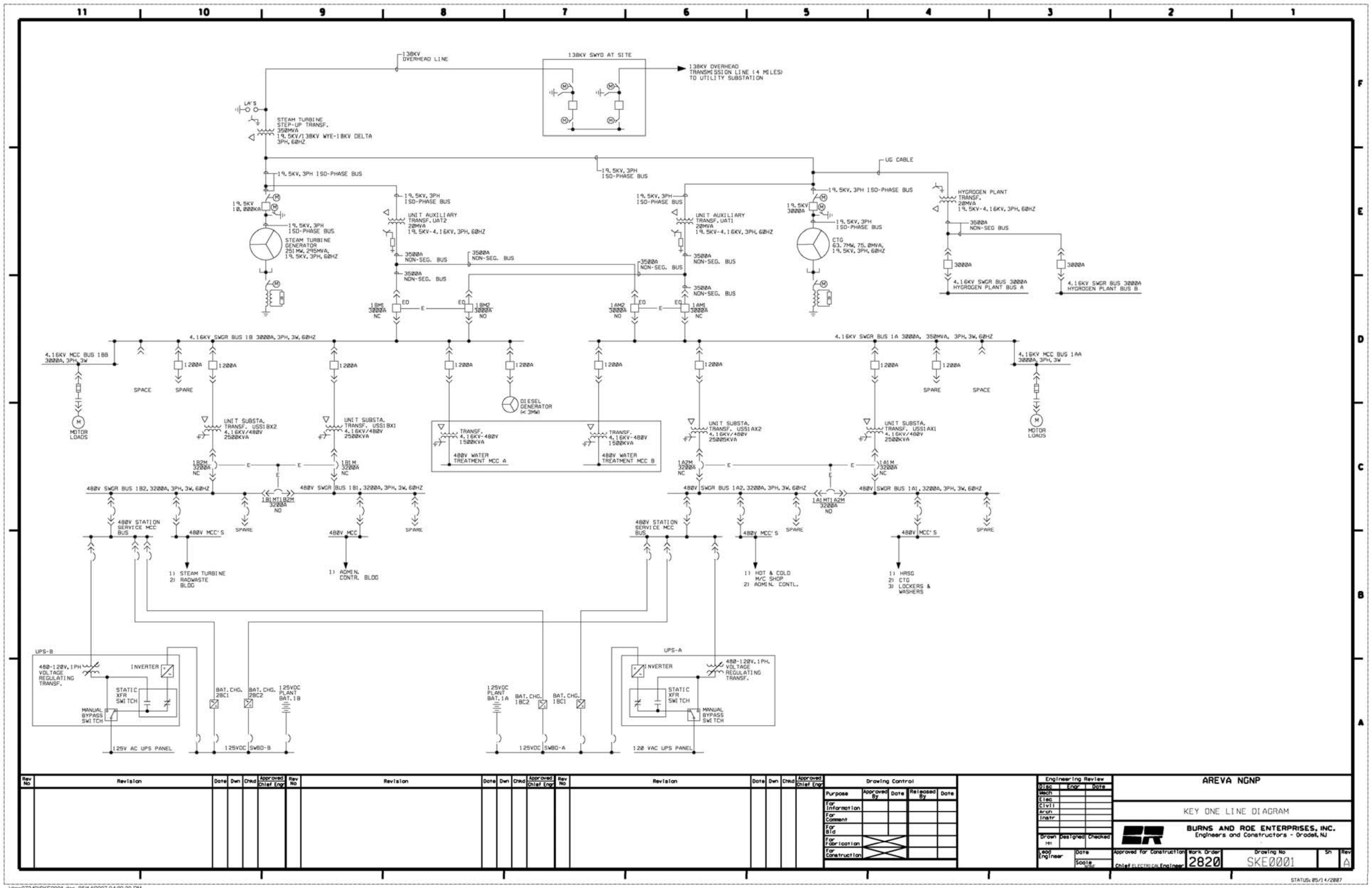


Figure 7-12: Key One Line Diagram

8.0 POWER CONVERSION SYSTEM

8.1 PCS Development Overview

As specified in the AREVA scope of work for the NGNP preconceptual design, the NGNP Power Conversion System (PCS) is based on the adaptation of the ANTARES concept to NGNP design conditions.

The ANTARES concept provides the framework for the basic PCS arrangement for the NGNP; however, based on the results of the special studies for power level and primary and secondary parameters and arrangement, a sufficiently different and unique set of NGNP design conditions evolved requiring detailed assessment and evaluation.

Furthermore, two PCS concepts for NGNP are proposed. The first concept is the first-of-a-kind (FOAK) concept that is proposed for the NGNP. The second PCS concept is the Nth-of-a-kind (NOAK) concept that is proposed for a future commercial plant dedicated to electrical generation. The NOAK PCS concept is somewhat more complex than the FOAK concept by virtue of the addition of steam reheating, the addition of an intermediate turbine stage on the high pressure turbine and more efficient gas turbine features.

It should be noted that the NOAK PCS concept presented in this section is different that NOAK plant concept discussed in Section 16, Economics. The NGNP NOAK plant sole mission is to provide process heat to produce hydrogen. The NOAK PCS presented here is to demonstrate the concept's capability and potential in the commercial electricity market.

8.2 ANTARES HTR Reference Concept

The reference ANTARES HTR concept shown Figure 8-1 is a modular HTR coupled to a combined cycle gas turbine (CCGT) generating system. Heat from the reactor or primary circuit is transferred through an Intermediate Heat Exchanger (IHX)to a secondary circuit that is a closed-loop Brayton cycle which generates a part of the plant's electrical output. A nitrogen-helium gas mixture is used in the secondary circuit in order to use conventional gas turbine technology. Energy in the gas turbine exhaust is then transferred through a heat recovery steam generator to a conventional sub-critical steam Rankine cycle that comprises the tertiary or "bottoming" circuit which converts the remaining useful energy into electrical power.

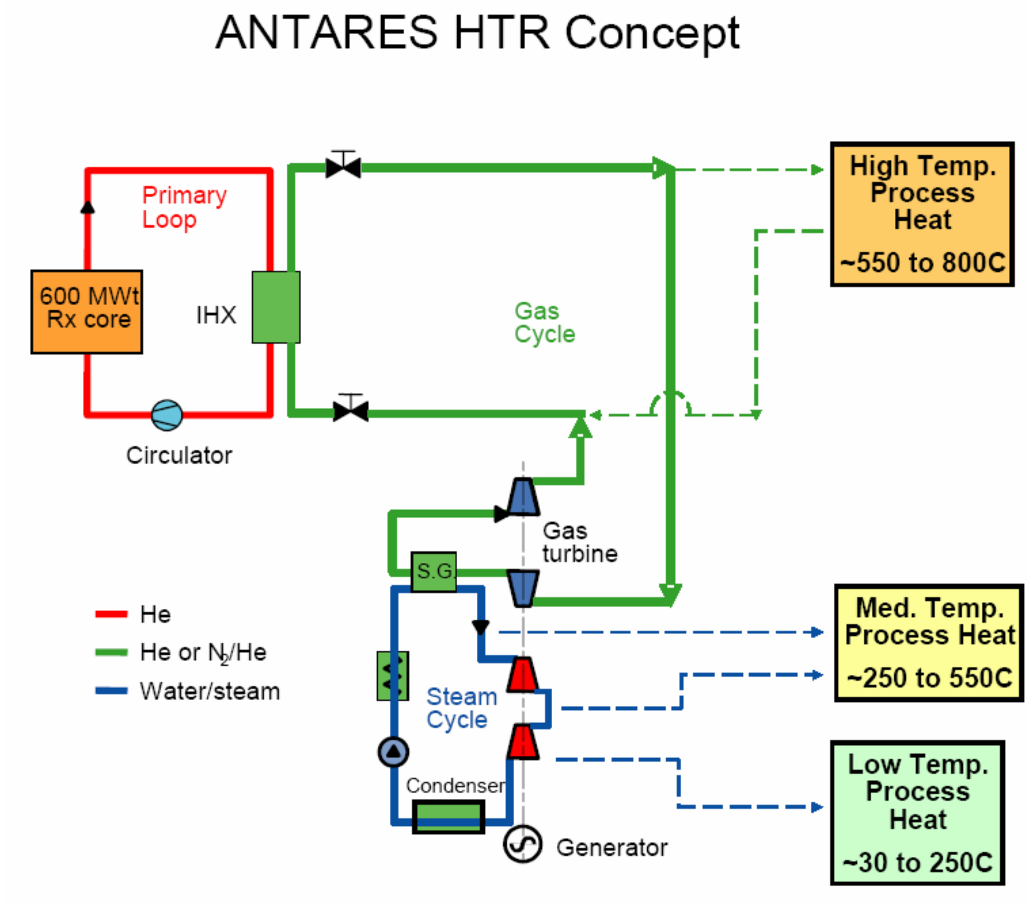


Figure 8-1: ANTARES HTR Concept

The power conversion system portion of the above schematic provides the starting point for development of the reference NGNP PCS design adaptation.

8.3 Reference NGNP Power Conversion System

8.3.1 NGNP PCS Design Condition

The PCS design conditions are taken from the NGNP with Hydrogen Production Design Baseline Document [5] . The design conditions specified in the baseline document are based on the results of the power level special study and the primary-secondary parameter and configuration study contained in Appendix B1 and B2, respectively.

Major design conditions of the PCS are as follows:

1. Compressor Outlet Temperature/IHX Inlet Temperature: 450ºC
2. IHX Outlet Temperature/Gas Turbine Inlet Temperature: 850ºC
3. Gas Turbine Outlet Temperature/HRSG Inlet temperature: 600ºC
4. Gas composition: N2: 80 wt.% He: 20 wt.%
5. Number of Loops : Single
6. Shaft Configurations: One multiple shaft Gas Compressor/Turbine/Generator

One multiple shaft HP/LP Steam Turbine/Generator

7. Gross Power Target: 280 MWe

The initial PCS configuration considered for NGNP is shown schematically in Figure 8-2 below. In this configuration, the heat generated in the reactor is transferred to the secondary coolant via three intermediate heat exchangers (IHX2). The maximum power capability of tubular IHXs is such that 3-IHX2s will be required for NGNP.

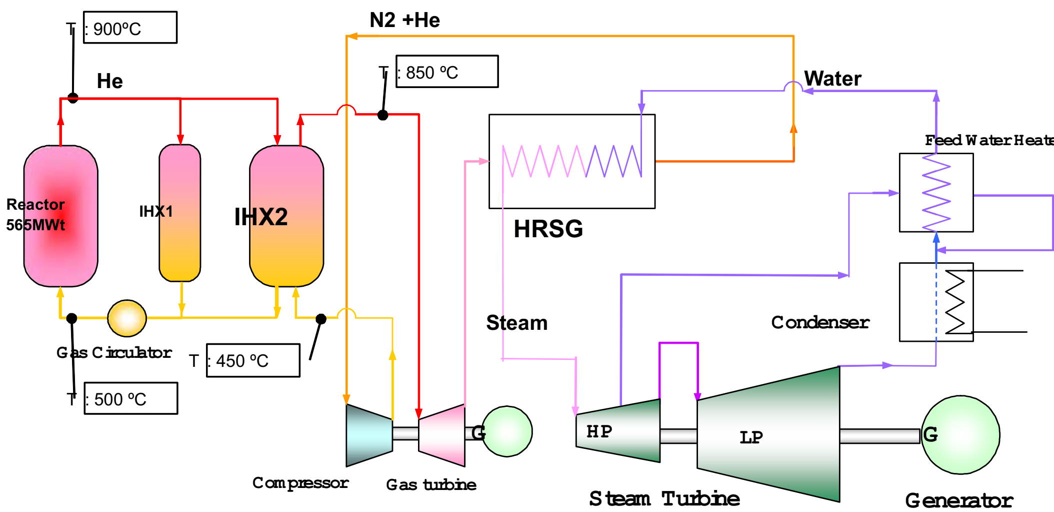


Figure 8-2: PCS Configuration w/o Reheat

The heated secondary coolant, which was compressed by the compressor prior to entering the the IHX, is expanded in the gas turbine which in turn drives both the generator to produce electricity and the gas compressor. The gas turbine exhaust contains significant residual heat most of which is transferred to tertiary water/steam in the Heat Recovery Steam Generator (HRSG). The steam generated in the HRSG drives the HP and LP steam turbine to drive its generator to produce electricity.

The above NGNP PCS configuration allows the demonstration of separate turbine generator sets and permits each power unit to be uniquely optimized for its given conditions. Furthermore, in a multiple-module setting, it is feasible to feed a common steam-turbine unit from two or more reactor modules. The NGNP PCS configuration allows the demonstration of the key control features that would be required by such an arrangement.

8.3.2 NGNP PCS Heat and Mass Balance (FOAK)

The heat and mass balance for the NGNP PCS (i.e., the FOAK prototype) is shown in Figure 8-3 below. An intermediate pressure (IP) turbine supplied with reheated HP turbine exhaust has been added for improved performance. The gross electrical outlet of the PCS is 279 MWe of which the gas-turbine generator unit produces 53 MWe and the steam-turbine generator unit produces 226 MWe. Based upon a 580 MWt thermal input, the resulting gross efficiency is 48.1%. Overall plant efficiency is less because of in-plant electrical loads, estimated at approximately 25 MWe; hence, the overall plant efficiency is approximately 45% (i.e., (279-25)/565).

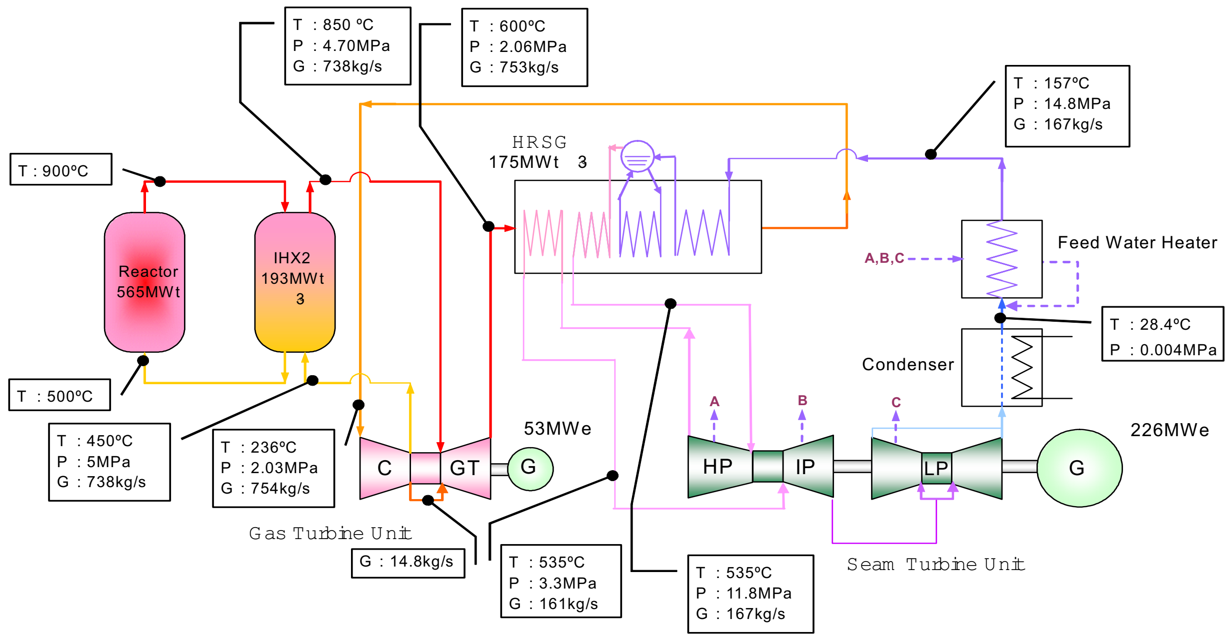


Figure 8-3: Heat and Mass Balance for FOAK NGNP

8.3.3 NOAK PCS Heat and Mass Balance

The overall power output of the FOAK NGNP PCS presented in the previous section was slightly under the target power of 280 MWe. For the NOAK commercial plant, improvements in turbo-machinery efficiency above the FOAK plant are expected and will result in improved overall performance.

The heat and mass balance for the NOAK commercial plant configuration is shown below in Figure 8-4.

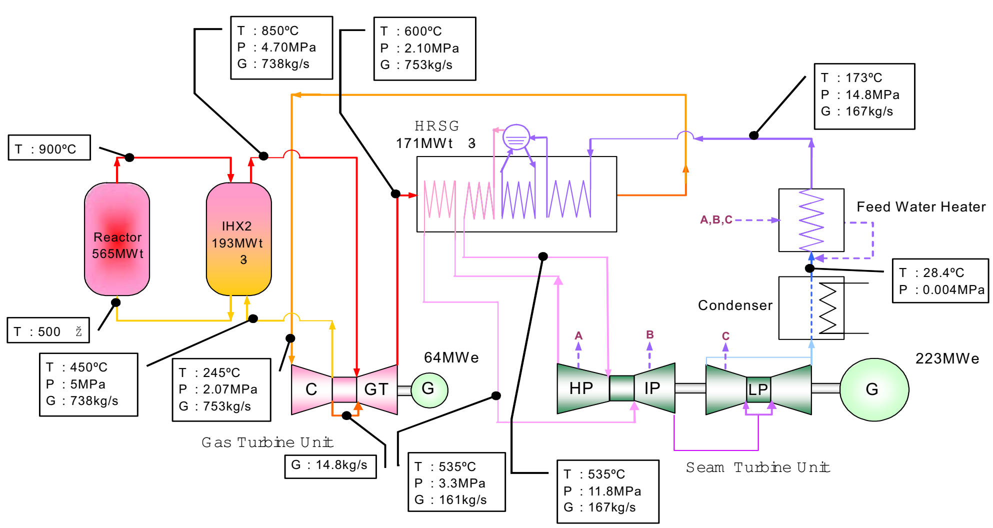


Figure 8-4: Heat and Mass Balance for NOAK Commercial Plant (w/Reheat)

8.4 Brayton Cycle Description

The major components of the Brayton cycle consist of the the gas turbine unit (gas turbine, compressor and auxiliaries) and the interconnecting ductwork (inner insulated hot gas duct).

8.4.1 Gas Turbine Unit

8.4.1.1 General Description

The main function of the gas turbine unit (gas turbine, compressor, and generator set) is to convert the thermal energy contained in secondary circuit gas exiting the IHX into electrical power. This gas, at a pressure of 4.7MPa and temperature 850 ºC, is expanded through the gas turbine and exhausted with a corresponding decrease in pressure and temperature. The shaft power generated by the gas turbine drives both the gas compressor and electrical generator. The heat content of the turbine exhaust gas is significant and much of it is transferred to the tertiary steam cycle via the HRSG. Upon exit from the HRSG, the cooled secondary circuit gas is conducted to the compressor inlet.

The main function of compressor is to raise the gas pressure and temperature to IHX inlet conditions (5MPa, 450ºC). Figure 8-5 below shows the gas turbine unit layout. The layout is horizontal which is similar to conventional gas turbine units. The gas turbine and compressor are each covered by their own individual casings. The rotors of gas turbine, compressor and generator are connected by couplings. The gas turbine and compressor rotational speed is 3600 rpm which is optimally matched to the rotational speed of the electrical generator and permits direct coupling.

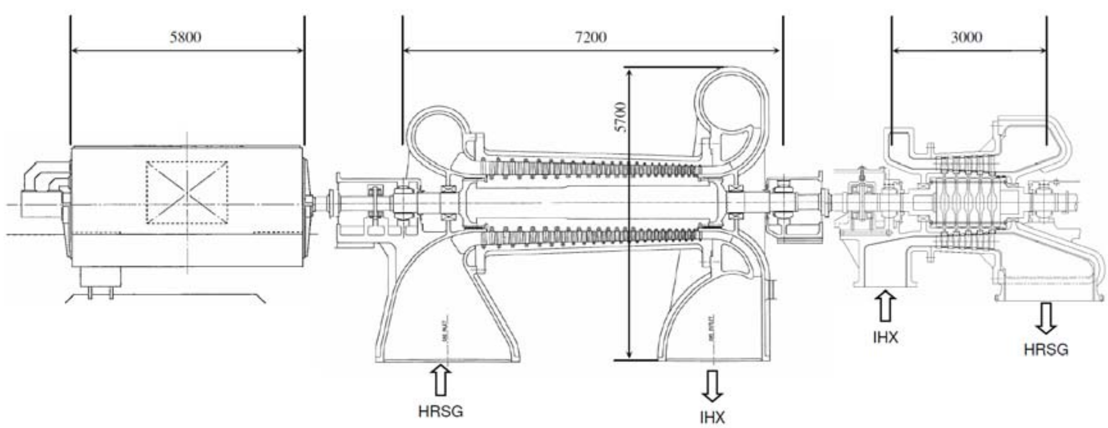


Figure 8-5: Gas Turbine Unit Layout

An oil-bearing system is used to support the common shaft of the gas turbine, compressor, and generator. A dry gas seal system is use to seal the shaft penetrations in the gas turbine casing and compressor casing.

The generator is a standard horizontal, air cooled design. The variable-voltage variable frequency (VVVF) inverter is provided in order to use the generator as start up motor for the gas turbine/compressor unit.

8.4.1.2 Gas Turbine

The design specification of gas turbine is shown in Table 8-1 below. The outline drawing is shown in Figure 8-6. The rotor and casing are cooled by a feed from the gas compressor extracted at an appropriate temperature level. The rotor is supported on oil bearings. The shaft penetrations of the casing are installed with a Dry Gas Seal system.

Table 8-1: Gas Turbine Design Data

Parameter	Unit	FOAK (NGNP Prototype)	NOAK
Fluid	-	He (20%) + N_2 (80%)	
Inlet/Outlet Temperature	ºC	850/600	
Inlet/Outlet Pressure	MPa	4.70/2.06	4.70/2.10
Flow Rate	kg/s	738	
Rotational Speed	rpm	3600	
Rotor Diameter	mm	1500	
Chip Diameter of Final Stage	mm	1795	
No. of Stages	-	5	
Bearing Center Line Span	mm	3000	

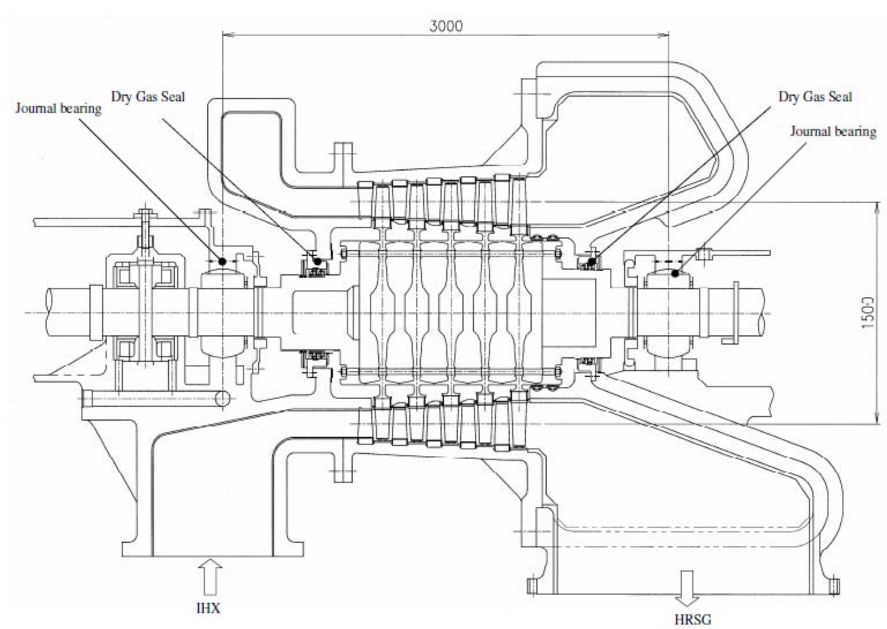


Figure 8-6: Gas Turbine Outline

8.4.1.3 Gas Compressor

The design specification of gas compressor is shown in Table 8-2. The outline drawing is shown in Figure 8-7 The rotor is supported on oil bearings. The shaft penetrations in the casing are sealed using a Dry Gas Seal system.

Table 8-2: Gas Compressor Design Data

Parameter	Unit	FOAK (NGNP Prototype)	NOAK
Fluid	-	He (20%) + N_2 (80%)	
Inlet/Outlet Temperature	ºC	236/450	245/450
Inlet/Outlet Pressure	MPa	2.03/5.003	2.07/5.003
Flow Rate	kg/s	753	
Rotational Speed	rpm	3600	
Rotor Diameter	mm	1140	
First Stage Blade Diameter	mm	1630	
No. of Stages	-	23	
Bearing Center Line Span	mm	7200	

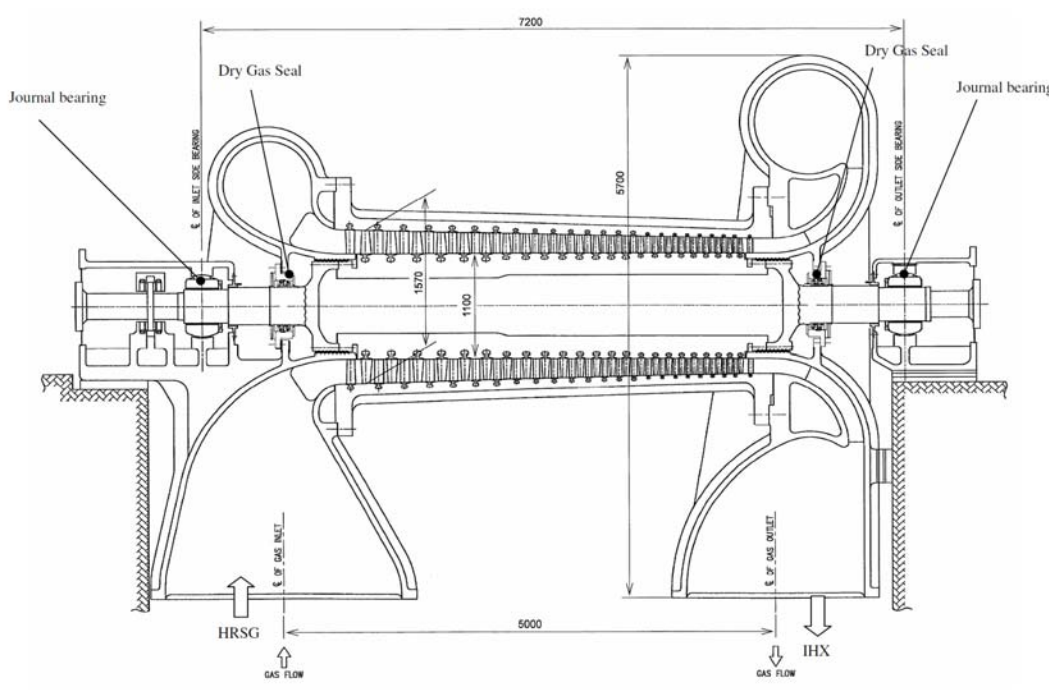


Figure 8-7: Gas Compressor Outline

8.4.1.4 Secondary Gas Control System

Secondary gas inventory is controlled by the secondary gas inventory control system which provides the following functions:

1. Pressure and inventory control

2. Gas supply and storage

The secondary gas inventory system consists of the following major equipment:

1. Gas storage tanks

2. Gas supply tanks

3. Gas compressors

Table 8-3 shows major system data. The plot plan of the secondary gas inventory control system is shown in Figure 8-8.

Table 8-3: Secondary Gas Inventory Control System Design Data

Item	Unit	Design Data
No. of Systems	Unit	1
Max. compressor flow rate	M^3/hr x No. of Units	75 x 3
Volume and No. of Storage Tanks	M^3 x No. of Units	12.6 x 26
Volume and No. of Supply Tank	M^3 x No. of Units	12.6 x 1

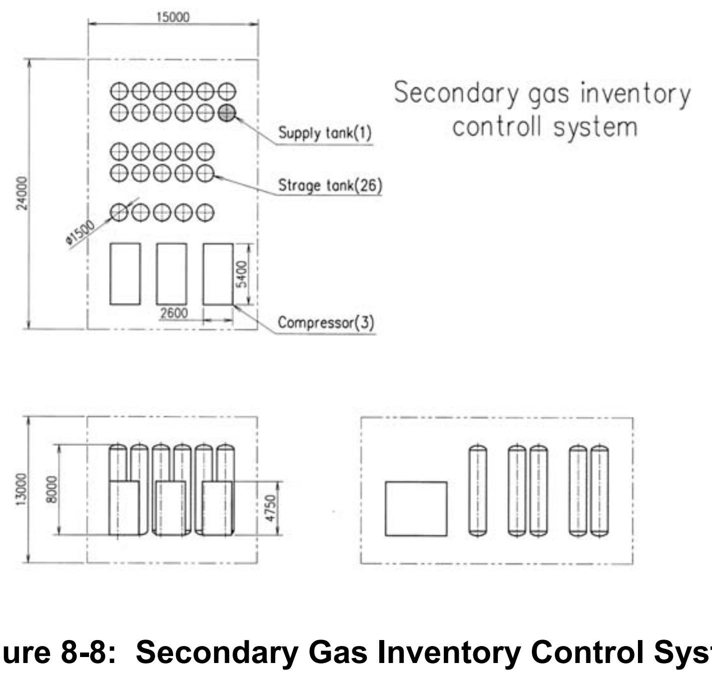


Figure 8-8: Secondary Gas Inventory Control System

8.4.2 Secondary Piping

Pipes are used to transport the hot gas (850ºC) from IHX outlet to gas turbine inlet and from gas turbine outlet to HRSG inlet. Pipes conveying high temperature gas is insulated on the inner surface to keep its operating temperature low so that carbon steal piping materials can be used.

Major design data for the ducting is given in Table 8-4 and a schematic layout is shown on Figure 8-9.

Table 8-4: Major Secondary Piping Design Data

Line	O.D (mm)	Thickness (mm)	Material	Length (m)	Fluid	Velocity (m/s)	Design Condition		Operating Condition		Remarks
							Temperature (ºC)	Pressure (MPa)	Temperature (ºC)	Pressure (MPa)	
IHX ? (1)	1295	46	C.S.	TBD	He+N2	49	350	5.8*	850	4.7	Inner insulation
(1) ? GT	2032	58	C.S.	TBD	He+N2	50	350	5.8*	850	4.7	Inner insulation
GT ? (2)	2489	71	C.S.	TBD	He+N2	58	350	5*	600	2.06	Inner insulation
(2) ? HRSG	1549	46	C.S.	TBD	He+N2	56	350	5*	600	2.06	Inner insulation
HRSG ? (3)	991	30	C.S.	TBD	He+N2	61	350	5*	222	2.03	
(3) ? Comp.	1676	45	C.S.	TBD	He+N2	63	350	5*	222	2.03	
Comp. ? (4)	1372	40	A.S	TBD	He+N2	54	480*	5.8*	450	5.0	
(4) ? IHX	813	25	A.S.	TBD	He+N2	52	480*	5.8*	450	5.0	

*: Tentative value IHX: Intermediate Heat Exchanger GT: Gas Turbine HRSG: Heat Recovery Steam Generator Comp.: compressor
C.S.: Carbon Steel A.S.: Alloy Steel

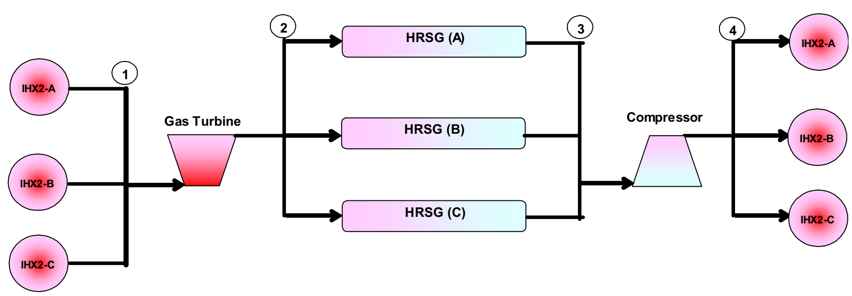


Figure 8-9: Schematic – Major PCS Ducting

8.5 Steam (Rankine) Cycle Description

The major components of the tertiary circuit are the heat recovery steam generator, the HP/IP/LP turbine units and the generator, and the condensate system.

8.5.1 Heat Recovery Steam Generator (HRSG)

The heat energy in the gas turbine exhaust is transferred to the tertiary circuit (i.e., steam cycle) via the HRSG. The HRSG operates between 600ºC and 236ºC in the case of the FOAK NGNP PCS and between 600ºC and 245ºC in the NOAK configuration.

The HRSG is very large heat exchanger contained within three large pressure vessels. Because the heat transfer process is limited by the heat transfer coefficient of the secondary gas which is much smaller than heat transfer coefficients experienced on the steam/water side of the HRSG, a very large heat transfer surface area is required.

Considering transportation package size limitations, the outer diameter of the HRSG pressure vessel is limited to 4.6m. Hence, this limit translates into three separate HRSG modules being required.

The design specification of the HRSG is shown in Table 8-5. The outline drawing is shown in Fig. 8-10. The HRSG pressure vessels are in a horizontal layout to minimize the supporting structure, and the insulation is installed on the inner surface to protect it from over heating by high temperature gas turbine exhaust gas.

Table 8-5: HRSG Module Design Data

Item	Unit	FOAK (NGNP Prototype)	NOAK
Type	Module	Horizontal pressure vessel	Horizontal pressure vessel
No. of Vessels	-	3	3
Fluid (shell side/tube side)	-	He + N_2/Water-Steam	He + N_2/Water-Steam
Shell side temperature (inlet/outlet)	ºC	600/236.4	600/244.5
Tube side temperature (inlet/outlet)	ºC	157.0/535	173.1/535
Inlet Pressure (shell side/tube side)	MPa (abs)	2.064/14.83	2.095/14.83
Flow rate, shell side	Kg/s	251.1	251.1
Flow rate, tube side HH SG SH	Kg/s	55.6	55.6
Flow rate tube side RH	Kg/s	53.8	53.8
Heat duty/vessel	MWth	175.1	171.3
Heating area/vessel	M^2	2458	2450
Tube bundle size, reheat zone	W x H x L (MxMxM)	2.89 x 2.89 x 2.29	2.89 x 2.89 x 2.29
Tube bundle size, superheating zone	W x H x L (MxMxM)	2.89 x 2.89 x 4.37	2.89 x 2.89 x 4.37
Tube bundle size, boiling zone	W x H x L (MxMxM)	2.89 x 2.89 x 7.77	2.89 x 2.89 x 7.77
Tube bundle size, subcooled zone	W x H x L (MxMxM)	2.89 x 2.89 x 10.41	2.89 x 2.89 x 10.41

8.5.2 Steam Turbine

Using steam from the HRSG, the steam turbine unit (steam turbine and generator set) generates electrical power. Superheated high pressure steam (11.8MPa, 535 ºC) from the HRSG goes into high pressure (HP) turbine and then, upon exhaust, is conducted to the HRSG reheating zone. The reheat steam from the HRSG goes to the intermediate pressure (IP) turbine. The exhaust steam from IP turbine is conducted to the low pressure (LP) turbine. The exhaust steam from the LP turbine is flows directly to the steam condenser.

The extracted steam from HP, IP and LP turbines are used for heating feed water. Figure 8-11 shows the steam turbine unit layout. The layout of steam turbine unit is horizontal similar to a conventional steam turbine unit. The HP and IP turbines are within the same casing, and the LP turbine is covered by its own casing. The steam turbine and generator rotors are connected by couplings. The steam turbines operate at 3600rpm in order to synchronize with the generator speed.

An oil bearing system is use to support and lubricate the shaft of the steam turbines and generator. The design specifications of the steam turbine are shown inTable 8-6

8.5.3 Condensate System

The condensate system condenses the steam from the LP turbine exhaust and supplies condensate to the feed water heaters. The condensate system includes the steam condenser, condensate water pump, vacuum pump unit, and other supporting sub-systems Figure 8-12 shows the steam condenser outline.

Table 8-6: Steam Turbine Design Data

Item	Unit	FOAK (NGNP Prototype)	NOAK
Type	-	Tandem compound reheat and condensing; two casing, double exhaust flow type	
Rated Output	MWe	226	223
HP Steam Pressure	MPa, abs	11.8	
HP Steam Temperature	ºC	535	
Reheat Steam Pressure	MPa, abs	3.3	
Exhaust Outlet Pressure	MPa, abs	0.004	
HP Steam Flowrate	kg/s	167	
Reheat Steam Flowrate	kg/s	161	
Rotational Speed	rpm	3600	

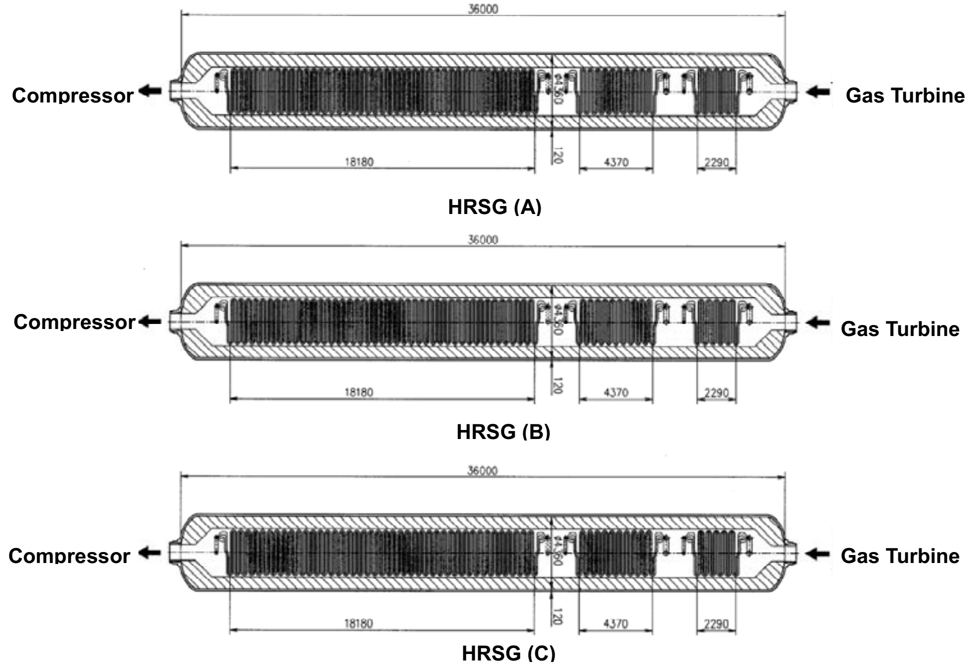


Figure 8-10: HRSG Module Layout

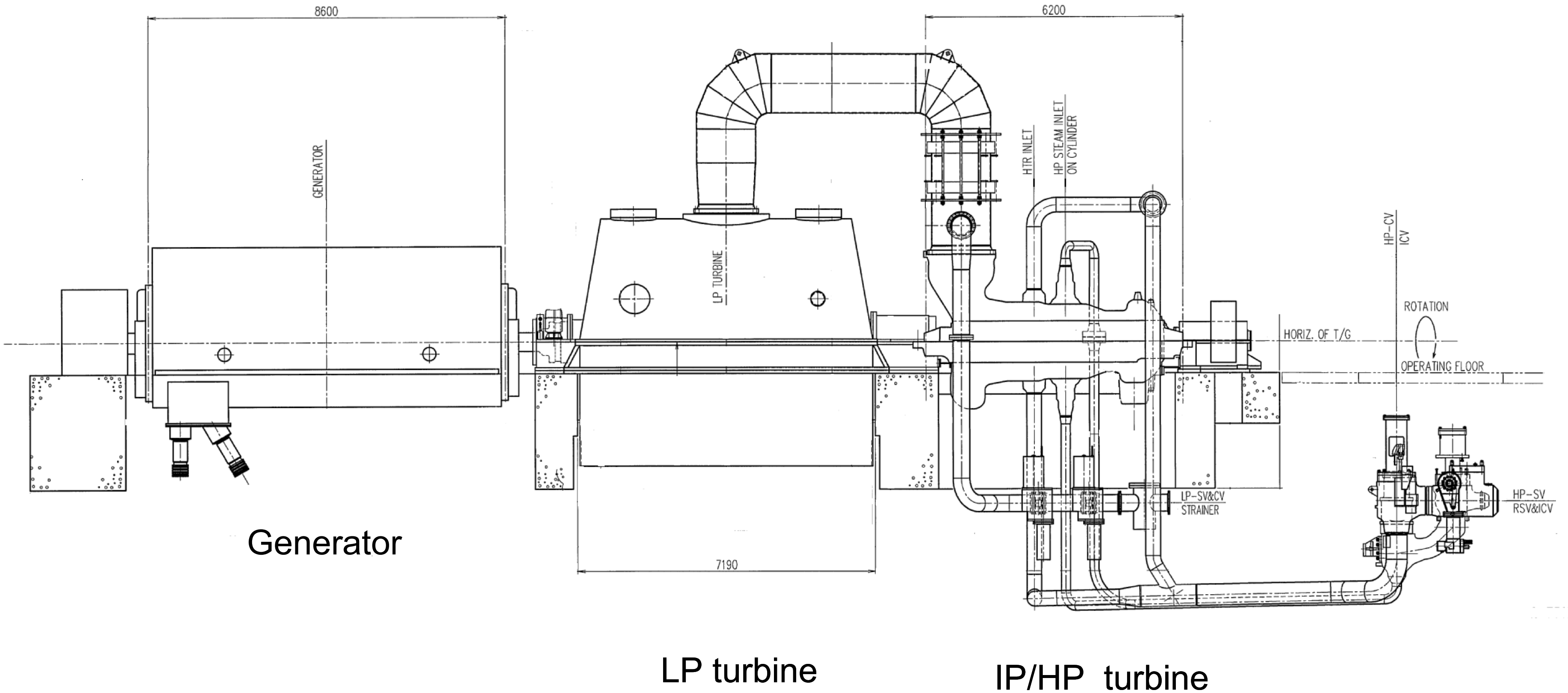


Figure 8-11: Steam Turbine/Generator Unit Outline

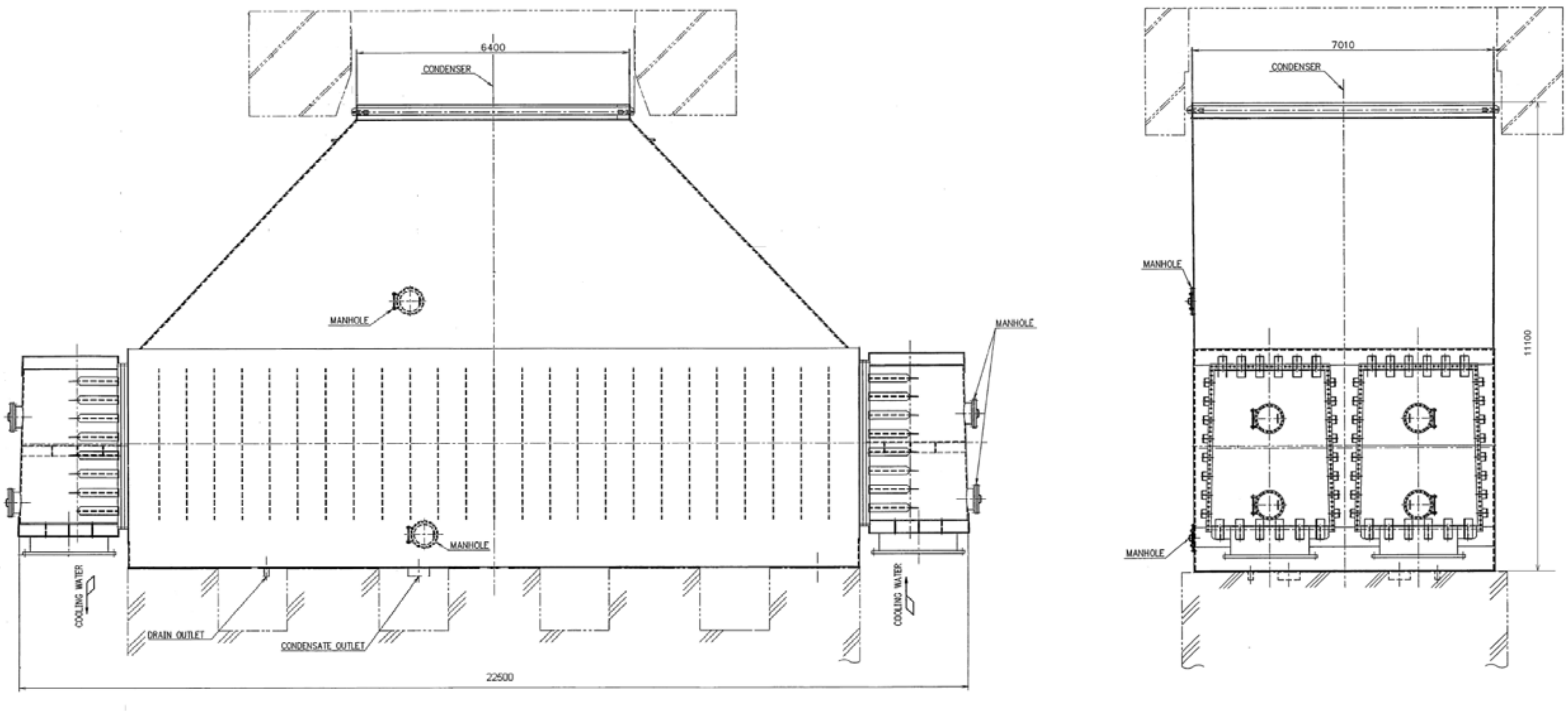


Figure 8-12: Steam Condenser Outline

8.6 PCS system performance

8.6.1 Objectives

PCS performance (Efficiency and Power Generation) for FOAK(NGNP Prototype Plant) and NOAK plant are prepared based on the results of design adaptation and by reflecting the results of major components design.

8.6.2 Conditions

8.6.2.1 Secondary Gas System

As described in Section 8.3.2 and 8.3.3, the PCS heat and mass balance for FOAK (NGNP prototype plant) and NOAK are calculated in Figure 8-3 and Figure 8-4. The difference of heat and mass balance between NOAK and FOAK is come from the difference of efficiency for gas turbine and compressor. The gas turbine unit efficiency of FOAK is assumed to be lower than that of the NOAK plant because the gas turbine and compressor for FOAK are the prototype products, therefore, it is reasonable to assume a more conservative value of the efficiency for the FOAK (NGNP prototype).

8.6.2.2 Steam Turbine System

The system flow sheet of steam turbine system is shown in Figure 8-4 also. The selected re-heat cycle was estimated as an alternative cycle in order to achieve higher efficiency than a non-reheat cycle and to satisfy with the limit of moisture ratio.

Figure 8-13 and Figure 8-14 show the temperature distribution in the HRSG for the FOAK and NOAK HRSGs. The 5 ºC pinch point of the HRSG between the secondary gas and steam/water was chosen to obtain as high a feed water temperature as possible to maximize the efficiency of the steam turbine system. On the other hand, the lower pinch point results in a larger HRSG. Additional optimization of this pinch point relative to cycle efficiency and economics is recommended for the next design stage.

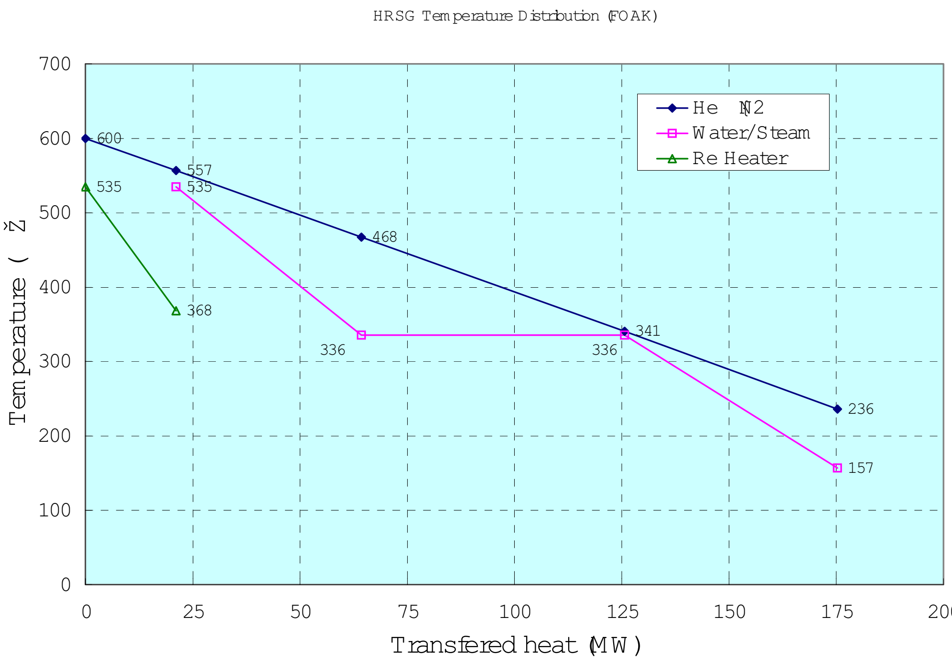


Figure 8-13: HRSG Temperature Distribution for the FOAK (NGNP Prototype)

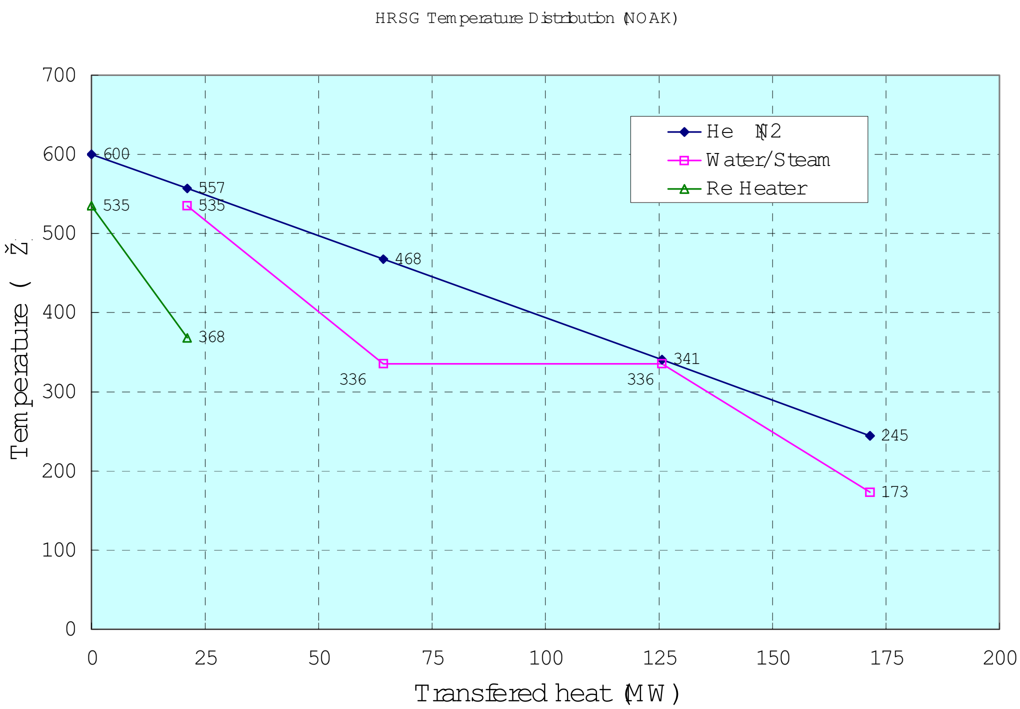


Figure 8-14: HRSG Temperature Distribution for the NOAK Commercial Plant

8.6.3 Performance

8.6.3.1 Power Generation

Gas turbine/generator unit:

- 53 MWe for FOAK (NGNP Prototype Plant)
- 64 MWe for NOAK (Commercial electric plant)

Steam turbine/generator unit:

- 226 MWe for FOAK (NGNP Prototype Plant)
- 223 MWe for NOAK (Commercial Electric Plant)

Total Power Generation:

- 279 MWe (53MWe+ 226MWe) for FOAK (NGNP Prototype Plant)
- 287 MWe (64MWe+223MWe) for NOAK (Commercial Electric Plant)

Note: steam turbine efficiency is depend on the cooling water temperature of condenser

8.6.3.2 Gross PCS Efficiency

The gross efficiency of the PCS is as shown below. The overall plant efficiency will be somewhat less after accounting for in-plant electrical loads (see Section 3.5.3).

- 48.2% for FOAK (NGNP Prototype PCS gross efficiency)
- 49.6% for NOAK (NOAK Commercial Electric Plant PCS gross efficiency)

Note: Gross PCS efficiency= power generation/heat duty of IHX-2

8.7 PCS Feasibility Issues

8.7.1 Objectives

The PCS is based on the ANTARES design concept and consists of the following major equipment:

1. Gas turbine, compressor and generator
2. Heat Recovery Steam Generator (HRSG)
3. Steam turbine power generation system
4. Gas inventory Control system
5. Piping (Hot gas duct)

The Critical components (which have critical R&D Items) and the technical issues are clarified in this work.

8.7.2 Evaluation Basis

This study is based on the following:

1. The ANTARES design concept
2. The results of the PCS design adaptation
3. Current technology
4. MHI experience
5. Published papers

8.7.3 Critical Components and Technical Issues

The following table summarizes reviews the key components of the PCS with respect to technical issues:

Table 8-7: Critical Component Technical Issues

Item	Technical Issue
Gas turbine	No Critical Issues
Gas compressor	Compressor blade designs are not verified for the FOAK plant.
Generator (i.e., driven by gas turbine)	No critical issues
Heat Recovery Steam Generator (HRSG)	No critical issues
Steam Turbine Generator	No critical issues
Gas Inventory Control System	No critical issues
Piping (i.e., hot duct)	No critical issues
Materials subject to high temperature	Nitriding affects materials exposed to high temperatures and nitrogen-rich environment.

Gas compressor blade performance tests will be necessary to confirm and optimize blade design in order to achieve higher performance for the NOAK commercial electric plant. For the FOAK NGNP prototype, a conservative blade performance was assumed in order to compensate for the potential performance risk.

The impact nitriding needs to be investigated by testing the candidate high temperature materials. The nitriding risk may be limited only to high temperature materials for IHX. Countermeasures may be sufficient to provide an optimized thickness for PCS materials subjected to nitriding except for IHX. At the present time, there is little quantitative information available describing the pertinent nitriding phenomena affecting materials at high temperature. Therefore, the severity impact of nitriding must be estimated for all critical PCS critical components.

9.0 PLANT LAYOUT AND STRUCTURES

NOTE: Plant Layout Drawings are included at the end of this section for reference.

9.1 Plant Layout

9.1.1 Site Data

The project site data is obtained from the results of the investigations performed at the NPR site and presented in References [6] and [7].

The proposed site for the construction of the new facility is located at the Idaho National Laboratory Site (INL). The INL Site is situated at the northern edge of the Eastern Snake River Plain adjacent to the southern foothills of the Lemhi and Lost River Ranges. The Eastern Snake River Plain is a 60 mile wide by 200 mile long basin. The site elevation is about 5,000 feet above the sea level. The ground surface may be classified as rolling to broken, with sagebrush vegetation. Depth from the ground surface to the Snake River Plain aquifer water table at the reference site is approximately 475 feet.

9.1.2 Geotechnical Data

Data obtained from the drilling program performed at the reference site included borehole logs, downhole video logs, seismic velocity measurements and preliminary field and laboratory testing results performed on selected samples of rock core and soil interbeds. Based on evaluation of collected data, preliminary values of static and dynamic rock and soil interbed properties were determined and an excavation concept developed.

9.1.2.1 Subgrade Evaluation

The geotechnical data indicates the strata composed of basalt of variable thickness alternating with thin layers of sediment. It has been concluded that the general excavation for the Reactor Service and Reactor Auxiliary Buildings, will only encounter minor sediment interbeds less than 0.5 meter thick. However, below first 30 meters, excavation for the Reactor Building shaft may encounter several major layers of sediment ranging from 1 to 10 meters in thickness. These layers primarily consist of sandy silt with cobbles of broken rock, and will require that a lateral support be provided to stabilize the shaft excavation. No weathered rock is associated with the soils at the site.

Results of preliminary uniaxial compressive strength tests performed on the intact rock specimens ranged from 20 to 110 MPa. Compression wave velocities for basalt ranged from 1100 to 4020 m/sec, and for the interbeds-from 450 to 1100 m/sec.

Results of laboratory tests on samples of the major soil interbed at a depth of about 30 meters indicate specific gravity of about 2.79, moisture content ranges from 16 to 37%; angle of internal friction ranges from 28 to 44 degrees; dry density ranges from 1500 to 1750 kg/m3 for undisturbed samples, and from 1200 to 1400 kg/m3 for disturbed samples.

9.1.3 Excavation Procedures

Excavation Procedures for the proposed site will vary depending on the excavation depth and type/size of the foundations.

The general excavation for the rectangular portion of the Reactor Building as well as Reactor Service and Reactor Auxiliary Building will be accomplished by blasting to a depth of about 20 meters in approximately 7 m benches over an area of about 60 by 100 meters. Access ramps will be provided for hauling the blasted rock.

The excavation for the cylindrical portion of the Reactor Building will be approximately 40 meters in diameter and extend about 50 meters below the bottom of general excavation. A drilling and blasting techniques including pre-splitting techniques will be utilized.

Both types of excavation will require ground support to maintain excavation boundaries and ensure safety. Specific support requirements will vary depending on local conditions.

Rock surface stabilization and lateral excavation support of the sediment interbeds will be required. To stabilize rock surfaces, anchored or expandable rock bolts will be utilized. Support of the interbeds may be accomplished by utilizing tangent or soldier piles, replacing the soil at the excavation boundary with a cement grout or other appropriate techniques.

9.2 Building and Structures

This section provides a description of the main buildings that have significant impact on the construction cost estimate. These buildings are:

- Reactor Building (RB)
- Reactor Service Building (RSB)
- Reactor Auxiliary Building (RAB)
- Personnel Service Building (PSB)
- Make Up Water and Auxiliary Boiler Building (MW & ABB)Control Building (CB)
- Radioactive Waste Management Building (RMB)
- Hot and Cold Machine Shops (H&CM)
- Power Conversion Building (PCB)

9.2.1 Reactor Building

Building Function

The Reactor Building shall, as its primary function:

- Provide a boundary, surrounding the reactor primary coolant system module, capable of withstanding design basis events and limiting the release of radio-nuclides.
- House the contained systems and components including the reactor vessel and four intermediate heat exchangers connected to the reactor vessel by four cross vessels.
- Protect housed components from natural phenomena such as tornado, earthquake, and flood.
- Provide passive heat dissipation as required to maintain temperature limits for housed systems, structures, and components for all design basis events (including air plane crash).

In addition, the Reactor Building shall:

- Provide space for in-service inspection and maintenance of housed system components.

- Provide access plugs, hoistways, access ways and laydown space for replacement of housed system key components.
- Provide the necessary degree of radiological shielding, including personnel protection.
- Provide necessary features for decontamination and decommissioning.
- Provide safe access and egress for personnel during periods when the building is occupied.

Building Description

The Reactor Building is a monolithic, reinforced concrete building. The Reactor Building is set predominantly below grade to take advantage of embedment in order to minimize earthquake induced loadings, in addition to providing advantages for refueling at grade.

The building is configured as a 31.7 m outside diameter cylinder from elevation (–) 44.55m to elevation -9.3 m. At elevation -9.3 m, the shape of the building changes to a rectangular structure. The size of rectangular structure is 48m by 68 m.

The cylindrical portion of the building is subdivided into a number of vertical cells that house the reactor vessel, four intermediate heat exchanger vessels (IHX), and four cross-vessels. Four IHX vessels are set in circular agreement at a 60-degree angle around the reactor vessel.

The below-grade placement of a large portion of the building provides improved response to seismic events protection from missiles generated by tornados and other events. The above-grade portion of the RB shall be designed not to collapse under design basis conditions and to resist missile impact.

An arrangement drawing of the Reactor Building, and the cross-sections, A-A & B-B, are included here as Figures (to be supplied later), respectively.

Cylindrical Portion of the Reactor Building

The reactor structure is a monolithic, cast-in-place, reinforced concrete cylindrical structure located below grade and supported by the base mat at elevation (-) 44.55 m. The roof slab over the cylindrical structure is a part of the operating floor for the RB at elevation 0.15m.

The reactor structure has the following dimensional characteristics:

- A 1.5 m thick top slab (top of concrete is at elevation 0.15 meter).
- A 1.35 m thick cylindrical outer wall with an inside diameter of 29 meters.
- A 3.0 m thick base mat with a diameter of 32.7 meters

The Reactor internal structure consists of the following major components:

- The 1.35 m thick, reinforced concrete, enclosure of an octagonal shape separating the reactor cavity from the IHX vessels and the rest of the building envelope. This enclosure extends from the base mat to the operating floor
- The 1.0 m thick, reinforced concrete, radial walls separating IHX vessels.
- The 1.0 m thick, reinforced concrete, walls and floors separating two stairwells, equipment access shaft and other spaces from the IHX cavities.
- Concrete floors of varying thickness

- Structural Steel platforms providing access to various components and are integral with the interior steel liner.

The reactor vessel octagonal enclosure and radial walls also provide lateral support for the reactor and IHX vessels. A vertical support to the reactor and IHX vessels is to be provided at elevation -33.7 m below grade by a reinforced concrete “ring” slab or a system of corbels and cross beams, and/or snubbers attached to the reactor cavity enclosure. Doorways and other openings are provided to allow access, HVAC air transfer, and blowdown pressure relief between the various sub-compartments.

Rectangular Portion of the Reactor Building

The primary function of the rectangular portion of the RB is to house and protect system components including the Reactor Maintenance Bridge Crane and Fuel Handling Equipment.

The rectangular portion of the building is divided into several compartments, occupied by the Reactor Cavity Cooling System (RCCS) ducting and the cavity vent path, protection and control equipment, and other nuclear auxiliaries.

The at-grade roof slabs of the RB form part of continuous operating floor for conducting refueling and other operations and maintenance activities. The operating floor is enclosed by a steel-framed Maintenance Enclosure with insulated metal roofing and siding. The structural framework for this enclosure supports the 300- tonne crane that also serves Reactor Service Building.

9.2.2 Reactor Service Building

Building Function

The reactor support activities to take place within the RSB include the following functions:

- Provide for handling, storage, inspection and shipment of new and irradiated fuel, target, and reflector elements.
- Provide for storage, decontamination, and maintenance of fuel and target handling equipment.
- Provide for storage, decontamination, and maintenance of some reactor system components.
- Provide Maintenance for Bridge Crane and Fuel Handling Equipment.
- Ensure the safety of operating personnel and the public on and off site by limiting releases of radioactive nuclides.
- Additionally, the RSB protects the systems and components from various hazards.

The function of the Washdown Bay is to allow for truck and rail access to the Reactor Service Building and to facilitate decontamination of vehicles and casks if required.

Building Description

The Reactor Service Building is a multi-level, reinforced concrete structure set entirely below grade at the southern end of Reactor Building and have a common operating floor and a common above-grade enclosure with the RB.

The RSB is divided into several functional areas as follows:

Operating Floor: The operating floor is set at elevation 0.15m. It is the area through which materials and personnel enter and leave the Nuclear Island. New fuel, reflectors and targets arrive by truck through the

Washdown Bay. They are then moved across the operating floor to the freight elevator which takes them to the lower elevations of the building where the new fuel/reflectors and target storage areas are located.

Embedded in the operating floor is a set of rails over which fuel and target handling equipment is moved.

The RSB is served by the 300-tonne bridge crane that also serves the RB.

Below-Grade Area: The following facilities are positioned below the operating floor:

- Storage area for the main circulators
- Storage area for the Shutdown Cooling System circulators
- Fuel Sealing and Inspection Facility
- Reactor Equipment Service Facility
- Intermediate Spent Fuel Storage

9.2.3 Reactor Auxiliary Building

Building Function

The primary function of the Reactor Auxiliary Building (RAB) is to house and support elements of the non-safety related equipment of the Reactor Cavity Cooling System, Shutdown Cooling System Cooling Water, Class 2E UPS, and other systems that support the safe operation of the Reactor System. Additionally, the Reactor Auxiliary Building protects these systems and components from various hazards, both internal and external.

The Reactor Auxiliary Building serves to limit both personnel access and area radiation levels as required to control occupational radiation exposure while providing access for plant operation and maintenance.

Building Description

The Reactor Auxiliary Building is a monolithic, reinforced, cast-in-place, concrete building, a large portion of which is set below grade. At grade, its overall dimensions are 13 meters by 41 meter. The below-grade portion of the building extends from elevation 0.15 m, down to elevation -9.3 m at the top of the mat. The building is adjacent to the RB.

9.2.4 Personnel Service Building

Building Function

The primary functions of the Personnel Service Building are as follows:

- Provide controlled access and egress to and from the RB above-grade enclosure and the Radioactive Waste Management Building.
- Houses offices for personnel involved in refueling and outage operations
- House facilities for monitoring and controlling personnel exposure to radioactivity.
- House the Main Change Area and Lunch Room.
- House the Fuel Handling Control Room and associated facilities

Building Description

The Personnel Service Building is a single-story, grade-level structure, located at the south end of the RSB and is adjacent to the RMB.

9.2.5 Make-Up Water and Auxiliary Boiler Building

Building Function

The primary function of the MW & ABB is to house system components including:

- Support Facilities HVAC System
- Plant Chilled Water System
- Compressed Air System
- Auxiliary Boiler System
- Potable Water System
- Demineralized Water System
- Makeup Water System

Building Description

MW & ABB is a single-story steel structure with insulated metal siding, metal roof deck and built-up roofing. The MW & ABB is 30 m wide by 34 m long. It is supported by spread footings. The ground floor is a 0.15m thick concrete slab on grade. The major equipment will be supported by individual concrete foundations.

The Water Treatment area is approximately 15 m high. The Auxiliary Boiler area is 10 m high. These two areas are to be separated by a concrete wall. A 5-ton bridge carne is required.

9.2.6 Control Building

Building Function

The functions of the Control Building are as follows:

- House the plant's main control room complex, Technical Support Center and associated electrical and HVAC equipment.
- House other functional areas such as the control room, the computer room, the electrical equipment room and miscellaneous rooms.
- Provide habitability during normal and accident conditions.

Building Description

The Control Building is a 15 m wide and 30 m long, three story structure of steel and reinforced concrete construction which is set at grade. The Building is supported on spread footings.

9.2.7 Radwaste Management Building

Building Function

The primary function of the RMB is to house system and components including:

- Liquid Radioactive Waste Management System
- Solid Radioactive Waste Management System
- Gaseous Radioactive Waste Management System

The functions of these systems are to collect, segregate, treat, store and dispose of radioactive waste originating within the plant. Additionally, this Building protects these systems and components from various hazards, both internal and external.

Building Description

The RMB is a at grade supported steel framed structure with insulated sheet metal siding and roof decking. Concrete cubicles are provided for components that require radiation shielding. The Structure is 26 m wide by 32 m long. The RWMB operations are served by a 13.6-tonne bridge crane.

9.2.8 Hot and Cold Machine Shop Building

Building Function

The function of the HCMSB is to house components to maintain and repair the nuclear plant components:

- Clean (non-radioactive) shops – machine, welding, sheet metal, electric, instrumentation, insulation, paint, and carpentry, along with associated tool rooms;
- Decontamination shop;
- Shops for contaminated equipment – machine shop, electric, instrumentation, a seal overhaul shop, measuring and testing equipments, calibration shop, equipment and tools decontamination shop, along with associated tool rooms;
- Contractor fabrication shops;
- The shop for training mockups;

Building Description

HCMSB is a single-story steel structure with insulated metal siding, metal roof deck and built-up roofing. The HCMSB is 20 m wide by 28 m long. It is supported by spread footings.

9.2.9 Power Conversion Building

Building Function

The primary function of the Power Conversion Building is to house and support non-safety related components associated with the conversion of thermal energy to electrical energy.

These major equipments and systems are:

- Gas Turbine and Compressor
- Heat Recovery Steam Generator
- Steam Turbine
- Feedwater System

- Condensate System
- Electrical Generators and components associated with various plant electrical systems.

Additionally, this Building protects these systems and components from various hazards, both internal and external.

Building Description

The PCB is a 50 m wide by 60 m long, three–level structure. Major levels are: the ground level, the mezzanine and the operating floor. The building is a braced steel-framed structure with metal siding, metal roof deck, and built-up roofing. The structure is supported by reinforced concrete mat foundation. A bridge crane is located at approximately 25 m above ground floor over the operating floor. A gantry crane is also needed to perform light lifts.

The mezzanine level is a reinforced concrete slab at approximately 7 meters above grade supported by structural framing.

The operating floor is a reinforced concrete slab at approximately 11.5 m above grade supported by structural framing.

PLANT LAYOUT DRAWINGS

The following four figures comprise the plant layout drawings:

Figure 9-1: SK_C001 Plot Plan Study

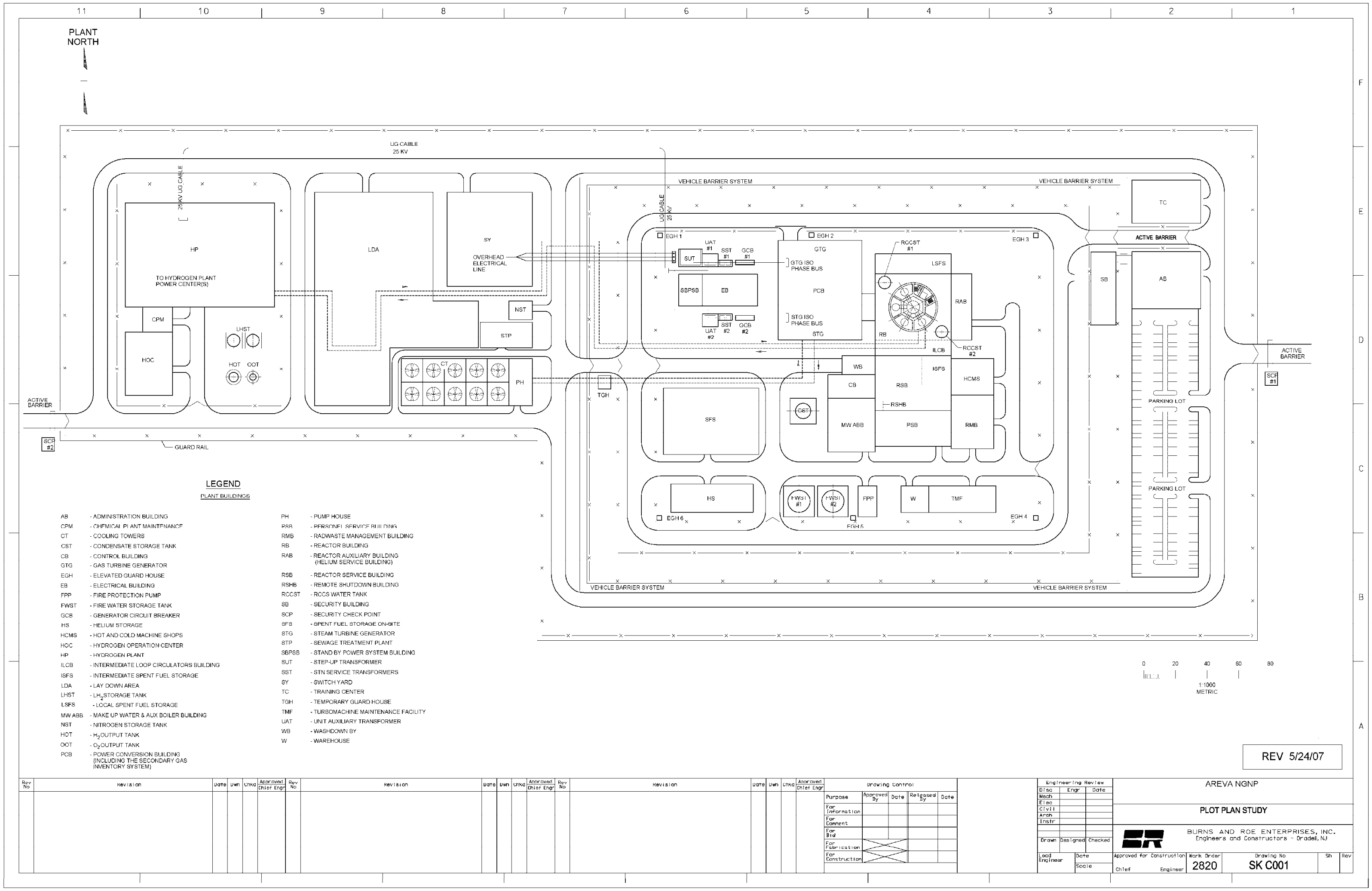


Figure 9-2: SK_S001 NGNP - Plan

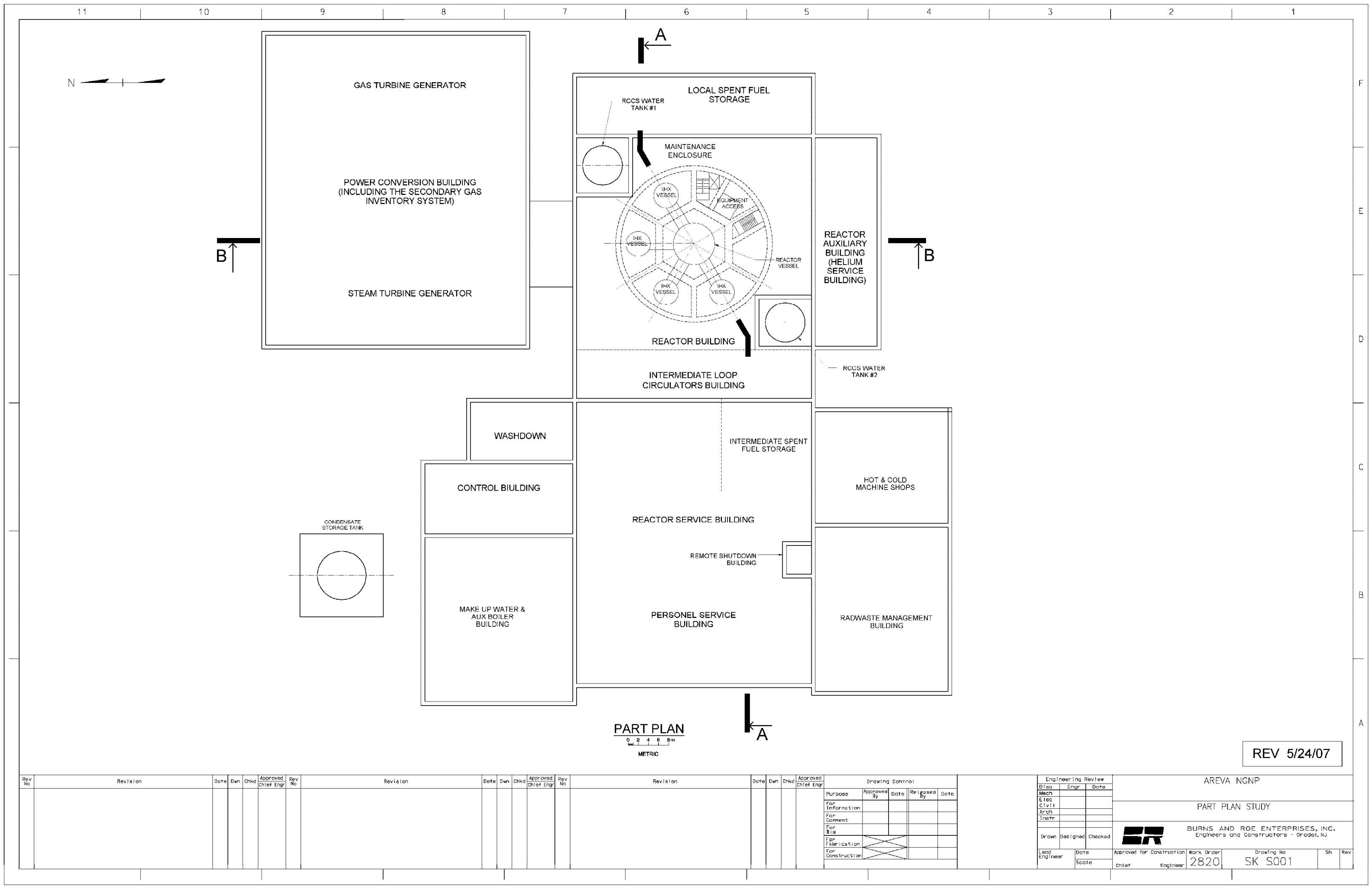


Figure 9-3: SK_S002 NGNP - Elevation - Section AA

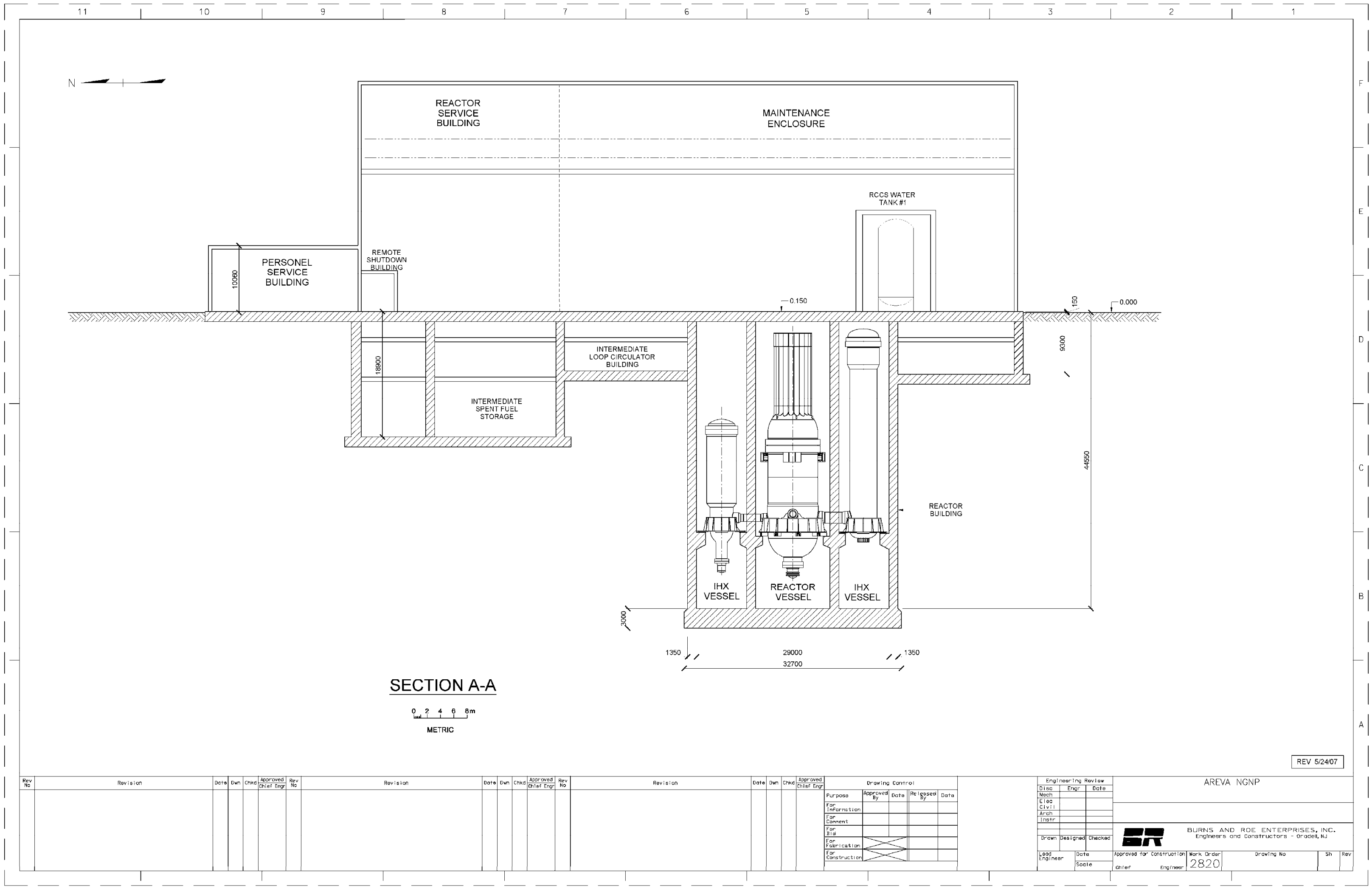
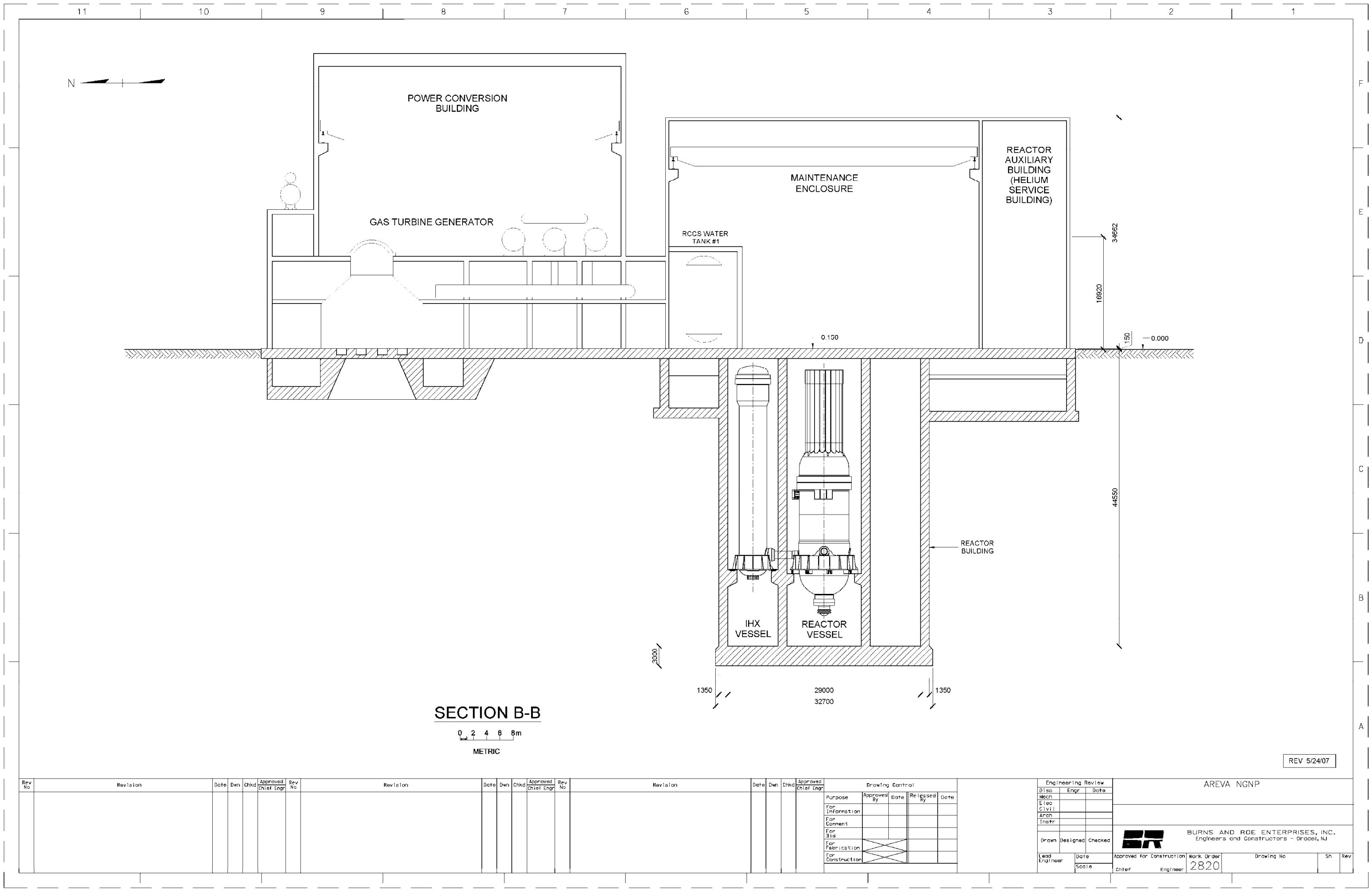


Figure 9-4: SK_S003 NGNP - Elevation – Section BB



10.0 OVERALL NUCLEAR SYSTEM OPERATION

The design and operation of the AREVA NGNP are based on the following objectives:

- Develop and demonstrate an advanced VHTR nuclear heat source (NHS) that supports the requirements for electricity and hydrogen production, while minimizing development requirements and risk.
- Provide a flexible platform for the demonstration of various high efficiency hydrogen production technologies, including, high temperature thermo-chemical and steam electrolysis processes.
- Demonstrate the design, construction, operation, safety, and licensing of the NGNP to support the development of a new generation of commercial nuclear plants featuring passive and inherent safety characteristics for electricity generation and process heat applications including hydrogen production.
- Provide a means for the testing and irradiation of system components, fuels, and materials, etc. as necessary to characterize design alternatives and related implementation constraints.

The performance objectives of the AREVA NGNP concept fully consider the functions and requirements that have been identified by INL as the basis for the NGNP project [8]. Table 10-1 cross-references those NGNP functional requirements against AREVA's NGNP operational framework plan.

Table 10-1: Operational Considerations in Support of NGNP Requirements

Top Level NGNP Requirement	Operational Considerations
1. NGNP prototype shall be designed, constructed, licensed, and operating by 2020 with initial operations in 2018.	To minimize risk and assure schedule, the NGNP operational framework should retain, to the degree possible, similarity to existing power generation plants.
2. NGNP prototype design configuration shall consider cost and risk profiles to ensure that the NGNP is directly scalable for future commercial plant deployment.[8]	
3. NGNP prototype shall produce high efficiency electricity and generate hydrogen on a scale that sets a foundation for future commercial deployment. [8]	The NGNP operational framework explicitly acknowledges the dual objectives of power generation and hydrogen production within the procedural framework defining operational modes as well as normal and unanticipated transitions between operational modes.
4. NGNP prototype shall be licensed by the NRC as a commercial cogeneration facility producing electricity and hydrogen. [8]	The NGNP operational framework will explicitly identify key measures, based on safety, component tolerances, and system efficiencies, etc., to define an operational envelope necessary for plant licensing through formal demonstration of the NGNP's passive and inherent safety features.
5. NGNP prototype shall include provisions for future testing. [8]	The NGNP operational framework shall accommodate planned testing programs and, to the extent possible, define procedures and measures for accommodating as yet-to-be-determined future missions for the NGNP plant.
6. NGNP prototype shall enable demonstration of energy product and processes utilizing its nuclear heat source. [8]	The NGNP operational framework will provide for full demonstration of processes relevant to energy production and heat rejection, including those processes with a direct bearing on safety, system/component performance, and equipment survivability.
7. The project shall include identification of necessary and sufficient R&D technical scope and priorities. [8]	The NGNP operational framework will be described to a level of detail to support project R&D activities

A description of the overall nuclear system operation is provided in this section.

10.1 Nuclear System Performance

The AREVA NGNP design is based on a commercial scale NHS that will be for both electricity/process heat generation and hydrogen production. This is a key advantage of the indirect cycle architecture selected for electricity generation and will support the initial commercialization of VHTR technology via the existing and expanding markets for electricity generation.

Consistent with the INL requirements, the AREVA NGNP design is based on a once-through low-enriched uranium core capable of high burnup. The plant will be designed for a 60 year life span and provisions will be made to replace components with shorter lifetimes. The number of primary system components to be maintained is minimized by the selection of the indirect cycle architecture and the remaining systems and components in the power generation and hydrogen processing sections will be readily available for inspection and repair/replacement. The indirect cycle design also minimizes the transient coupling of the NHS and the heat utilization systems.

The AREVA concept for the NGNP combines the VHTR NHS with a multi-purpose energy utilization architecture that will demonstrate both advanced electricity and process heat generation and hydrogen production applications. The AREVA NGNP design incorporates an indirect combined cycle power generation system that is capable of using the full thermal capacity of the reactor, 565 MWt at an efficiency greater than 46%, exceeding the NGNP requirement of 44%. Heat from the NHS is supplied to the Power Conversion System (PCS) through an intermediate heat exchanger (IHX) for electricity production. The heat supplied to the PCS first drives a Brayton topping cycle and then a Rankine bottoming cycle. The NGNP Brayton topping cycle is assumed to use a nitrogen-based fluid to allow the use of air-breathing gas turbine technology.

The heat transport architecture of the AREVA NGNP provides an optimum platform for the demonstration of high-temperature hydrogen production processes such as thermo-chemical and electrolysis, while allowing full utilization of the reactor output for electricity generation, when the hydrogen production section is not in service. The 60 MWt secondary helium loop that supports the energy requirements of a collocated Hydrogen Production Pilot Plant (HPPP) minimizes any chance of radiological contamination, including tritium, and allows modification of the hydrogen production section, independent of operations in the electricity generation portion of the plant. The 900 ºC temperature provided at the process coupling heat exchanger fully supports the NGNP efficiency requirements, based on data pertaining to the sulfur-iodine thermochemical water-splitting cycle. The tertiary loop and associated circulator further provide for independent control of heat transport to the hydrogen production process and minimizes the potential for adverse interactions with the remainder of the plant.

The indirect cycle architecture of the AREVA NGNP, in conjunction with the energy transport architecture provides a flexible basis for future testing, including those for demonstrating investment protection and safety margins. This is facilitated by the simple and reliable design of the reactor and primary heat transport loop that minimize the likelihood of unforeseen consequences that might result from unusual operating conditions associated with testing.

10.1.1 Design Point at 100% Rated Power

Table 10-2 provides the operating parameters for AREVA's NGNP NHS and associated systems at 100% rated power. This set of parameters provides for optimum plant power generation efficiency, considering the use of proven technologies.

The main thermo-dynamic parameters which influence the plant efficiency are:

- Temperature of the heat source and sink: the plant efficiency increases with the core outlet temperature.

- Core temperature difference. The gross plant efficiency (efficiency of the thermo-dynamic cycle) increases when the core temperature difference decreases, but this decrease entails a primary flowrate increase and then an increase of the required electrical power of the circulators that decreases the net plant efficiency.
- The gas expansion factor in the turbine is a parameter that is linked to the core temperature difference. This parameter can be optimized because it fixes the gas Heat Recovery Steam Generator (HRSG) inlet, the hot source for the tertiary steam/water circuit.
- The tertiary circuit arrangement (number of pressure levels in the HRSG, number of steam turbine stages, regenerating flows, reheats,…). This choice has to be optimized closely with the secondary circuit.

Table 10-2: Recommended NGNP Operating Parameters

Parameter	Value
General Plant	
Thermal Power	565 MWth
HPPP Power Requirements	60 MWth, 20 MWe
Reactor Outlet Temperature	900 °C
Reactor Inlet Temperature	500 °C
System Configuration	H_2 and PCS in parallel
Number of Loops	4 loops 3 with tubular IHXs for PCS 1 with compact IHX for H_2 production
Secondary Temperatures	450 – 850 °C for PCS 475 – 875 °C for H_2 production
System Pressure	5 MPa
Gross electrical power	279 MW (w/o HPPP); 249 MW (w/ HPPP)
Plant auxiliaries	About 18 MW (w/o HPPP); 38 MW (w/ HPPP, S-I)
Net electrical power	261 MW (w/o HPPP); 211 MW (w/ HPPP)
Gross efficiency	>49%
Net efficiency	>46% (w/o HPPP)

The HPPP is assumed to be directly coupled with the NGNP NHS through the High Temperature Heat Transport Loop (HTHTL), which carries heat between the NHS and the hydrogen plant. The HTHTL is a closed helium system connected to the primary loop via an IHX and the other connected to a tertiary loop serving the hydrogen plant via a process heat exchanger. The hydrogen plant is assumed to demonstrate two high temperature hydrogen production processes: the direct Sulfur-Iodine (S-I) thermochemical process and the high temperature electrolysis (HTE) process.

The S-I process puts a greater burden on the high temperature process heat delivery capability of the NGNP and, as such, governs both the process efficiency and the process equipment sizing for the thermochemical hydrogen production process. The process efficiency is essentially the thermodynamic efficiency of the process. It provides a measure of the net energy content of the hydrogen produced for a given thermal energy input to the process. The temperature also affects the reaction yields within the process. While this does not directly affect the process efficiency, it has a significant impact on the size of the piping, pumps, heat exchangers and other equipment used in the process plant. Therefore, this also affects the plant economics.

HTE requires a relatively small amount of thermal energy and a large amount of electrical energy. Therefore, the plant's electricity generating efficiency is very important for HTE. The HTE process does require process temperatures in the neighborhood of approximately 800°C. However, the process performance does not vary strongly above this temperature. So there is little incentive to apply significantly higher temperatures for HTE.

The temperature drop between the reactor and the hydrogen process must be considered in assessing the impact of the hydrogen process on the required reactor outlet temperature. This temperature drop must take into account all temperature drops between the NHS core outlet and the hydrogen process peak chemical reactor temperature. Specifically, the resulting temperature drop includes the temperature drop across the main IHX between the NHS primary circuit and the HTHTL, heat losses in the HTHTL, and the temperature drop across the process heat exchanger (chemical reactor) which separates the intermediate heat transport loop fluid and the chemical reactants. For a well designed commercial scale HTHTL, the temperature drop is expected to be minimal. The actual drop for the NGNP will have to be evaluated considering the smaller heat transport lines and the actual distance to the HPPP. This evaluation is beyond the scope of this study, but it is reasonable to assume that this temperature drop will be small compared to the heat exchanger temperature drops for a well designed NGNP system.

Therefore, an NGNP hydrogen production process temperature 50°C below the reactor outlet temperature is assumed. This is based on a 25°C approach temperature in the IHX and a 25°C approach temperature in the process heat exchanger. This represents reasonable performance based on the anticipated heat exchanger technology.

For this application the main data are:

- power to be supplied: 60 MWth, 20 MWe
- secondary HTHTL loop pressure: 5 MPa,
- secondary IHX output temperature: 875 ºC,
- tertiary IHX output temperature: 850 ºC

10.1.2 Performance Limitations and Uncertainties

Derivation of performance measures considered several factors including thermal-hydraulic dynamics, material limits, and economics based on available experience in the design of high temperature gas reactors, in particular, and nuclear power plants, in general. AREVA has completed several trade studies examining uncertainties associated with those basic parameters with the greatest influence on the performance of the NGNP, as represented by the design point presented in Section 10.1.1. These include:

- Reactor inlet and outlet temperature
- Parallel vs. series loop configuration and number of loops
- Secondary temperatures and heat exchanger performance
- System pressure

The value of the reactor power of 565 MWth with a reactor inlet and outlet temperature of 500 ºC and 900 ºC respectively is defined by the objective that fuel peak temperature remain below 1600 ºC during a loss-of-flow accident.

Limitation on reactor outlet temperature are based on the minimum needed by the HPPP and the maximum that sensitive component materials can sustained, such as turbine blades, hot duct and core support structure materials. The value of 900 ºC reflects contemporary understanding of the heat-resistance of VHTR structure materials.

Similarly, the reactor inlet temperature has an influence on fuel temperatures. With a smaller core temperature differential, power efficiencies increase, while fuel temperature uncertainties decrease. Concurrently, higher reactor inlet temperatures can challenge heat loads on the reactor vessel and can impact the maximum core thermal power or impose a requirement for thermal protection of the vessel. The value of 500 ºC meets the requirements for both fuel and vessel temperature limits.

Incorporating the HPPP and the anticipated R&D activities impact several of the design point considerations. Since the objective of the NGNP NHS is to provide high temperature conditions to the HPPP, the temperature range becomes a key plant system variable for which design optimization must consider. To provide the most flexibility with regard to R&D activities, the range of temperature supported by the NGNP should be as large as practical. For the reactor inlet and outlet temperatures of 500 – 900 ºC, the broadest practical temperature range for the HPPP is on the order of 500 – 850 ºC. Such a broad temperature range is not productive for the PCS; hence, a parallel loop configuration is preferred. In addition, a separate loop for the HPPP provides the opportunity to examine alternative working fluids, different IHX designs, and provides for independent control of HPPP heat load requirements.

With this decision point, the number of loops question logically follows. Resolution of this question is drawn from design considerations of the IHX and circulator. In both situations, a three loop configuration appears to optimize the current state-of-the-art capabilities and economics.

Optimal secondary temperature is primarily a tradeoff of improved plant performance weighed against the cost of increased IHX effectiveness. System performance can improve significantly if heat is delivered to the process or PCS at higher temperature. However, the IHX cost also increases significantly as higher effectiveness is specified.

The approach temperature between the primary and secondary fluid depends on the IHX effectiveness. In order to achieve a 50 ºC approach temperature such that the hot secondary fluid leaving the IHX is at 850 ºC, an IHX effectiveness of 89% is required. For a 25 ºC approach, the required effectiveness increases to 94%. A key point which makes this difference very significant is the cost related to higher effectiveness. As such, this is impractical for the PCS loop IHXs; however, for hydrogen production, the benefit of increased effectiveness is greater.

Selection of a pressure value in the primary circuit is a balance of vessel cost with system pump power. Increasing pressure generally increases vessel cost and reduces pumping power requirements. For indirect cycle systems, the pressure difference across the IHX can be important in determining the operating stress in the IHX and the corresponding component lifetime. In using multiple loops and, hence, multiple circulators, a lower primary pressure is possible because of reduced pumping power requirements relative to a configuration involving fewer loops. As such the target NGNP primary coolant pressure is 5.0 MPa.

Beyond these major design point parameters, several component parameters influence plant efficiency. These are identified in Table 10-3. The optimization of these parameters must balance the primary and secondary system measures identified above within the operation domain of the each component, which is highly dependent on possible operational states, control strategies, and general parameter variations during operation. Reported values represent best-estimate results from AREVA’s experience in the design of high temperature gas reactors.

Table 10-3: Main Component Parameters Impacting the Plant Efficiency

Circuit	Parameter
Primary circuit	Primary coolant/fuel heat transfer
	Pressure losses in the primary circuit
	Circulator efficiency
Secondary circuit	IHX thermal efficiency
	Polytropic efficiency of the gas turbine
	Polytropic efficiency of the compressor
	Pressure losses in the secondary circuit
Tertiary circuit	Polytropic efficiency of the steam turbine
	Steam generator design
Electricity production	Generator efficiency

10.2 Operating Modes

The NGNP plant is designed for the control and operation of a single reactor module, a power conversion system consisting of a Brayton topping cycle and a steam bottoming cycle, and/or the delivery of process heat by operators from a central control room. Monitoring systems enable maintenance and operation tasks during normal and abnormal operation. The main objective in defining the conditions for steady-state operation is that normal operations, upsets, or unanticipated transients in either the NGNP or the hydrogen production plant must not compromise safe operation of the other portion of the plant. The main objective in developing control and operation strategies is to address the impact of operational changes or equipment failures in the HPPP on the operation of NGNP. Conversely, it is also necessary to address the impact of changes in the operations of the NGNP (power levels, electrical output) on the operation of the HPPP.

The main situations which simultaneously define basic plant operational strategies and the limitations of electrical power output capability are summarized in the bulleted list below:

- Power Operation, PCS only
- Power Operation, PCS and HPPP
- Power Operation, HPPP only
- Start-up from and shutdown to cold conditions
- Shutdown
- Refueling
- NGNP Safety Testing
- HPPP Testing

10.2.1 Power Operation, PCS Only

This state covers the reactor power range from approximately 15% - 30% (TBD) to 100% of rated capacity. Operation in the energy production state usually occurs only if a turbine-generator is operating and able to accept the reactor energy output.

The power operating mode includes the automatic power control in the power output range between a technical minimum and the Nominal Power and the operating mode at house load (about 3% of rated power). Operation in this mode usually occurs only if a turbine-generator is operating and able to accept the reactor energy output. In this range the power unit is capable of changing reactor power output under automatic control at a rate up to and including 5% of rated output per minute.

10.2.2 Power Operation, PCS and HPPP

For HPPP operation, alignment of the HPPP loop components with the PCS loops will be necessary. The requirement for operation and control independence will likely dictate the final procedures for this activity. Additional house load of up to 3% rated power may be necessary for some HPPP applications (total of 6% rated power with full PCS operation).

10.2.3 Power Operation, HPPP Only

A common operation mode is envisioned in which only the power necessary to sustain the process heat needs of the HPPP is supplied by the NHS. In this configuration the plant will need to respond to a process heat demand of up to 60 MWth for HPPP operations. The electrical demand, expected to be up to 20 MWe, would likely require a source other than the NGNP.

10.2.4 Startup and Shutdown

This state represents the transition between the shutdown and energy production modes. It covers the power range from reactor subcritical to approximately 15% – 30% power. In this state, all plant service and supporting systems are made functional and ready to support electrical and/or hydrogen production operation. During this mode, the turbine-generator is brought on-line or shut down.

The startup objective is to start the plant from a cold shutdown state to power state (technical minimum) within 24 hours. A four stage approach is envisioned involving

- Stage A: Preheating of the primary circuit (including IHX) to obtain 500°C at the reactor outlet temperature, duration of this stage of about 13 hours.
- Stage B: Coupling NHS with PCS, duration of this stage of about 4 hours.
- Stage C: Rise in power to technical minimum (TBD), duration of this stage about 7 hours.
- Stage D: Rise to nominal power, at a rate of about 5%/min, duration of less than 1 hour

Shutdown is the reversal of the startup model. The primary objective is that the plant is ready for refueling within 24 hours.

10.2.5 Shutdown Mode

During this state the reactor is sub-critical. The residual heat is normally removed by the SDHRS via the primary circuit in forced convection.

In case of unavailability, repair or maintenance of SDHRS, IHX or primary circulator, residual heat is removed by SCS or RCCS.

10.2.6 Refueling Mode

This state is a special case of the shutdown state. The refueling mode represents transition of the shutdown reactor to the state necessary for core off-loading/reloading. This mode is characterized by low reactor temperature and low pressure in the circuit. In addition to the reactivity requirement, unique primary system pressure and core inlet temperature requirements are maintained to allow refueling of the core.

10.2.7 NGNP Safety Testing

As a prototype for a possible fleet of Generation IV commercial nuclear power plants, the NGNP is expected to demonstrate the plant's passive and inherent safety features through a series of tests emulating various anticipated operational occurrences and design basis events. Unique operation and control strategies are envisioned such that key measures, based on safety, component tolerances, system efficiencies, etc. can be identified to define the operational envelope expected for future licensing activities related to a commercial plant while providing sufficient protection of the plant staff, the public, and the investment in the various NGNP systems, structures, and components.

10.2.8 High Temperature Testing

To characterize the performance of the processes associated with the HPPP as a function of temperature, the NGNP will be expected to provide helium temperatures in the range of 1000 – 1100 °C. To sustain such temperatures, the NGNP will provide only the power demand required by the HPPP and shutdown helium circulation in the power generation loops. This testing mode could also facilitate the study of as yet-to-be-determined future missions of the NGNP plant that may require alternative components, materials, and/or fluids.

10.3 Plant Protection and Operability

Underlying common operation conditions, strategies for plant protection and operability must be in place. Key components of the plant influencing its operability and safety are to be identified in the design and monitored during the plant operation. Potential modes of failure and maintenance actions are to be defined for minimization of their appearance frequency and early detection of both deviations from normal operation conditions and abnormal operating performances indicating commencement of such failures.

As a consequence of the specific properties of the NGNP ensuring passive safety of the reactor, neither active engineered safety features nor operator actions are necessary to provide protection for a population beyond the plant boundaries. Both safety- and non-safety-related instrumentation and controls are considered in addressing ideal approaches to fault recognition and mitigation. Interfacing the operator to the necessary plant information is the function of the Process Information and Control System (PICS) and the Safety Information and Control System (SICS).

The PICS and SICS are designed to provide relevant and reliable information to the operator on the status of the Reactor Protection System (RPS) and the plant, in general. The SICS is the central location for alarm annunciations from the RPS in the control room. It will display the condition of the RPS in normal operation, during anticipated operational occurrences and accidents. In addition to these information tasks protection functions can be manually initiated from the SICS, e.g., the start-up bypass. In essence, the SICS contains analog indications for displaying process variables, initiation status signals from instrumentation channel groups, actuation signals from the reactor protection system and check-back signals from actuated components. The PICS is expected to provide supplementary information for the operator, e.g., the individual alarms that make up a group alarm on SICS.

10.3.1 Reactor Protection System (RPS)

The reactor protection system (RPS) is required to monitor and process variables essential to the safety of reactor and the environment, to detect accidents and automatically initiate protective actions. According to the concerned accident, the reactor protection system shuts down the reactor and actuates the protective actions required for mitigating all accidents.

The choice of monitored process variables, derivation of suitable initiation criteria and the generation of actuation signals for protective actions are performed on the basis of the accident analyses.

The protection system will implement the automatic and manual actuation of safety systems and the relevant monitoring functions which are necessary to reach the controlled state in case of abnormal events by initiating reactor trip and starting the safety systems:

- Reactivity control,
- Residual heat removal,
- Limitation of radioactive releases at the site boundary to an acceptable limit and maintaining integrity of the primary and secondary systems.

The RPS initiated protective actions are of highest priority, thus the RPS commands override the actions initiated by PAS and RCLS with the aid of PACS.

Information on the process variables and actuators is transferred via MSI and PACS respectively directly to SICS as well as control and manual actions are transferred from SICS to PACS. During normal operation an operator terminals can be used for these purposes. Service and diagnostic tasks are performed by DMS / SPACE & ES via plant bus, Gateway and MSI on the RPS.

10.3.2 Plant Protection System (PPS)

The PPS performs equipment unit protection. It contains the high-grade protection circuits for important equipment units and the protective interlocks for all equipment units, thus protecting these active components from damage or malfunction caused by unacceptable operating conditions or operator error.

The PPS initiated protective actions are of high priority, thus the PPS commands override the actions initiated by PAS and RCLS with the aid of PACS.

Information on the process variables is transferred via MSI and PACS directly to SICS as well as control and manual actions are transferred from SICS to PACS. During normal operation operator terminals can be used for these purposes Service and diagnostic tasks are performed by DMS / SPACE & ES via plant bus, Gateway and MSI on the PPS.

10.3.3 Process Automation System (PAS)

The PAS system is in charge to monitor and control the plant in normal operation conditions.

A further task of the PAS is to monitor and operate the mechanical systems for operation of the plant in all plant conditions as long as the PAS and the mechanical systems are operable.

PAS performs the typical application functions:

- Closed loop controls,
- Sequence controls,
- Combinational controls,
- Data acquisition functions,
- Drive control functions,
- Alarm and information functions

During normal operation PAS is operated by operator terminals. In case of loss of operator terminals or under periodic testing several systems of PAS can be operated by local control. Diagnostic tasks are performed by DMS/DS via plant bus.

PAS takes credit from components within the safety-related systems, but the PAS induced actions are of lowest priority, thus the PAS commands are overridden by the actions initiated from the safety-related systems and PPS with the aid of PACS.

10.4 Transient Operation and Control

As is customary with modern reactor designs, plant control and protection systems are to be highly automated. An operator is expected to supervise all automatic control actions and have the means to control actions during both anticipated and unanticipated events. Use of a common, integrated control system for the two portions of the plant (nuclear and hydrogen) is not desirable. It is considered preferable that the two systems be capable of being operated independently.

Several controllers are available to respond to normal and off-normal operation demands. Table 10-4 identifies a partial list of such controllers and their expected function in plant control.

Table 10-4: Controllable Parameters with Suggested Actuators

Parameter	Possible Actuators to be Used
Electrical Power	Secondary By-pass, IGV, Steam Turbine By-pass Valves Primary Inventory, Secondary Inventory
Reactor Inlet Temperature or primary/secondary flowrate	Primary Circulator Speed
Reactor Outlet Temperature	Control Rods
IHX Primary/Secondary Pressure Difference	Primary Inventory, Secondary Inventory
Gas Turbine Speed	Turbine By-pass (when grid is disconnected)
Compressor Inlet Temperature	Feedwater Pumps
Steam HRSG Outlet Pressure	Steam Valves
Water HRSG Level	Feedwater Valves
Steam Turbine Speed	Steam Valves (when grid is disconnected)

10.4.1 Philosophy of Nuclear System Operational Control

Strategies for operational control of the NGNP plant will optimize the reliability, availability, and safety of the NGNP plant systems, structures, and components. This objective is achieved through the identification and implementation of state-of-the-art component designs, predictive maintenance methods, advanced information technologies, and incorporation of human factors and safety culture. First among these considerations are the following reliability, availability, and safety requirements, established in the framework of AREVA's system requirements for the NGNP [5]:

- The electric plant shall feed an outside commercial electrical grid system.
- The NGNP shall be designed in accordance with defense-in-depth principles and philosophy.
- Hydrogen production plant will require high temperature process heat from the nuclear plant.
- The plant shall be designed to the USA industrial Codes and Standards as necessary to meet the USA industrial and regulatory requirements.
- The NHS shall be passively cooled in case of loss of all off-site and on-site motive power.
- Public safety due to any accident at the nuclear plant shall not depend on public or personnel beyond site boundary.
- The nuclear plant shall be designed for an availability of greater than 90%

The primary uncertainty in achieving these goals is related to the thoroughness with which all engineering tasks have been completed. In addition, even after detailed design and construction is completed, uncertainty remains in several areas including:

- Actual plant configuration
- Completeness of design-basis and beyond-design-basis scenario lists and the accuracy of their probability assessment
- The ability to predict the transient response of the design to a given event scenario and to correctly quantify the uncertainty in that prediction and the margin to relevant undesirable plant states

A robust operation and control strategy, built upon thorough development methods, can have a strong influence on minimizing this uncertainty, thus enhancing the reliability of plant systems, structures, and components and overall plant safety and availability. In addition, with the choice of the VHTR design as the NGNP, several safety goals are achieved. Specifically, the VHTR uses inherent characteristics in a configuration which provides assured, simple, and passive safety response features tolerant of failures associated with engineered event-mitigation equipment and operator action.

10.4.2 Electrical Power Transients

Of greatest interest for anticipated transient operation are those controllers influencing electrical power.

The proposed control for electrical power transients uses a secondary by-pass that by-passes the IHX and the turbine to adapt the power conversion system to the demand. This control is performed by the Power Output Controller. The steam/water circuit is adjusted to the new conditions by means of the Steam Generator Level Controller and the Steam Pressure Controller. The primary circuit must also be controlled to adapt the primary circuit to secondary circuit and to maintain constant the core outlet temperature using the Primary Flowrate Controller and the Core Outlet Temperature Controller.

Optimization of the primary/secondary IHX pressure difference can be controlled by actuating the primary and secondary inventory valves.

The organization of this control is shown in Figure 10-1.

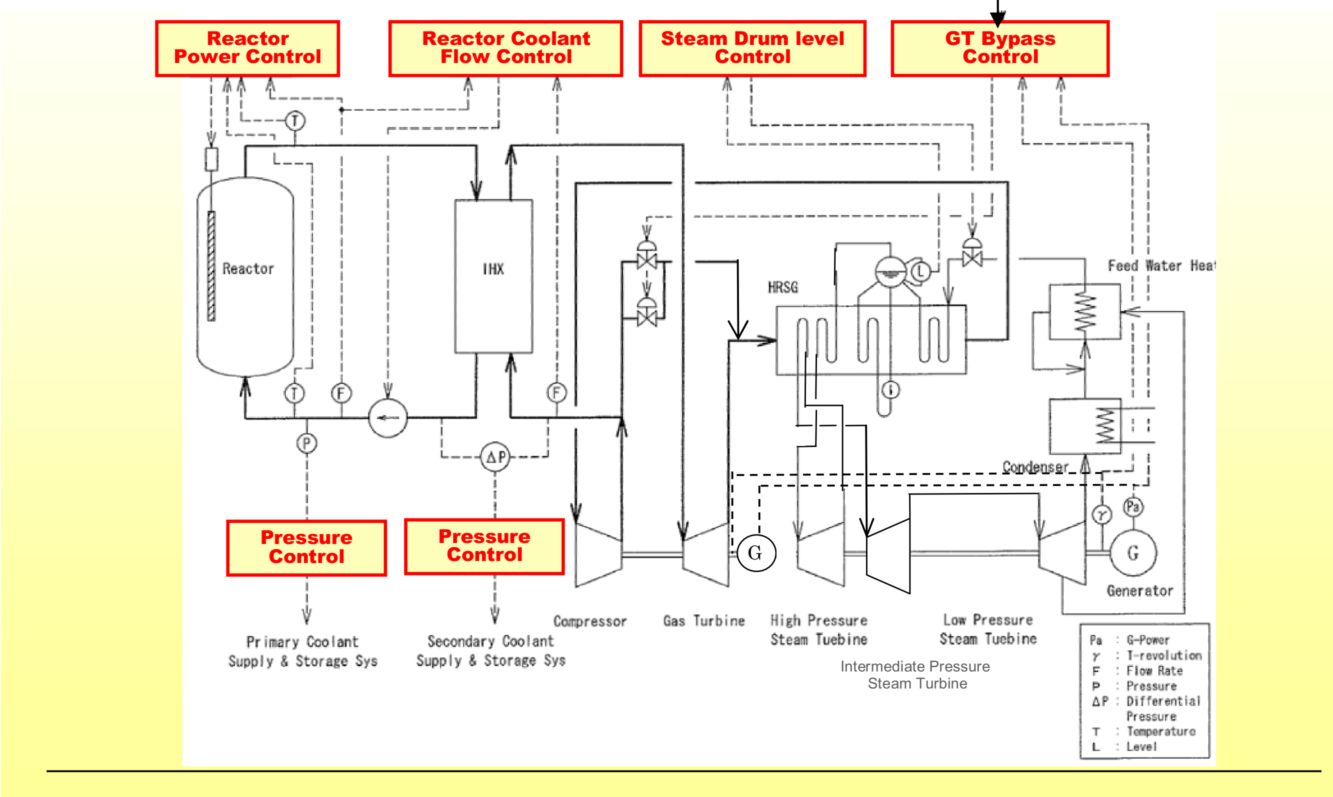


Figure 10-1: PCS Main Control Channels

Several anticipated load varying events to consider include:

- Load following
- Load rejection
- Reactor trip, and
- Turbine trip

10.4.2.1 Load Following Strategy

The event sequence is as follows:

- Gas turbine by-pass actuated (power output controller)
- Primary circulator speed change to be adapted to the secondary IHX flowrate (primary flowrate controller)
- Control rod movement to maintain constant the core outlet temperature (core outlet temperature controller)
- Eventually the secondary inventory valve is activated to reduce the primary/secondary IHX pressure

- Steam turbine valves (including by-pass) and feed water flowrate control to maintain the SG level and the steam turbine inlet pressure

10.4.2.2 Load Rejection

The event sequence is as follows:

- Gas turbine by-pass is actuated (power output controller) to avoid turbine overspeed then to maintain the turbine at its nominal value and to adjust the power output to house load.
- Primary circulator speed change to be adapted to the secondary IHX flowrate (primary flowrate controller)
- Control rod movement to maintain constant the core outlet temperature (core outlet temperature controller)
- Eventually the secondary inventory valves is activated to reduce the primary/secondary IHX pressure
- Steam turbine valves (including by-pass) and feed water flowrate control to maintain the SG level and the steam turbine inlet pressure. The house load can be supplied either by secondary side or by both secondary and tertiary sides.

If the house load situation can not be maintained, the turbine is tripped (by-pass is fully opened).

10.4.2.3 Reactor Trip

The event sequence is as follows:

- Protection signal initiates a rod insertion procedure
- Turbine trip by by-pass opening when output power is lower than a specified value
- Primary circulator trip
- Eventually the secondary inventory valves is activated to reduce the primary/secondary IHX pressure

10.4.2.4 Turbine Trip

The event sequence is as follows:

- Turbine trip signal actuates the gas turbine by-pass and the main steam stop valve
- Primary circulator trip to avoid a core inlet temperature increase due to the loss of heat sink (circulator trip initiated by temperature maximum at the core inlet)
- Reactor trip (rods drop)
- Eventually the secondary inventory valve is activated to reduce the primary/secondary IHX pressure

Note: In case of event leading to turbine trip, the secondary part of the IHX could be blown up to decrease the primary/secondary IHX pressure difference (closure of valves at the secondary IHX inlet and outlet and filling of the secondary IHX volume)

10.4.2.5 HPPP Trip

With the maximum load burden of the HPPP being only 80 MW (combined thermal and electrical), the event sequence is similar to the Load Rejection scenario presented in Section 10.4.2.2; however, the load drop is sufficiently small (14%) such that plant electrical production can be maintained following HPPP trip. In this case the scenario would be:

- Trip of the primary circulator of the process heat loop and GT Bypass Control (and partial opening of the bypass to the condenser)
- Core outlet temperature control with control rod insertion (Reactor Power Control) to quickly match reactor power to the PCS load at 90%.
- Control rods raise and main primary circulator speed control increases circulator speed to accommodate the 10% climb back to full power) and closure of the GT bypass to supply all core power (565MWth) to the PCS.
- Note: In full power operation of the PCS+HPPP (steady state), the main primary circulators should run at 90% of nominal speed and the GT Bypass should be slightly open (an alternative control proposal is to decrease the primary and secondary pressures).

10.4.3 NHS Control for Hydrogen Production

As a demonstration plant, control of the NHS for hydrogen production activities must be tolerate to all possible operation conditions anticipated for that application. As such, separate and independent control schemes are proposed to address the demand for a broad operation domain associated with the HPPP.

The HPPP is designed to work with the NHS either in a full base load mode or in a HPPP load mode only. In either case, the primary NHS control function for the HPPP is reactor inlet and outlet temperature, relying on a primary flowrate controller and control rod maneuvering.

Given the unique load demands expected for the HPPP, the sudden loss of the process has to be examined. The NHS is expected to be capable of sustaining the sudden loss of process heat load and continue to produce electricity. With the heat transfer to the HPPP installed on a secondary circuit, the addition of a thermal load absorber on the secondary circuit may be necessary.

10.5 NHS/HPPP Inter-Dependencies

To provide maximum schedule flexibility for the HPPP, NHS and HPPP inter-dependencies are to be minimized. Additional details are outside the scope of this report.

11.0 SAFETY

The safety design for any nuclear power plant addresses the selection of the inherent features of the reactor, i.e.,. the choice of a fuel, moderator, and coolant and the basic materials and their design characteristics, the design of the barriers to contain the inventory of radioactive material, and the safety functions that are provided to maintain and protect these barriers. In addition to the inherent reactor features, the safety design includes the design of systems, structures, and components (SSCs) that provide the safety functions and the approaches that are taken to support the capability and reliability of these SSCs. These approaches include the use of passive and active SSCs, and appropriate application of redundancy and diversity.

The AREVA NGNP concept benefits from the experience of the prototype, testing, and demonstration HTRs designed, built, and operated to date with their ceramic-coated fuel particles, graphite moderator, and inert helium coolant, which comprise the inherent characteristics of all modular HTRs. A safety objective of the AREVA NGNP concept is to achieve public safety without needing to implement off-site emergency management measures. To accomplish this with high assurance, the safety design emphasizes radionuclide retention at the source within the fuel particles with minimal reliance on active design features or operator actions.

In the frame of the Generation IV International Forum (GIF), the following safety and reliability goals have been established [6]:

- Generation IV nuclear energy systems operations shall excel in safety and reliability,
- Generation IV nuclear energy systems shall have a very low likelihood and degree of reactor core damage,
- Generation IV nuclear energy systems shall eliminate the need for offsite emergency response.

This section presents the design philosophy, features and functions addressing safety issues of the AREVA NGNP concept. Results from a nuclear safety assessment are outside of the scope of this report; however, a qualitative evaluation of safety characteristics is provided.

11.1 Safety Philosophy

As with all nuclear power plants, safety focuses on the containment of radionuclides through both preventive features and mitigation actions. Consistent with the defense-in-depth tradition, the AREVA NGNP concept addresses safety concerns by incorporating several measures into the system design, fuel product manufacturing, plant operation and control, and emergency planning. This safety philosophy has a profound impact on the design of the AREVA NGNP concept in two important ways. First, the philosophy requires control of radionuclide releases primarily by retention within the coated fuel particles. This leads to important design selections: the type of fuel, the specification of its as-manufactured quality, and its required in-service performance.

Second, the philosophy requires that control of radionuclides be accomplished with judicious reliance on passive systems and minimal reliance on active systems and prompt operator actions. This leads to fundamental design selections such as the core size and geometry, the power density and vessel type. By minimizing the need to rely on active systems or operator actions, the safety strategy becomes a demonstration of the integrity of passive design features. As a consequence, the reliability of active pumps, valves, and their associated services or the reliability of an operator to rapidly take various actions has a much smaller impact on risk measures. In fact, an explicit objective of the AREVA NGNP safety design approach is to provide long response times for operator actions, on the order of several hours or days instead of a few minutes.

An important element of the safety design of the AREVA NGNP is to demonstrate that the principles of defense-in-depth are effectively applied. These include implementation of barrier defense-in-depth, process defense-in-depth, and scenario defense-in-depth principles. The AREVA NGNP uses inherent and passive capabilities to prevent the release of radioactive material from the fuel during both design basis and beyond design basis accident conditions. The overall intent is to provide a simple safety case that provides high confidence that the safety requirements are met.

11.2 Safety Features and Functions

The design philosophy for the AREVA NGNP concept is the application of safety features that provide a high confidence that the plant safety requirements are met with increased use of passive and inherent design characteristics. The key safety issues associated with the AREVA NGNP concept are: 1) radionuclide retention, 2) control of heat generation, 3) control of heat removal, 4) control of chemical attack on fuel particles, and assurance of the reactor system geometry under a spectrum of design basis events that include event scenarios that are initiated by external hazards such airplane crash, internal and flooding, extreme weather conditions and seismic event.

11.2.1 Key Inherent Safety Characteristics

The high level of safety of the AREVA NGNP concept stems from the following design choices:

Helium Coolant - The use of an inert, single phase gaseous coolant eliminates the possibility of a complete loss of coolant event. Pump/circulator cavitations can not occur, and no chemical reaction between the coolant and graphite, or fuel is possible. Furthermore, adequate active core heat removal can be achieved under either pressurized or depressurized primary system conditions.

Graphite Core - The high heat capacity and the low power density of the graphite annular core results in very slow and predictable temperature transients. In addition, the strength of graphite increases with temperature up to levels well above those associated with licensing basis events. Adequate passive core heat removal can be achieved under both pressurized or depressurized primary system conditions.

Ceramic Fuel Particles - The primary fission product barriers in the VHTR fuel are multiple ceramic coatings covering the fuel kernel. These coatings will retain a large fraction of the fission products.

Negative Reactivity Feedback – Reactor core design retains an inherent negative reactivity feedback characteristic. If core temperatures increase, power level decreases. This property ensures that fuel temperature rise is self-limited for loss of heat removal events.

Core Region Structural Configuration – The annular geometry of the core, low power density, and high thermal capacity ensure the cooldown of the shutdown reactor under emergency conditions by passive removal of heat from the reactor vessel using heat emission, conduction, and convection. As a result, fuel and core temperatures remain within allowable limits in all relevant accident scenarios.

In addition, the AREVA NGNP concept has multiple reactor trip mechanisms and decay and residual heat removal systems, including the normal shutdown system, the reserve shutdown system, the secondary nitrogen/helium loop in conjunction with the primary heat transport loop, the SDHRS, the reactor SCS, and the RCCS. Coupled with the inherent safety characteristics, the peak fuel temperature that occurs during the passive residual and decay heat removal condition will always fall below the range of fuel temperatures that would cause significant fuel damage and release of fission products to the primary system.

An added benefit of the inherent characteristic of the AREVA NGNP concept plant is that its benign response which, combined with passive residual heat removal, simplifies the operator's role and provides long time intervals for deliberate actions, thus minimizing the opportunity for operator error.

11.3 Design Provisions

The general safety objectives, relevant for all current and future nuclear plant designs, are:

- To protect individuals, the public and the environment from harm by establishing and maintaining in nuclear installations effective defenses against hazards, in particular but not only, radiological hazards,
- To ensure that in all operational states radiation exposure within the installation is kept below prescribed targets and as low as reasonably achievable (ALARA principle),
- To take all practicable measures to prevent accidents in nuclear installations and to mitigate their consequences, should they occur.

These safety objectives are achieved through the application of the defense-in-depth for ensuring the safety of the public and plant personnel and the general protection of the environment. It consists of an appropriate set of successive provisions to protect the plant staff, the public, and the environment against hazards and in particular radiological ones. Precedent for defense-in-depth strategies can be found from many sources; however, of particular note is the INSAG 10 report by the International Atomic Energy Agency (IAEA)'s International Nuclear Safety Advisory Group [7] . In that report, five specific defense-in-depth levels are identified. These are:

1. Prevention of initiating faults,
2. Control of abnormal events,
3. Mitigation of accidents up to the consideration of multiple failures,
4. Mitigation of severe core damage,
5. Off-site emergency response in case of large radiological releases.

The assessment of design adequacy relative to these safety objectives begins through the identification of design provisions that explicitly address the operating conditions and relevant accident scenarios that potentially challenge safety measures for the AREVA NGNP concept. The following areas are described in this section:

- Barriers to confine fission products
- Control of core heat generation
- Control of heat removal
- Control of chemical attack
- Assurance of reactor geometry

As a design objective, events in the INSAG 10 levels 4 and 5 are not expected to be physically plausible in the AREVA NGNP concept. As such, the consequences from those situations are not considered in the design. Nonetheless, aspects of the plant's design that prevent severe core damage and confine potential radiological releases are discussed in this section.

11.3.1 Radionuclide Containment Features

Maintaining control of radiation and radionuclide retention during normal operation and during a comprehensive set of accident conditions is at the center of the AREVA NGNP safety design. The top safety requirement (consistent with the GEN IV safety objective) is to not disturb the normal day-to-day activities of the public. To accomplish this, there must be high assurance that over a wide spectrum of events the offsite doses are below levels of public concern. The spectrum of events must include a full range of potential accident scenarios that involve:

1. The normal operation activity circulating in the helium, plate-out activity on the metallic parts of the primary circuit and trapped activity within the graphite (fuel blocks, internal structures and dust),
2. Release from fuel particles that do not have intact coatings as a result of the fuel manufacturing process, as a result of either normal operation service or an event,
3. Release from intact fuel particles by diffusion of radionuclides through the coating, as a result of normal operation service or an event and/or,
4. Release of activity from systems other than the primary circuit (e.g., helium purification system, fuel handling system, spent fuel storage).

The AREVA NGNP concept provides a very high confidence in the minimization of the radiological release for any condition. The radionuclide retention is performed by multiple barriers:

Ceramic Coated Fuel Particles With High Temperature Capability Dispersed in Fuel Compacts Within the Prismatic Graphite Fuel Elements

This fuel design effectively confines both solid and gaseous fission products. Engineering development of this fuel compact design has advanced such that large quantities can be manufactured to high quality standards, that is, low levels of defective coated fuel particles and heavy metal contamination. In addition, fuel performance under prototypic conditions (normal operation, events, or accidents) has been assessed to assure that this design is robust under normal operation, such that there is no unacceptable failure of intact particles or unacceptable radionuclide release from as-manufactured defective fuel particles. As such, circulating and plateout activity within the helium pressure boundary during normal operation will be acceptably low. Neutronic characteristics provide limits on both fuel temperature and burn-up that ensure the most effective confinement of fission products within the coated fuel kernel. The graphite of fuel compacts and fuel blocks containing the coated fuel particles also serves as a secondary barrier for solid fission products.

Retention of radionuclides at the source within the fuel assists in meeting not only public safety but personnel safety requirements, resulting in greater equipment access and reduced maintenance times. In addition, retention within the fuel also supports the investment protection goals by minimizing the risk of costly radioactive material clean-up.

Helium Pressure Boundary

The helium pressure boundary encompasses the fuel barrier during power operation. The vessel system components (reactor vessel, power conversion system, and cross duct vessel) are designed to conform to existing codes and standards for nuclear power plant equipment design. The vessel system maintains its service characteristics (including leak-tightness) in all anticipated operating modes including emergencies.

Over-pressurization events are minimized through design by:

- Incorporating safety/relief valves
- Whole pressure boundary designed to vessel code
- Using a single phase coolant gas
- Using compatible fuel, core, moderator, and coolant materials to preclude the potential for chemical reactions

Probabilistic risk analysis will be employed to demonstrate that accidental leaks or breaks in the helium pressure boundary are limited in frequency and size to acceptable levels. In addition, during such accidents, the helium pressure boundary will provide radionuclide retention, additional to that provided by the fuel, by means of natural passive mechanisms including plateout, deposition, and decay.

Below-Grade Reactor Building Barrier

The reactor building will encompass the helium pressure boundary. In the event of leaks or breaks in the helium pressure boundary, the reactor building will have vented dampers to allow helium blowdown and elimination of subsequent radionuclide transport mechanisms. During accidents the reactor building will provide radionuclide retention, additional to that provided by the fuel and helium pressure boundary barriers, by means of natural passive mechanisms including deposition, agglomeration, and decay.

11.3.2 Accident Preventive Features

Accident prevention in the AREVA NGNP concept begins with using simple and reliable systems identified from experience and lessons learned from previous plant design and operation. The inherent safety characteristics greatly simplify the entire plant, which contributes to overall plant safety. Elimination of active equipment reduces maintenance demands and separate controls, simplifying operation. As such, the operator's job is easier, helping to avoid incidents caused by operator error and making it easier for the operator to recover from initiating events.

Resolution of potential safety vulnerabilities through design begins by first designing to eliminate the potential safety vulnerability. If an identified safety vulnerability cannot be eliminated, the risk must be reduced through the use of protective safety design features or devices.

When neither design nor safety-related protection system devices can effectively eliminate identified safety vulnerabilities, monitoring systems are to be used to detect the condition and to produce an adequate warning signal to alert personnel of potential safety concerns, followed by appropriate emergency response procedures.

Maintaining effectiveness of the protective barriers and limiting their damage in all anticipated operating states represent the main task for ensuring safety during the plant operation. Safety systems are included in the design of the AREVA NGNP concept to preserve the function of these protective barriers during all design basis accidents. The overall protective barrier between fission products and the environment is maintained through systems that act to either prevent or mitigate an undesirable plant state.

11.3.2.1 Safety Design for Control of Core Heat Generation

The safety design for the control of heat generation is closely related to the safety design for core heat removal discussed in the next section in that the design must assure an acceptable heat balance in the core so that radionuclide retention in the fuel is achieved. As in the case of the radionuclide barriers, many of the design selections that perform the function of controlling heat generation during accidents are selected for safe plant operation during normal operation and for protection of the plant investment. In contrast to these largely passive engineered barriers to radionuclide release, the design selections for control of heat generation have a blend of active and passive engineered design selections, that is, the protection systems require DC battery power for their sense, command, and execute mission and some DC-power sources are required to prevent the control rods from dropping into the core by gravity.

The primary design selections for control of heat generation include:

- A strong negative temperature coefficient of reactivity
- Startup control rods inserted from the top upon signal from an investment protection system or loss of power
- Operating control rods inserted from the top upon signal from a safety protection system or loss of power
- Reserve shutdown control pellets from top hoppers upon signal from the operator

- High core temperatures sufficient to eliminate Wigner effect contributions

Upon a reactor protection signal, the control rods will drop into the core. The automatic control rod systems are designed with the following characteristics

- redundant protection system logic
- margin for stuck rod(s)
- slow rate of withdrawal limited by motor speed
- physically prevented from fast withdrawal
- do not require AC power or operator insertion
- drop by gravity when power is lost.

The negative temperature coefficient of reactivity and neutronically-inert helium coolant inherently stabilize the heat generation during any situation in acceptable conditions before the occurrence of significant xenon effect. There is a large grace period before unacceptable consequences of xenon effect occur. This allows active operation of the control rod system or the reserve shutdown system. The control rod system and the reserve shutdown system are both capable to shut down the reactor during any condition including xenon effect occurrence. Any situation which could not be mitigated by these provisions is practically eliminated by design.

If neither control rods nor reserve shutdown material are inserted, the temperature coefficient of reactivity will tend to shut down the reactor from any power level following loss of forced convection cooling, such that the RCCS alone can safely cool the core beyond 24 hrs after the initial shutdown.

The Wigner effect is a secondary source of core heat generation and is characterized by the accumulation of energy in the graphite due to irradiation. At graphite temperatures of more than 250° C the Wigner energy is released. As such, if the core graphite or a part of it is irradiated at a low temperature, the Wigner energy is accumulated and its sudden releases might occur leading to core damage. This phenomenon was involved during the Windscale accident in 1958. This can be eliminated if the graphite irradiation occurs at high temperature. In this case, there is a permanent release of the Wigner energy and no significant energy is accumulated in the graphite.

The temperature of the irradiated graphite in the AREVA NGNP concept will be significantly higher than 250°C during any operation with irradiation of graphite; e.g., during normal power operation the coldest helium temperature in the reactor should be about 400°C for the reference plant configuration and 350°C for the option with LWR-steel vessels.

11.3.2.2 Safety Design for Control of Heat Removal after Reactor Shutdown

The safety design for control of heat removal after reactor shutdown complements the safety design for control of core heat generation discussed in the last section so that radionuclide retention in the fuel is achieved. The design selections for control of core heat removal after reactor shutdown are a blend of active and passive engineered design selections, that is, a number of lines of defense are available for heat removal. Those associated with normal operation and plant protection tend to be the more active systems, while that relied on for public safety is intentionally designed to utilize the intrinsic properties of the reactor materials and to exhibit a high degree of passivity that does not require power or prompt operator action.

The primary design selections for control of core heat removal after reactor shutdown include:

- A primary heat transport pathway that utilizes forced helium convection through the reactor to transfer the core sensible and decay heat to the power generation system heat sinks (including the SDHRS)

- An independent, dedicated auxiliary shutdown heat transport pathway that utilizes forced helium convection through the reactor to transfer the core sensible and decay heat to independent heat sinks (i.e., the SCS)
- A third independent diverse reactor cavity heat transport pathway that utilizes conduction, radiation from the reactor core to the RCCS. The failure of the RCCS can be practically eliminated because there is a very large grace period allowing implementation of corrective actions.

Characteristics of the SDHRS and SCS (individually) ensure the possibility to cooldown the reactor to the level of repair temperatures and to maintain it in the required temperature state in shutdown mode and during refueling provided that they preserve their operability. The systems remove heat by forced circulation of coolant in their loops. The SCS is used to cooldown the reactor when the SDHRS is inoperable and its cooling pathway is independent of that for the SDHRS. The SCS can also be used for the cooldown of the reactor in loss-of-power events by switching to the reserve power system (an alternate power source, independent of the emergency power system).

If it is impossible to use either the SDHRS or the SCS, the AREVA NGNP concept can provide cooldown through the RCCS. The heat released in the core is removed by conduction through the reactor vessel and radiation to the RCCS surface cooler, which is placed on the reactor cavity walls. The water circulates in the RCCS due to natural circulation. Within this circuit heated water will arrive in the water storage tank where heat is transmitted to the non-safety recirculation water system. In loss-of-water flow events the heat is removed by means of water evaporation in the heat exchanger and generated steam is dumped into the atmosphere. The low power density and high heat capacity core design characteristics ensures that the core temperature in the course of cooldown remains within the range necessary to prevent significant failure of fuel particles.

The passive nature of RCCS operation during normal operation and in accidents ensures the system's continuous availability and does not require actions of operator or control systems during transition from normal operation to emergency cooldown mode.

All three heat pathways, primary, shutdown, and reactor cavity, will remove core heat with or without pressurized helium coolant.

11.3.2.3 Safety Design for Control of Chemical Attack

Chemical attack has the potential for degrading the primary protective barriers containing reactor fission products. Air and water are the two most relevant species having the potential to enter the system during an accident and then migrate to the core and fuel compacts. Preventive measures have been applied to address the potential hazard of this phenomenon. These measures represent a blend of active and passive design selections which includes:

1. A pressurized inert coolant and fuel, core, and moderator materials that are chemically compatible
2. Silicon carbide coated fuel particles resistant to oxidation embedded in fuel compacts within graphite fuel elements within the reactor graphite reflectors
3. High purity nuclear grade graphite that is resistant to self-sustained oxidation
4. A reliable helium pressure boundary which is adequately designed for limiting the risk and the size of breaks (limitation of the number of connected pipes and limitation of their diameter)
5. A core geometry with a large length-to-diameter ratio for the coolant passages that provides a large resistance to oxidant ingress and transport to the fuel elements
6. A water shutdown cooling system that normally operates at a lower pressure and has a limited volume of water

7. A limited source of air within the reactor building volume
8. A helium purification system that maintains the helium content within the primary coolant system at or below an acceptable level of impurities
9. Isolation devices on the connected pipes and secondary circuits

The simplified design of the AREVA NGNP concept minimizes piping and interfacing systems to reduce the frequency of air and water ingress events. The use of the cross duct vessel rather than piping further reduces the likelihood of a significant primary system loss-of-coolant event. Experimental investigation and analysis of similar HTR designs support the contention that during a primary system loss-of-coolant event, the replacement of helium with air is the result of mass-exchange and diffusion over a long time; that is, air flow is small at the leak location. In addition, gases generated by graphite oxidation prevent the air from entering the core graphite structures, thus moderating the oxidation process. The long response time provided by the small leak rates will allow the graphite to cool. When graphite temperatures drop below 600 °C, the oxidation rate becomes negligible.

In addition, interfacing systems containing air or water are designed with pressures lower than the helium pressure boundary. This includes the SCS system heat exchanger located in the reactor vessel lower head, which represents the most likely source of water that could enter the reactor vessel. However, since there is no obvious mechanism to transport this water source to the core graphite structure, this possibility is not considered plausible.

11.3.2.4 Safety Design for Assurance of Reactor Geometry

The four previous sections discussed the safety design for the functions needed to achieve radionuclide retention within the fuel. The safety design for those functions relies to varying degrees on assurance of reactor geometry, for example, insertion of control rods for control of core heat generation and core and reactor vessel geometry for passive core heat removal, and reactor vessel integrity and reactor geometry for air ingress prevention and mitigation. The passive design selections for assurance of reactor geometry include:

- High temperature reactor internals and core support materials
- High quality vessel system and supports
- Below-grade structurally robust reactor building

The core support function is required to maintain the core geometry. It participates in the heat generation control and decay heat removal functions which could be disturbed in case of core support failure. Special care is taken to ensure that key structural components remain within the applicable design constraints under all required modes. As such, failure of the core support structures will be minimized by engineering the structures to a set of measures defining the highest quality level, considering design rules provided by Codes and Standards, ISI and repair capability for these structures.

11.3.2.5 Application and Site-Specific Safety Design Issues

Beyond the plant application of power generation, the AREVA NGNP concept is designed to provide process heat for the purpose of hydrogen generation. The design of the Hydrogen Production Pilot Plant (HPPP) is independent of the heat source providing process heat. Collocation safety considerations are discussed in Section 11.6. Safety considerations specific to the NGNP for this application are limited to those associated with the NHS.

The unique design consideration accommodating the HPPP is the primary coolant circuit dedicated for this application. It is unique in that it is sized for the smaller amount of energy needed for this application relative to that for power generation, about 60 MWth. It is expected that the performance of equipment and components in this circuit will be of high research and development interest. Frequent inspection, maintenance, and design modification is anticipated.

For this system in which frequent human contact is planned, the key safety concern related to the NGNP NHS is the need to maintain all sources of radiological releases as low as reasonable achievable during all modes of operation. This is achieved by the limitation of the radionuclide content into helium resulting from:

- The fuel characteristics which limit the risk of fuel failure during normal operation
- The fuel fabrication and the fuel fabrication controls which minimize the risk of using failed fuel elements.
- The limitation of helium impurities (i.e., impurities able to be activated).
- The monitoring of the activity into helium. The plant is shut down if abnormal activation is detected.
- The monitoring of any release in the environment during normal operation. If needed, dedicated filtering and isolation systems will be used.

For the indirect cycle concept applied for hydrogen production, the amount of tritium released through the IHX in the non-nuclear facility using the nuclear heat is the primary concern. Compared to other releases during normal operation, tritium in this area is not diluted as it is the case, for example, with the releases at the stack. As such, the IHX leaktightness during normal operation will be ensured with respect to tritium confinement. Indeed, tritium transported in the primary coolant may be released in the secondary coolant due to permeation at IHX level. This permeation phenomenon is driven by the partial pressures on the primary and secondary side and is dependent on the IHX material. The high ability of tritium to diffuse is notably due to its small size. The AREVA NGNP concept will be such that the tritium concentration in the secondary circuits during normal operation is maintained close to the natural value. For the levels of tritium anticipated in the AREVA NGNP concept, this function will be provided by the non-safety helium purification system.

With regard to the protection from HPPP internal faults, leaks resulting in contamination of the HTHTL must be considered. Such leaks, if unmitigated, may have the potential for damaging equipment serving a protective function for the plant such as the IHX. As such, appropriate measures such as a fluid purification system, monitoring instrumentation, and unique PPP control functions will be included in the design.

11.3.3 Accident Protective Actions

In the event of an accident or other uncontrolled plant condition, there are several engineering protection systems which actuate protective trips automatically to maintain reactor and primary system conditions within predetermined limits to protect the integrity of the primary coolant pressure boundary, the core and the fuel particles. These automatic protective trip functions will override manual controls to ensure that the limits are not exceeded. However, the protective trip functions can be manually actuated, should they fail to activate automatically when they are needed and if plant response under automatic plant control is not appropriate or sufficient, the operator can provide backup control with the installed manual override capabilities.

Engineered safety-related protection system controls in the AREVA NGNP concept have been developed considered several principles providing for robust design, including:

1. single failure principle – the failure of any particular component of safety-related protection systems will not undermine the basic function of the protection system
2. independence principle – this requirement is realized through availability of at least two independent instrumentation sets and application in each set of multiple independent channels for each parameter being under control

3. diversity principle – alternative means for performing the same function are to be applied, if available
4. redundancy principle –implementation of multi-channel structure of instrumentation sets
5. safe failure principle – safety-related systems and controls are designed such that the failure of any component providing a safety-related function does not put the system into a diminished condition.
6. Priority principle – improvement of safety functions performance reliability by establishing preference of certain functions over others

With these automated control functions, the operator is able to focus attention on the integrated operation of the plant and on abnormal conditions as they arise.

11.3.3.1 Core Heat Generation Control Function

Two independent safety-related systems are provided to shutdown the reactor and maintain it in a subcritical state. The relevant means are:

- The Reactor Protection System (RPS): leading to insertion by gravity of control rods,
- The Reserve Shutdown System (RSS): leading to the drop of boron carbides balls within dedicated channels inside the core

Conditions resulting in an emergency reactor protection action affecting the neutron control rods might include:

- Current value of power exceeds the RPS setpoint
- Current time of power doubling is less than the RPS setpoint
- Loss of offsite power
- Generation of trip signals on process parameter deviation (i.e., power to helium flow, core inlet/outlet temperature, primary circuit pressure, helium moisture concentration, acitivity in primary equipment rooms, pressure of water in cooling loops, etc.)
- Manual action

Failure of the RPS to insert the control rods into the core provides an automatic signal for the RSS. In addition, the RSS may activate on a manual action.

Additional systems provide control functions to maintain the reactor within the plant's limiting conditions of operation.

11.3.3.2 Heat Removal after Shutdown Function

The function to remove the heat once the reactor is shutdown is identified as decay heat removal function. This includes generally decay heat and part of the heat accumulated before reactor shutdown. There are three systems for heat from the NHS. The relevant means are the following:

- The SDHRS implemented on the secondary circuit: The operation of this system requires the availability of forced helium circulation in the primary circuit and the IHX integrity. For the multiple loop arrangement of the NGNP, the SDHRS is designed to remove heat through 1 or more primary circuit loops; however, the SDHRS is expected to be served by a single heat sink.
- The SCS implemented inside the reactor vessel: This system can operate even if the secondary circuit and the primary forced helium circulation are not available. SCS is designed for achieving this function in pressurized and depressurized conditions. The SCS is made by a circulator and a heat exchanger transferring the decay heat from the core to a water circuit.

- In case of failure of these systems, the decay heat is transferred from the reactor vessel to the RCCS mainly by radiation. The RCCS consists of two independent and redundant trains operating in natural circulation. During any conditions, its function is to maintain acceptable temperature of the reactor cavity concrete and the vessel support devices.

Characteristics of the SDHRS and SCS (individually) ensure the possibility to cooldown the reactor to temperatures considered normal for maintenance and refueling activities. The systems remove heat by forced convection of resident helium in the loops. The SCS is used to cooldown the reactor when the SDHRS is inoperable. The SCS can also be used for the cooldown of the plant in loss-of-power events by switching to the reserve power system (an alternate power source, independent of the emergency power system).

11.3.4 Defense-In-Depth Measures

The requirements established for Generation III plants are already very stringent with regard to the consideration of Defense in Depth and the radiological impact on the public. In particular, the level of the occurrence frequency of large radiological releases is already extremely low.

It is anticipated that the safety improvements for Generation IV plants will concern in particular:

- The confidence in the demonstration of the adequate prevention of severe consequences in accordance with the Defense in Depth philosophy:
    - The identification of initiating events used to confirm the adequacy of the safety provisions shall be exhaustive,
    - The phenomena occurring in any condition shall be simple and well known,
    - The defense against the risks shall be graduated and progressive by implementation of several independent lines of defense,
    - When operator actions are needed, long grace periods shall be provided,
    - The defense against the risks should be balanced: no initiator family should participate in an excessive and unbalanced manner to the global frequency of the plant damage states,
    - The defense should be robust: there are no small deviations in design/plant parameters that could give rise to severely abnormal plant behavior, i.e., “cliff edge effects” are prevented,
    - Bounding degraded situations shall be considered for achieving the exhaustiveness of the faults likely to occur,
    - The complexity of the whole safety architecture as well as the complexity of the operations should be minimized as far as possible,
- The protection of workers against the radiation exposure in particular when implementing the necessary corrective actions in accidental conditions,
- The minimization and mitigation of hazards other than radiological ones (e.g., chemical hazards),
- The minimization of production of wastes and effluents and consideration of their future,

The prevention by design of possible types of malevolence and proliferation, and minimization of their potential consequences.

11.4 Safety Related Structures, Systems, and Components

The key safety features and safety functions of the AREVA NGNP concept have been described. On the basis of preliminary safety assessments, the major systems, structures, and components which are relied upon to perform one or more these safety functions (e.g., ensuring safe shutdown and protection of the primary coolant pressure boundary) or are otherwise relied upon to meet the siting dose criteria have been identified. The set of plant features, classified as safety-related, is comprised of the following:

- Reactor System including neutron control assemblies, ex-vessel neutron detectors, the reactor internals, reactor core, and fuel
- Vessel System including the ASME Section III reactor pressure vessel (structure, not pressure boundary)
- Reactor Cavity Cooling System with exception of the active cooling component on the RCCS secondary circuit
- Reactor Protection System including all sensors, control logic, and housings supporting safety reactor trips
- Fuel storage pools and wells which are part of the Reactor Service Building
- Essential AC and DC power systems

Consistent with the simple yet robust safety design approach employed in the AREVA NGNP concept, only a relatively modest number of systems, structures, and components are seen to be important in ensuring public health and safety. Equally important, this equipment can be seen to reflect the use of passive features in this design. Thus not only is susceptibility to failures in power systems, moving parts, and operator error reduced by the safety systems incorporated into the AREVA NGNP concept, but the operating staff's maintenance burden is minimized.

11.5 Nuclear Safety Assessment

The demonstration of the adequacy of the design with the safety objectives is made through the consideration of four comprehensive lists of events:

- Anticipated Operational Occurrences (AOO): Such events might occur one or several times during the plant life. The mitigation of the consequences of AOO is performed by means of available equipment except if it can be damaged by the initiating fault.
- Design Basis Events (DBE): Events not expected during the lifetime of the plant but might occur in a fleet of plants; nonetheless, their consequences are considered for the design of the plant. Appropriate criteria have to be respected such that the plant is shown to be tolerant to these events. The consequences of multiple failure conditions are also assessed.
- Beyond Design Basis Events (BDBE): These events comprise any DBE combined with the complete failure of any active safety system up to enveloping situations only mitigated by means of inherent behavior and long term corrective actions if necessary. Probabilistic studies are frequently used to establish and check the appropriateness of the foreseen mitigation measures.
- Excluded events: These events are eliminated through design as such their consequences are not considered in the plant design. Probabilistic studies are applied to confirm their expected extreme low frequency.

For all events, the radiological consequences on the public should be such that there is no necessity of protective measures (e.g., sheltering, evacuation) for people living in the vicinity of the plant. Concerning workers, their protection is required if actions are requested and necessary for limiting the consequences of the accident. In this case, the actions have to be performed in an acceptable environment for the operator.

This section assesses the classification of these events as they apply to the AREVA NGNP concept with a particular focus on select DBE that are expected to define the plants limiting conditions of operation.

11.5.1 Anticipated Operational Occurrences

An AOO is classified as an event that might occur one or several times during the plant's operational lifetime. These events correspond to initiating faults combined with equipment failures such that the mean frequency is higher than 10^{-2} per reactor year. The mitigation of the consequences of AOO is performed by means of available equipment except if it can be damaged by the initiating fault.

With respect to availability and investment protection, the plant is expected to be able to return to power shortly after fault rectification.

The safety analysis of their consequences is performed as follows:

- The event occurs during the most penalizing normal operating conditions,
- As an AOO analysis, aggravating failure is not systematically considered.
- Combination of the initiating fault with the failure of a highly reliable mitigating system (including loss-of-offsite power) is analyzed as a DBE.
- Analysis of the consequences is performed with adapted uncertainties.

A preliminary list of AOO adapted to the Pre-Conceptual Design Phase is provided in Table 11-1.

Table 11-1: Preliminary List of AOOs

Initiating Faults Considered as AOO
Reactor trip
Control rod drop
Inadvertent Reserve Shutdown System actuation
Inadvertent control rod withdrawal (the number of control rods, i.e., single or group is depending on the design)
(Reserve Shutdown absorber elements withdrawal is not considered since it does not correspond to normal operation as initial condition)
Inadvertent operation of the Shutdown Cooling System at power
Loss of heat sink:
o Loss of secondary gas circulation: gas turbine failure, spurious isolation valve actuation),
o Loss of steam generator cooling: loss of water/steam circulation (steam turbine, water circulator), loss of condenser cooling, steam/water leakage
Malfunction of the secondary gas system leading to overcooling:
o Increase of secondary gas circulation
o Increase of secondary gas cooling
Malfunction of the helium/steam circuit leading to the increase of heat removal
Steam Generator leak (water inlet to the secondary helium system)
Small leak on the secondary gas circuit
Inadvertent secondary gas safety valve opening
Secondary gas system (primary pressure control, purification, transfer and storage) malfunction
Primary circulator malfunction (leading to increase or decrease of the speed)

Initiating Faults Considered as AOO
Malfunction of Shutdown Cooling System when operating
Malfunction of Secondary Decay Heat Removal System when operating
Malfunction of Reactor Cavity Cooling System
Reactor Cavity Cooling System leakage
Loss of Off-site Power (short duration)
Primary helium system (primary pressure control, purification, transfer and storage) malfunction
Flow blockage in Intermediate Heat eXchanger
Primary helium leakage
Primary helium leak within the Pressure Boundary (e.g., through the hot duct)
Primary helium leakage through the Intermediate Heat eXchanger
Inadvertent Pressure Boundary safety valve opening
Auxiliary (e.g., SCS, primary circulator) heat exchanger leakage
Fuel block loaded into improper position (DBE if not detected)

11.5.2 Design Basis Events

DBEs are not expected to occur during the life of the plant but their consequences are nevertheless analyzed in order to show that their consequences remain tolerable should they occur. These events correspond to initiating faults and AOO combined with equipment failures and are expected to have a frequency with the range of 10^{-2} – 10^{-4} per reactor year. The mitigation of the consequences is performed by considering only safety classified equipment. With respect to availability and investment protection, plant restart is expected following a DBE.

The safety analysis is performed considering the outcome of risk-informed evaluations (i.e., PRA).

A preliminary list of DBE adapted to the Pre-Conceptual Design Phase is provided in Table 11-2.

Table 11-2: Preliminary List of DBE Initiating Events

Initiating Faults leading to DBE
AOO in combination with the failure of a highly reliable mitigating system
Loss of Off-site Power (long duration)
Mechanical failure of the rotating parts of the primary circulator (including shaft seizure)
Single fuel channel blockage
Primary helium pipe break (pipes connected to the vessels, excluding cross vessel)
Large leak on the secondary gas circuit
Several Steam Generator tube ruptures
Auxiliary (e.g., SCS, primary circulator) heat exchanger large leak
Block drop on the core
Load drop during handling

Certain DBEs are particularly important for defining key limiting conditions of operation and the demonstrating the inherent nature of the AREVA NGNP concept to prevent serious core degradation or radiological releases. Among these are:

- Conduction cooldown
- Loss of Heat Sink
- Air Ingress
- Water Ingress
- IHX failure
- Reactivity excursion

These events are highlighted in the following subsections.

11.5.2.1 Loss of Primary Forced Convection – Conduction Cooldown Events

Event Description

In the AREVA NGNP concept, pressurized and depressurized conduction cooldowns are important types of loss-of-primary-forced convection (LOPFC) event. Conduction cooldown with depressurized primary circuit are termed Depressurized Conduction Cooldown (DCC). Conduction cooldown with pressurized primary circuit are termed Pressurized Conduction Cooldown (PCC). The AREVA NGNP concept is designed to accommodate conduction cooldowns by means of inherent characteristics and actions of passive systems. The primary objective is the removal of residual heat in the core. Following a reactor shutdown, the core region includes sensible heat stored in the graphite and heat continuing to be generated by radionuclide decay. The heat has to be removed in order to prevent damage to the fuel, the components inside the reactor vessel and the vessel itself.

The detection of PCC or DCC events leads to reactor shutdown by automatic insertion of the control rods by the RPS. In case of failure of the RPS, core temperature increases due to mass flow rate decreases. Coincidently, the reactor negative temperature coefficient results in decreasing the reactor power. Inherent core shutdown is achieved and power evolution is then governed by residual power.

Forced cooling systems, i.e., either SDHRS or SCS, are normally available to ensure heat removal after reactor shutdown. The SDHRS provides for the primary circuit to function in a manner similar to normal operation. The primary circulator continues to operate and heat is removed via the IHX. The SCS is used when the SDHRS is not available. The residual heat is removed from the core by circulating helium with the shutdown circulator. This heat is taken to the Start-Up/Shutdown System (SUSDS) heat exchanger, where it is rejected to an intermediate dedicated cooling water loop.

Without forced cooling fuel temperatures will rise. Vessel temperature also begins to rise as heat removed is overwhelmed by heat transmitted from the core by conduction, radiation and, for the PCC event only, natural convection. Natural convection in the primary circuit is significant for pressurized conditions and results in increased core heat removal compared to DCC. Heat is evacuated from the reactor primary vessel to the RCCS through radiation and natural convection in the reactor cavity. Temperatures of the reactor vessel support and concrete of the reactor cavity walls are not significantly modified because of operation of the RCCS.

Temperature increase is rather slow (several dozens of hours before reaching peak values) due to high core thermal inertia and low power density. Moreover, due to specific design measures (core annular shape, limited nominal core power, low core power density…), peak temperatures for the fuel and core support structures are limited.

Radiological releases in the primary circuit occur as fuel particle temperatures rise. Even if fuel particles performances remain high, diffusion through intact layers and fuel particles failure rate increase as fuel particle temperature increases. Total radiological release in the primary circuit during the conduction cooldown is the integration of releases on the whole core during the full transient plus the initial radiological content. The particle failure rate in normal operating conditions is maintained very low. A top value is defined and defines the initial conditions. Radiological content trapped in the primary circuit (and possibly in the purification system) with likelihood of release will be taken into account. Because of the role that natural convection can play in heat removal during a PCC event, the radiological impact is expected to be lower than for an equivalent DCC event.

In an effort to minimize any potential radiological releases, several actions may be relied on such as early depressurization in the helium storage tanks through manual or automatic action and the isolation of a leak, in the case of the DCC, or potential leak locations such as the IHX. Further radiological releases outside the primary circuit may occur as long as the core is heated up and as helium thus expands. During this core heat up phase, filtering or isolation capabilities may be beneficial.

Automatic and Manual Actions

Coincident with the detection of a LOPFC event, the following actions are to be taken:

- Heat generation control
    - Control rods insertion (by automatic action) as abnormal parameter value is detected (note that in case of loss of electrical power, control rods drop by gravity in the core).
    - Reserve Shutdown System insertion by operator action, if control rod insertion fails.

Note: As long as the reactor is not tripped, active core cooling with SDHRS or SCS will be prevented.

- Heat removal after reactor shutdown
    - SDHRS or SCS startup (manual since large duration should be available. If necessary, e.g., for reliability purpose, automatic startup may be defined) ensuring heat removal through primary helium forced convection.

      In case of primary circuit depressurization, only SCS is available, but with respect to air ingress concerns, (see Section 11.5.2.3), its startup has to be defined. It could favor higher air ingress rate in the core to limit graphite oxidation by reducing graphite temperatures.
    - RCCS with passive heat removal capacity.
- Isolation and pressure management
    - In case of detection of pressure boundary leakage:

Isolation of the leak is performed,

Isolation valves are closed and IHX pressure managed in case of IHX leak,

Primary circuit depressurization in reservoirs is achieved (not efficient for fast depressurization caused by medium or large leak),

Delayed filtering of releases to the environment is ensured (not possible during fast primary circuit depressurization),

(Possible additional actions due to air ingress concern are not considered here.)

In case of pressurized conduction cooldown, depressurization in reservoirs could be done for limiting the loading on the pressure boundary

11.5.2.2 Loss of Heat Sink

Event Description

The events initiating Loss of Heat Sink (LOHS) are mainly faults located on the secondary circuit or the tertiary circuit. The initial characteristic expected with the LOHS is an increase of primary helium temperatures at the IHX outlets. In the AREVA NGNP concept, the detection of such an abnormal increase leads to a reactor trip and an automatic trip of the primary circulators. Electrical braking is applied to rapidly terminate primary loop circulation. In addition, switching off the power to the primary circulator motor electrical supply will accomplish this objective, subject to the natural coastdown characteristics of the circulators.

Following these events, active core cooling is ensured by the SCS. Among the LOHS initiators, some of them may result in the unavailability of the SDHRS. For such conditions, the SCS becomes the primary residual heat removal system. As with conduction cooldown events, if SCS is unavailable (a BDBE situation), the RCCS ensures passive heat removal.

For AREVA NGNP concept, a highly reliable primary circulator trip function is required for excluding unacceptable consequences. In order to exclude the complete failure, an additional device, independent and diverse, should be implemented for allowing achievement to gain the necessary high reliability function. This device should preferably be based on passive capabilities, which provide significant diversity compared to the automatic trip.

Automatic and Manual Actions

Coincident with the detection of LOHS, the following actions are taken:

- Heat generation control
  - Control rods insertion (by automatic action) as abnormal parameter value is detected (note that in case of combination with loss of electrical power, control rods drop by gravity in the core),
  - Reserve Shutdown System insertion by operator action, if control rod insertion fails.
- Primary circulator trip:
  - Automatic trip:

    Due to the potential for severe loadings, the primary circulator automatic trip system must be:

- Highly reliable.
- Independent of the reactor trip system, in particular, concerning detection.

  - Passive trip:

An additional device (to be confirmed) to stop primary circulator will be implemented. This device has to be diverse with regard to the previous automatic system. This will preferably be performed by the implementation of a passive system, with no common mode with reactor and primary circulator trip systems, operating within a short duration following a LOHS with failure of the automatic primary circulator trip.

- Heat removal after reactor shutdown
  - SDHRS, if not impacted by LOHS initiator,
  - SCS startup if primary circulator is trip and reactor shutdown (manual since large duration should be available).
  - RCCS with passive heat removal capacity in case of SCS failure.

- Isolation and pressure management
  - Isolation valves are closed in case of IHX leak ,
  - Possibility to rely on primary relief valves to limit the increase in pressure (to be assessed).

11.5.2.3 Air Ingress

Event Description

The air ingress event is an oxidation concern for VHTR reactor designs, such as AREVA's NGNP concept. Air ingress into the primary circuit is likely to occur as a consequence of:

- a prior depressurization of the primary circuit or connected helium auxiliary circuits (pipe break(s), safety valve opening),
- a prior depressurization of the secondary gas circuit (pipe break, safety valve opening) combined with the subsequent failure of an IHX,
- air supply due to error during handling operation or due to error in the filling of helium storage tank.

Three oxidation regimes are possible:

- At low temperatures (lower than 500°C), in the "chemical regime", the reactions are so low that oxygen can penetrate the graphite in depth, causing rather uniform attack. In this case, the geometry is not challenged but graphite mechanical properties can potentially be changed.
- At high temperatures (higher than 900°C), in the "boundary layer controlled regime", the chemical reactivity is so high that all oxygen penetrating the laminar sublayer of the gas flowing past the hot graphite surface reacts immediately at the surface. The oxidation attack here causes geometry changes of the graphite body without damaging the material in depth.
- Between these two regimes, in the "in-pore diffusion controlled regime", the gas diffusion in the pore structure of the graphite becomes a reaction rate determining factor. A "corrosion profile" is developing close to the surface of the graphite body, the penetration depth decreasing with increasing temperature.

There are two potential consequences of the oxidation reaction that must be considered: structure changes to the graphite in the core region and reduced fuel particle confinement capability. During an air ingress event, the graphite structure is likely to change and this may have an impact on reactivity coefficients and on absorber rods. In addition, conduction cooldown transients have to consider the possible modification of graphite structures and properties (e.g., conductivity, emissivity) as well as the thermal impact of oxidation reactions. With regard to the fuel particle confinement, this capability relies on several coating layers, including pyrocarbon and SiC layers. The SiC layer withstands oxidation but the confinement capability is weakened by the challenge on the first pyrocarbon layer. Besides, radiological consequences are probably increased due to the release of radio-elements trapped in the graphite.

In addition, at this preconceptual design stage, several other uncertainties must be characterized and resolved, including:

- the influence of conduction cooldown uncertainties,
- the benefit of primary circuit loop isolation strategies,
- the benefit of SCS actuation,
- the influence of air on fuel particles performances as well as on the radio-elements trapped in the graphite blocks,

- the onset of global natural convection and, particularly, the determination of the time when it starts,
- the consequences of CO release,
- the limitation of air available in the pressure boundary cavity by design and possible operator actions.
- For large breaks, the assumptions concerning the shutdown of the reactor and the main circulator have to be assessed in order to evaluate if their failure could drastically increase the consequences. If it is the case, these actions should be performed with a high reliability for practically eliminating their occurrence.

Automatic and Manual Actions

Specific actions with regard to the air ingress event require that current uncertainties associated with this event be resolved. Nonetheless, certain strategic design choices have been incorporated in AREVA's NGNP concept to minimize the potential impact of an air ingress event. The design provisions to prevent the initiator are as follows:

- Minimization and localization of connecting pipes on the pressure boundary
- High quality of the pressure boundary
- Adequate handling procedures around the pressure boundary
- Adequate procedures for primary circuit filling

Regarding the design strategy for the mitigation of the consequences of air ingress, the objective is to show that severe consequences are excluded by means of inherent behavior providing a sufficient grace period allowing implementation of corrective actions which have to be defined (e.g., closure of circuits or vessels system cavity, injection of inert gas). The layout of the circuits (e.g., implementation and diameter of the connected pipes) or reactor building (e.g,. localization of the exhaust devices) is selected to provide the adequate demonstration.

In addition, the reliability and role of the heat removal systems, SDHRS and SCS, during this event has to be defined. An objective to maintain graphite temperatures below 500 °C would allow limiting the consequences of oxidation even if a greater amount of air is introduced in the primary circuit once it is depressurized.

11.5.2.4 Water Ingress

Event Description

The water ingress event presents several safety issues for VHTR reactor designs, such as AREVA's NGNP concept. Water ingress into the primary circuit is likely to occur as a consequence of:

- a leak from a heat exchanger cooling primary helium (e.g., Shutdown Cooling System, cooling system for a primary circulator, operational component cooling system for helium purification),
- a leak from the Steam Generator combined with the failure of an IHX,
- a leak from the heat exchanger cooling the SDHRS combined with the failure of an IHX,
- internal/external flooding.

There are two potential consequences unique to the water ingress event that must be considered: reactivity insertion and combustible gas control. With regard to the impact of a water ingress event on reactivity, the reactivity is expected to increase first with water density and then decreases due to water absorption. In addition, graphite structure is likely to change and this may have an impact on the capability to insert the control rods. Coincidently, the graphite-water reaction at temperatures greater than 500°C impacts the graphite core support structure and generates CO and H_2 that present a combustion risk for the reactor building.

In addition, at this preconceptual design stage, several other uncertainties must be characterized and resolved, including:

- the benefit of start up of the SCS,
- the benefit of primary circuit loop isolation strategies,
- the impact of water on graphite structure and its heat transfer properties,
- the influence of water on fuel particles performances as well on the radio-elements trapped in the graphite blocks,
- the consequences of CO and H2 release,
- the limitation of water available to enter the pressure boundary,
- the impact of possible actuation of safety valve (primary and secondary) on potential radiological releases.

Automatic and Manual Actions

Specific actions with regard to the water ingress event require that current uncertainties associated with this event be resolved. Nonetheless, certain strategic design choices have been incorporated in AREVA's NGNP concept to minimize the potential impact of an water ingress event. The design provisions to prevent the initiator are as follows:

- Minimization and localization of connecting pipes on the pressure boundary
- High quality of the pressure boundary
- Adequate handling procedures around the pressure boundary

Regarding the design strategy for the mitigation of the consequences of water ingress, the objective is to show that severe consequences are prevented by means of inherent behavior providing a sufficient grace period for implementing corrective actions (e.g., insertion of absorbing elements, draining of water circuits). The layout of the circuits (e.g., volume, level) and their operating conditions (e.g., SCS cooling circuit pressure below primary helium nominal pressure) are selected in order to provide the adequate demonstration.

Additional measures include:

- such as closure of isolation valves (e.g., IHX, steam generator) allow to minimize the risk of water ingress into the core.

The absorber elements are designed so as to be capable to withstand water ingress reactivity consequences during any condition (including cold shutdown conditions).

11.5.2.5 IHX Failure

Event Description

As an indirect cycle design, nuclear heat generated in the reactor core of AREVA's NGNP concept is transmitted to the power conversion or process heat system via an IHX. Failure of the IHX is any breach of the physical boundary between the primary and secondary circuits. AREVA's NGNP concept is designed with zero pressure differential across the pressure boundary common to both the primary and secondary circuits. As such, fluid exchange between the circuits following an IHX failure is driven by momentum and diffusion phenomena, rather than pressure.

The main safety issue is the confinement of radiological content. The AREVA's NGNP concept employs two separate IHX designs (i.e., plate-type IHX supporting the HPPP process heat application and tube-type IHX supporting power generation). While the likelihood of failure is considered to be smaller with the tube-type design, the radiological consequences of an IHX failure are independent of IHX-type. The primary defense against a radiological release is the maintenance of low activity in the primary circuit. The key factors that contribute to a low activity in the primary circuit during normal operation are:

- The low level of radioactivity in the primary circuit. This results from the characteristics of the coated particle concept for a very low failure rate in normal operation. Additionally, the required quality of helium, provided by the Helium Purification System, and graphite should further limit their activation. The activity of the primary helium will be monitored and operation with higher failure rate will not be allowed.
- The slow and limited evolution of the fuel temperature during any accident, thus excluding the possibility of a significant increase of the fuel failure rate during the first hours following any accident.

These characteristics are based on the previous HTR experience (e.g., the German experience). They are considered as design requirements for the AREVA NGNP concept.

With the expectation that the IHX failure is limited to a single IHX, heat removal can continue through the SDHRS associated with the intact loops. The loop containing the failed IHX can be isolated by a trip of the primary circulator associated with the effected IHX and closure of isolation valves on the secondary circuit surrounding the effected IHX. In the event that the SDHRS is not available, the SCS will be responsible for removing decay heat. As with the conduction cooldown events, if the failure of the SDHRS is further combined with the failure of the SCS, the decay heat is removed by radial conduction and radiation from the core to the reactor vessel and then, to the Reactor Cavity Cooling System (RCCS) by radiation.

Automatic and Manual Actions

IHX failure detection is achieved by activity detection in the secondary side of the IHX combined with an overpressure of the primary helium compared to the secondary gas during normal operation. In case of IHX failure detection, the following actions are to be taken:

- Heat generation control
    - Control rods insertion (by automatic action) as abnormal parameter value is detected (note that in case of combination with loss of electrical power, control rods drop by gravity in the core),
    - Reserve Shutdown System insertion by operator action, if control rod insertion fails.
- IHX isolation valves are closed. Isolation valves are implemented on the secondary circuit, close to the IHX vessel.
- Automatic primary circulator trip on effected loop or, if the effected loop cannot be identified, on all loops.
- Heat removal after reactor shutdown
    - SDHRS, if the effected IHX can be isolated,
    - SCS startup ensuring decay heat removal through primary helium forced convection. This start-up should be manual since large duration should be available. (If necessary, e.g., for reliability purpose, automatic startup may be defined).
    - RCCS with passive heat removal capacity.

Efforts to depressurize of the primary system and, thus, further minimize leakage to the secondary will also be considered.

11.5.2.6 Reactivity Excursion

Event Description

In normal operating conditions, the fission reactions in the core are kept in a stable critical state which corresponds to zero total reactivity. The reactor negative temperature coefficient stabilizes the critical state. The control rods allow achievement of the requested reactor power. In case of a reactivity insertion event, AREVA NGNP concept is designed to accommodate reactivity insertion events by means of inherent characteristics.

The detection of reactivity insertion events leads to reactor shutdown by automatic insertion of the control rods by the RPS. A second system, the RSS can also achieve the function. RSS is manually actuated. The two neutron absorbing systems are designed so that the insertion of at least one these systems ensures and maintains sub-criticality in any conditions. This includes in particular the reactivity due to core cooling down to the coldest shutdown state combined with the xenon effect and the reactivity insertion due to the initiating event.

If the reactivity insertion and the reactivity insertion speed are limited and if the reactor is not shut down, the situation is potentially controllable though power, fuel temperature and helium temperature should rise. In particular, as power increases, fuel temperature rises rapidly and, due to Doppler effect, results in negative reactivity feedback. Heating of the graphite moderator and most of all of the reflectors occurs more slowly, and, as a consequence, the associated temperature feedbacks come relatively later. Thus, Doppler-effect only in a first time then moderator negative temperature coefficient ensure limitation of generated power through reactivity feedbacks. For longer term, if the reactor is not shut down, other reactivity effects occur due to thermal expansion or contraction of the core, the structures and the control rods. If absorber elements are not inserted, the xenon effect could make the reactor critical again about several dozens of hours after the beginning of the accident.

Even if the fault initiating the reactivity insertion is a failure of the pressure boundary in the area surrounding control rod (e.g., it the initiating event is a control rod ejection), the shutdown systems ability to perform their function is maintained.

Following detection of reactivity insertion events and the reactor is shutdown, active core cooling is provided by the SDHRS or the SCS. As with conduction cooldown events, if the SDHRS and SCS are unavailable, the RCCS ensures passive heat removal.

Radiological releases in the primary circuit occur as fuel particle temperature rises. Even if fuel particle performance remains high, diffusion through intact layers and fuel particles failure rate increase as fuel particle temperature increases.

Locally, the loss of fuel particles' high radiological contaminant retention capability could be tolerated; acknowledging that the primary goal is compliance with the global radiological criteria for such accidental situations.

Automatic and Manual Actions

Coincident with reactivity insertion events, the following actions are to be taken:

- Heat generation control
  - Control rods insertion (by automatic action) as abnormal parameter value is detected (note that in case of loss of electrical power, control rods drop by gravity in the core).
  - Reserve Shutdown System insertion by operator action as parameter value is detected.
- Main primary circulator automatic trip (and reduction of IHX heat removal due to turbine trip).
- Heat removal after reactor shutdown:

  - o SDHRS or SCS startup ensuring decay heat removal through primary helium forced convection. This start-up should be manual since large duration should be available. (If necessary, e.g., for reliability purpose, automatic startup may be defined).
  - o RCCS with passive heat removal capacity.
- Isolation and pressure management
  - o In case of increase of the primary radiological content,

Primary helium will be purified by the helium purification system,

Primary circuit discharge in helium storage tanks could be done.

11.5.3 Beyond Design Basis Events

BDBE events comprise DBE with the complete failure of any active system up to enveloping situations only mitigated by means of inherent behavior and long term corrective actions, if necessary. BDBE can involve additional design specific limiting events, studied in order to prove that no cliff edge effect occurs (e.g., simultaneous failure of several pipes). A BDBE is expected to have a frequency with the range of 10^{-4} – 10^{-7} per reactor year.

The radiological consequences on the public should be limited: there should be at least no necessity of protective measures (e.g., sheltering, evacuation) for people living in the vicinity of the plant.

Concerning workers, their protection is required only if operator actions are requested and necessary. In this case, the actions have to be performed in acceptable environment for operator.

The safety analysis is performed as follows:

- The event occurs during the most penalizing normal operating conditions,
- No consideration of aggravating failure,
- Combination with loss of offsite power,
- Best estimate analysis, but assessment that no cliff edge effect occurs.

With regard to the core heat generation control function, the enveloping situations to be analyzed correspond to the combination of any accident with the failure at short term to insert neutron absorbing element. The design objective is to provide sufficient grace period for convincingly proving that at least one of the shutdown systems can be operated. During the grace period, the generated core heat decreases due to the negative temperature coefficient. The operation of shutdown system allows avoiding reactivity consequences due to xenon effect combined with core cooldown. In addition, design provisions have to be implemented in order to justify the exclusion of large reactivity insertion accidents which occur before sufficient grace period for implementing corrective actions (e.g., large and quick water ingress, large control rod ejection).

For the decay heat removal function, the enveloping situation to be analyzed corresponds to the combination of any accident with the failure of the SDHRS and the SCS. The decay heat is then removed by conduction and radiation within the reactor vessel, radiation from the reactor vessel to the RCCS. The redundant, passive operation and the large grace period offered for repair practically eliminates the long duration failure of RCCS.

If a degradation of the core (e.g., due to not detected manufacturing defect on a large number of fuel particles) is postulated during normal operation, the failure of the fuel particles is expected to be progressive. The surveillance of the activity in the primary circuit will allow the detection of abnormal radiological releases and a sufficient grace period is provided to implement corrective actions (e.g., reactor shutdown, helium treatment and core unloading). The justification of the progressive degradation of the core relies on the very large number of fuel particles; consequently, the weakest particles are first to fail. With a spectrum of local conditions (e.g., temperature, burn up, location), these failures will not occur simultaneously.

The monitoring of activity during normal operation allows an operator to anticipate any failure likely to occur during accidental conditions: due to the large number of fuel particles, it is not credible to consider a significant number of failures without prior detection during normal operating conditions. Nevertheless, if a degradation of the core would occur during accidental conditions, the following arguments are valuable:

- For slow temperature transients such as conduction cooldown, if the measured activity is beyond the enveloping value determined by prior studies, the strategy is the following depending on the primary circuit conditions:
  - Primary circuit is depressurized: thermal transient is slow and corrective actions regarding confinement function by means of the last barrier (reactor building) are possible,
  - Primary circuit is pressurized: thermal transient is slow and an approach similar to the one implemented when the damage occurs in normal operation is implemented. Confinement function is performed by means of the second barrier (primary circuit),
  - During primary circuit depressurization: thermal transient is slow. A test regarding the behavior of the fuel particle in case of rapid depressurization (fast increase of the differential pressure on the fuel particle layers) should be performed to show there is no cliff edge effect. Additional filtration capability (sand filters) would allow to reduce the radiological releases.

For fast temperature transients such as some reactivity accidents, the thermal consequences on the fuel particles should be limited by design (no severe temperature increase or severe temperature increase limited to a small part of the core). Tests regarding the behavior of the fuel particle should be performed to show there is no cliff edge effect. Concerning the mitigation of the radiological consequences, the availability of the remaining confinement barriers is considered.

11.6 Collocation Considerations

This section discusses some of the safety considerations associated with collocating AREVA’s NGNP prototype with a HPPP. Similar considerations would also be relevant to the use of a commercial VHTR as process heat facility to support a chemical or petrochemical facility. However, the focus of the discussions in this section is on collocation with hydrogen production as this is a functional and design requirement for the NGNP prototype. Furthermore, it is expected that hydrogen production demonstrations of two processes with be part of NGNP: (1) the S-I thermochemical process, and (2) HTE.

In this regard, the hydrogen facility may be viewed as both a chemical plant with its own operational issues that processes and produces large quantities of hazardous materials, as well as a facility that is integrated into the design and operation of the NGNP (services, receipt and return of heat transfer media, common controls, and physical proximity). In accordance with the guidance and requirements of the NGNP High Level Functional Requirements [5], industry codes and standards will apply to the hydrogen facility. Some of the key aspects of the design that will affect safety include:

- Distance between the facilities
- Use of common control facilities
- Common support services (ES&H, security, operations, O&M, management, support staff, QA/QC, licensing, maintenance).

The HPPP will not only involve the conversion of feedstock (water) to hydrogen and oxygen, but inherently the limited storage and distribution of the products (e.g., hydrogen and oxygen). There are many organizations that have issued codes and standards, as well as guidelines for safe production and storage of these gases. These include NFPA (e.g., Standards 50, 69 and 70), NEC, ASME Section VIII, ANSI, ASME B31.1 and B31.8. About twenty organizations have issued other guidelines for the generation, storage and use of hydrogen including

- DOE (Department of Energy),
- DOT (Department of Transportation)
- NHA (National Hydrogen Association)
- ASME (American Society of Mechanical Engineers)
- CSB (US Chemical Safety and hazards Investigation Board)
- NREL (National Renewable Energy Laboratory)
- GRI (Gas Technology Institute)
- UL (Underwriters Laboratory)
- CGA (Compressed GasAssociation)
- API (American Petrolium Institude)
- OSHA (Occupational Safety and Hazard Association)

Federal regulations are prescribed under the following code sections:

- 29 CFR 1910.103 Occupational Safety and Health Standards, Subpart H – Hazardous materials, Hydrogen.
- 29 CFR 1910.104, Oxygen.

11.6.1 Hydrogen Flammability

While hydrogen is an extremely flammable substance, experience suggests that unconfined explosion events are likely to produce on deflagration pressures. An example of the type of pressure wave that could be produced by a moderate size hydrogen leak is shown in Figure 11-1. This curve of pressure versus distance is based on Hopkinson-Cranz Z scaling ($Z=x / (lbs\ TNT)^{1/3}$) with the energy released by the hydrogen is scaled to an equivalent TNT mass. The leak is assumed to be 1200 ft^3 at 20 Bar with an energy equivalence of 1000 lbs TNT. The dashed line in the figure shows a potentially permissible blast loading on SSCs at NGNP. This would suggest that the distance between the facilities could be less than 500 feet and perhaps only a few hundred feet.

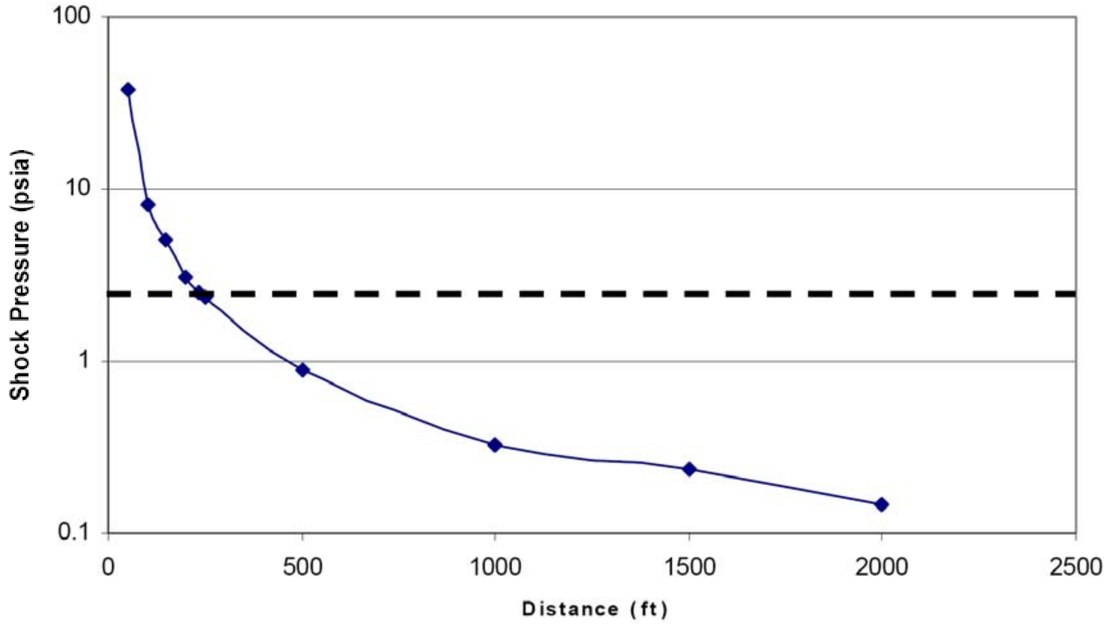


Figure 11-1 Pressure Wave from Nominal Unconfined Hydrogen Explosion

Other explosion analyses also show that while not insignificant, hydrogen explosion events can be accommodated in the NGNP design. For instance, while Figure 11-1 is based on essentially a point source explosion and analysis of detonable cloud after a hydrogen pipeline rupture (deflagration explosion of a 75 ft radius spherical cloud) produces only a 0.3 psi overpressure at 400 feet. Generation of missiles must also be considered – with industrial hydrogen explosions, large section missiles can be shown to generally travel less than 100 feet.

Figure 11-2 also shows a prediction of the pressure waves caused by the ignition of 120 pounds of hydrogen gas (about 1200 ft^3 at 20 Bar) through a 2-inch square in leak from a 12 inch pipe, 1500 feet long. As shown in this figure, the results (using the EPA ALOHA Code) are similar to those shown in Figure 11-1 which is based on TNT equivalent.

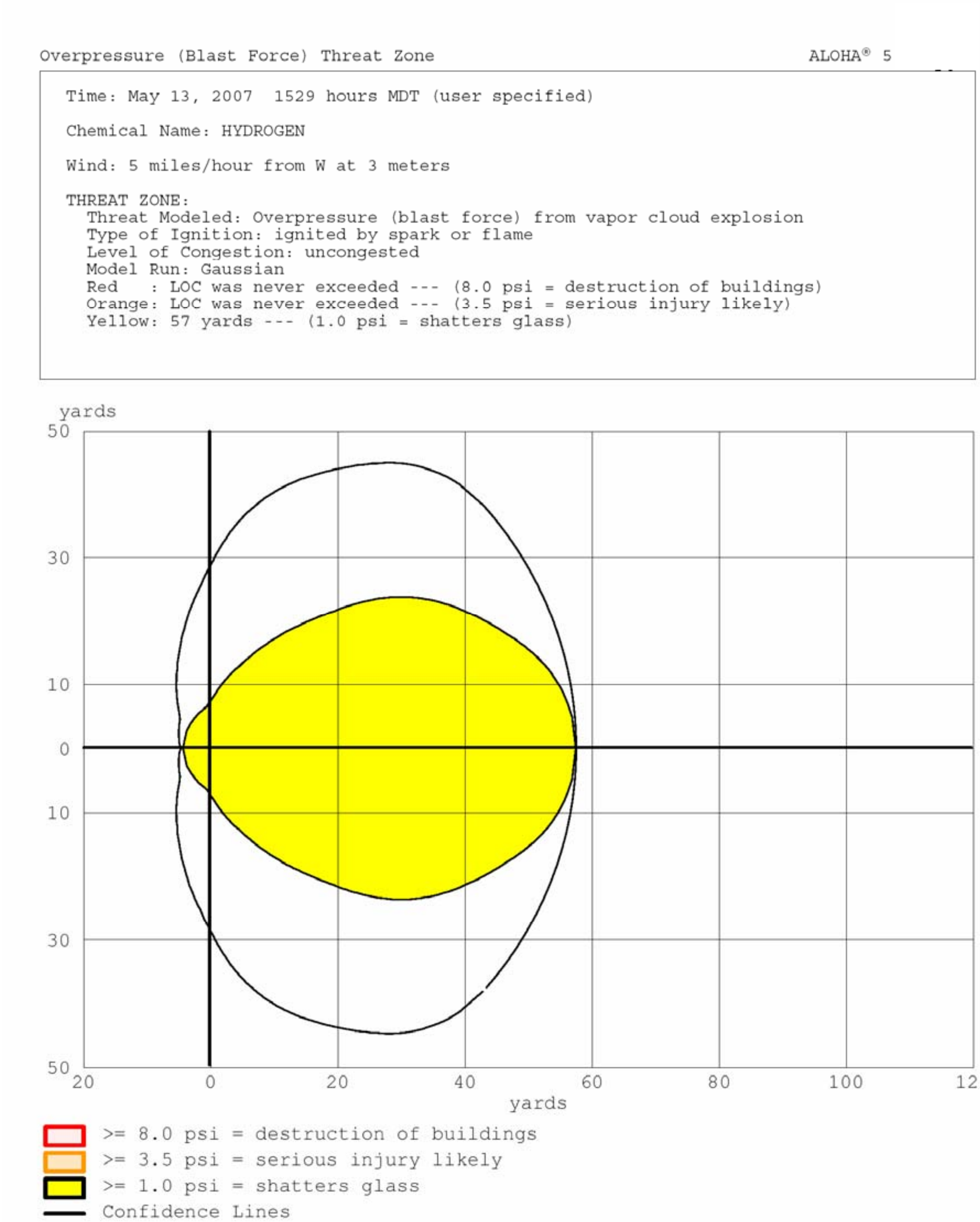


Figure 11-2 Pressure Waves from a Hydrogen Leak and Explosion

11.6.2 Other Chemical Hazards

Properties and issues associated with other chemicals in the HPPP include:

- Helium - asphixiant

- SO_2 - toxic (15 ppm)
- SO_3 – toxic (30 mg/m3)
- Iodine – slightly toxic (5 ppm)
- Sulfuric Acid - toxic in liquid or gaseous form (30 mg/m3)
- HI - toxic
- Oxygen - increases hydrogen explosion risk

Figure 11-3 shows an example of the chemical hazard associated with a leak of sulfur dioxide from a high pressure, elevated section of the sulfuric acid decomposer which could be used for either the thermochemical S-I process or for HTE. Note that the Emergency Response Planning Guide (ERPG) 3 limit exists at about ¼ mile from the origin of the leak.

The particular release modeled in this case was leakage of 185 pounds of SO_2 from a 6 inch pipe at 70 bar through a 28 square inch opening. The initial temperature of the gas was 800°C.

Toxic Threat Zone

```
Time: May 13, 2007  1529 hours MDT (user specified)

Chemical Name: SULFUR DIOXIDE

Wind: 5 miles/hour from W at 3 meters

THREAT ZONE:
  Model Run: Gaussian
  Red    : 537 yards --- (15 ppm = ERPG-3)
  Orange: 978 yards --- (3 ppm = ERPG-2)
  Yellow: 1.2 miles --- (0.3 ppm = ERPG-1)
```

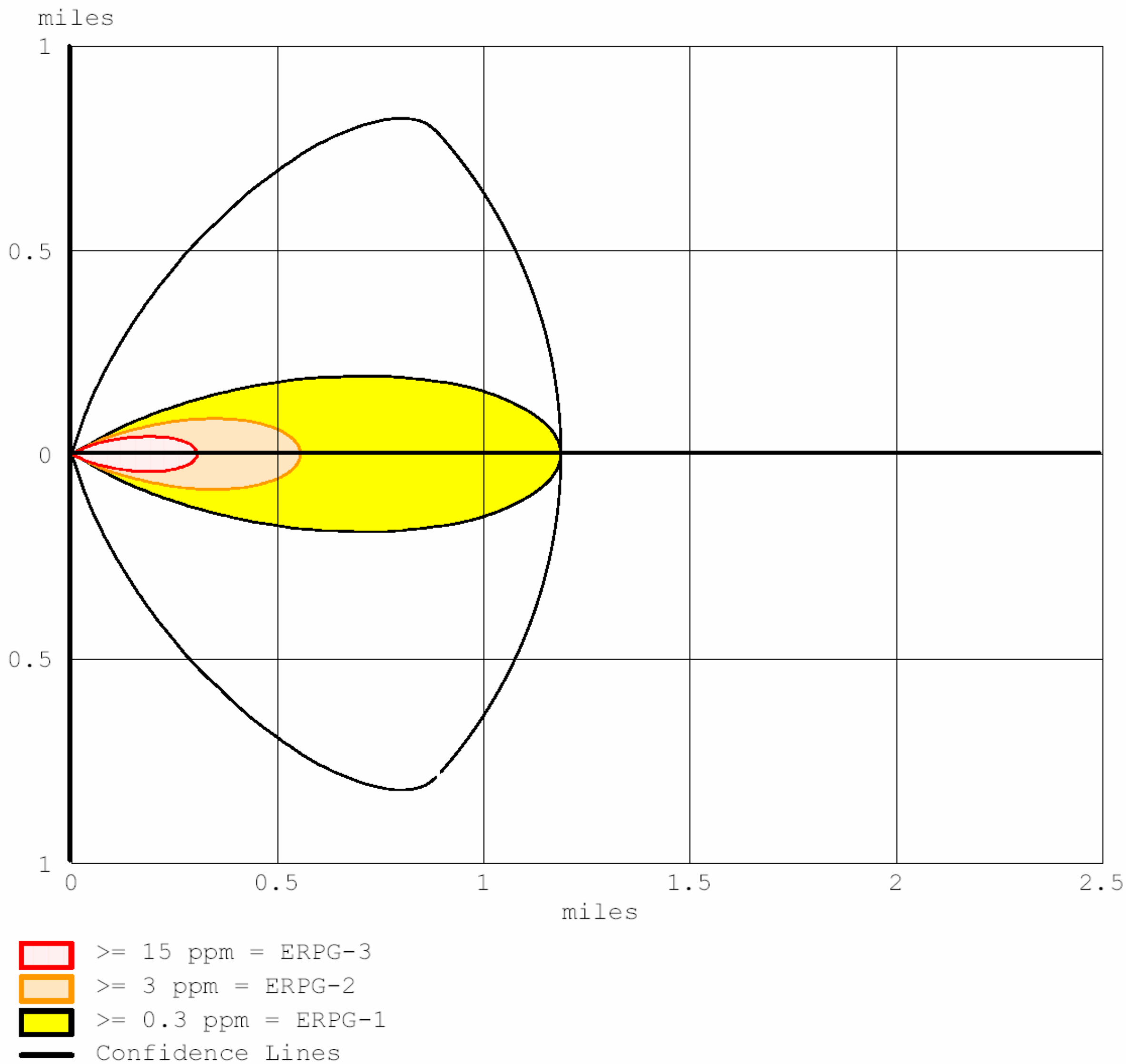


Figure 11-3 Sulfur Dioxide Dispersion after Leak

11.6.3 HPPP Design and Siting Considerations

While most of the previous discussion has focused on the implications of collocating the NGNP near the HPPP, and the effects on the nuclear island, there are also hydrogen plant design and siting considerations. Guidelines for design and siting chemical and petrochemical facilities have been developed by organizations such as the American Institute of Chemical Engineers (AIChE) [9]. These guidelines discuss issues such as: (1) process design options to maximize safety, (2) separation distance requirements between complexes, sites and unit operations, (3) safety and process devices, instruments, alarms and controls, and (4) administrative processes and controls. Methods for hazards screening, hazard analysis, toxic release scenarios, fire and explosion scenarios, and environmental regulatory compliance are discussed. Use of these types of guidelines will be an important element in the design phase of the hydrogen production facilities at NGNP.

Waterford 3 Example

While the licensing and collocation issues for NGNP may appear challenging, there is significant amount of precedence in licensing nuclear facilities near petrochemical and chemical production facilities and pipelines. For example, the Waterford Unit 3 plant near New Orleans, Louisiana has 15 industrial facilities less than 5 miles from plant. These include:

- A facility producing 1.2 billion pounds chlorine per year - 4200 feet from the plant
- A chemical facility (variety of products) producing 2.5 billion pounds chemicals per year - 6000 feet from the plant
- A petrochemical refiner treating 235,000 bbls oil per day including a SMR reformer - 2.5 miles from the plant
- An ammonia plant fed by hydrogen pipeline carrying 120,000 pounds H_2 per day - less than 1.2 miles away
- A facility producing 20 million pounds H_2SO_4 per year which stores 24,000 gallons at one time – located less than 1 mile from the plant.

There is also a large 12 inch diameter hydrogen pipeline 650 psig pipeline. Rupture and explosion of escaped gas from this pipeline was included in USFAR and not found to be a bounding event.

12.0 LICENSING AND PERMITTING

The NGNP must obtain a license from the NRC in accordance with the requirements of 10 CFR Part 50 or 10 CFR Part 52. NRC licensure is a key requirement specified in the NGNP Functions and Requirements Report [8]. Furthermore, as a first-of-a-kind non-light-water reactor licensed as a demonstration plant for both electricity production and hydrogen generation, the NGNP will serve as the licensing basis for the future commercialization of similarly designed plants. Therefore, the recommended licensing strategy will have the ultimate goal of supporting the application for and receipt of a design certification for the future commercial application of the high temperature reactor technologies.

The current licensing requirements set forth in Title 10 of the Code of Federal Regulations (10CFR) [10] were developed for light water reactors (LWRs). The United States (U.S.) Nuclear Regulatory Commission (NRC) issued operating licenses for medium-temperature gas reactors (HTGRs) at Peach Bottom 1 and Ft. St. Vrain and has reviewed several other HTGR designs, namely the Modular High Temperature Gas Reactor (MHTGR), Gas-Turbine, Modular Helium Reactor (GT-MHR), and Pebble Bed Modular Reactor (PBMR). These pre-application reviews were based on a combination of interpretations of and exceptions to these LWR-based requirements and did not benefit from important developments in NRC policies regarding safety goals, risk-informed and performance-based regulation, and expectations for the safety of advanced reactors.

The NRC's Advanced Reactor Policy [11] states that for advanced reactors the Commission expects, as a minimum, the same degree of protection of the public and the environment that is required for current generation LWRs. Thus, the Commission expects that advanced reactor designs will comply with the Commission's safety goal policy statement [12].

Furthermore, in their staff requirements memorandum (SRM) to SECY 03-0047[13] [14], the NRC clarified its expectations that advanced non-LWRs should be held to account for the same levels of enhanced safety as established for advanced LWRs whose designs have been certified. In these policy statements, the Commission has made clear its expectation that advanced reactors will provide enhanced margins to safety while utilizing simplified, inherent, passive, and other innovative means to accomplish their safety functions.

12.1 Licensing Regulations

AREVA proposes a top-down, safety-focused method to create a new set of regulatory requirements for NGNP using a blend of deterministic and risk-informed techniques. This approach builds on the experience in developing a risk-informed licensing process that was developed by DOE for the MHTGR [15] [16] and by Exelon for the PBMR [17][18], and will closely follow the risk-informed technology neutral licensing process being developed by the NRC staff in response to the SRM for SECY 03-0047.

In developing this process, it has been recognized by AREVA, the nuclear power industry, and the NRC that the current regulations include some rules that are fully applicable to any design, some that are not applicable to gas cooled reactors and many may be partially applicable (literally or for analogous reasons). There may also be some features of the NGNP design that cannot be addressed by any current regulatory document, thus requiring some unique requirements, and likely some new guidance documents to be crafted to interpret existing requirements. This risk-informed process is graphically represented in Figure 12-1.

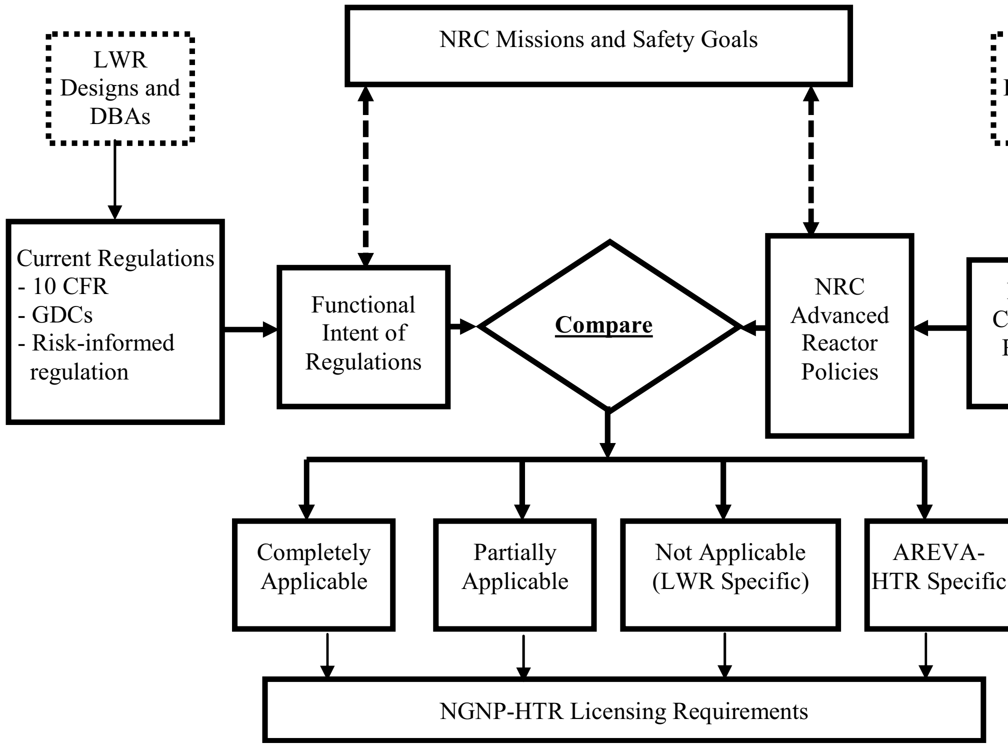


Figure 12-1: Risk-informed Process for Defining AREVA-HTR Licensing Requirements

AREVA's proposed approach to the acceptance and licensing of the NGNP has a clear link between NRC's regulatory missions and the specific regulatory requirements to be applied to the design. Figure 12-1displays: (1) the NRC mission, (2) the public safety objective for nuclear power plants as contained in 10CFR, general design criteria (GDC), and recent enhancements to the regulatory process through application of risk insights and use of the Safety Goal Policy to set forth expectations for the safety of advanced reactors.

Based on these fundamental objectives, a top-down licensing approach for the NGNP is proposed that consists of the following elements:

1. An evaluation of the current regulations will be performed to yield top level regulatory criteria (TLRC) that support the safety goals as well as meet deterministic requirements for licensing basis events.
2. A full-scope probabilistic risk assessment (PRA) will be performed to guide the selection of an appropriate set of licensing basis events that reflect the unique and specific design elements and safety design philosophy of the NGNP reactor.
3. A combination of probabilistic and deterministic approaches will be used to guide the selection of safety-related SSCs in the context of reactor-specific safety functions that will be defined for the licensing basis events (LBEs).
4. A review of existing requirements for LWRs will be performed to capture important deterministic criteria such as those motivated by the defense-in-depth philosophy. Such criteria will be applied to the NGNP allowing for differences in design and safety design philosophy.
5. Certain regulatory objectives are not amenable to probabilistic treatment in the present regulatory environment. These include occupational exposure minimization, environmental impacts other than radiological, and security and safeguards. These objectives will be met in the conventional manner as consistent with existing practice.

Commercialization

The licensing strategy for the commercialization of the NGNP is based on the NRC Design Certification Process provided in 10CFR52 [19].

Experience from the operation and prototype testing of the demonstration NGNP will be needed to support the design certification of future commercial versions. Therefore, the NGNP will be licensed with the conventional two step 10CFR50 licensing process:

1. Step 1 is a construction permit based on the review of a preliminary safety analysis report (PSAR);
2. Step 2 is an operating license based on the review of a final safety analysis report (FSAR).

The commercial plant will utilize the design and safety insights from the demonstration plant testing and operation in its 10CFR52 design certification. Pre-application interactions with the NRC will be necessary to develop technology neutral and HTR-specific licensing bases for the demonstration and commercial plants. Figure 12-2 illustrates the close coordination required between the licensing of the NGNP and the commercial HTR plants.

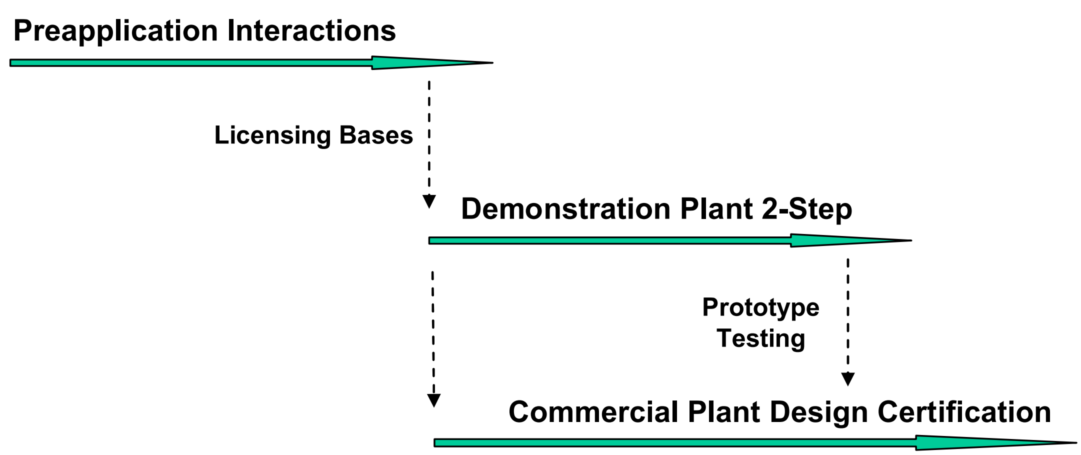


Figure 12-2: Major Phases of Commercial HTR Licensing

The safety design and licensing approach for the NGNP will follow the guidance of the NRC's Advanced Reactor Policy as amended with the recent SRM on SECY-2003-047 [14]. This SRM states that for advanced non light water reactors, the Commission expects to approve a level of enhanced safety through a process similar to that used in the evolutionary LWR and advanced light-water reactor (ALWR) design certification reviews. The Commission expects that advanced reactor designs will comply with the Commission's safety goal policy statement. Furthermore, consistent with the Advanced Reactor Policy Statement, the Commission expects that advanced reactors will provide enhanced margins to safety and/or utilize simplified, inherent, passive, or other innovative means to accomplish their safety function relative to currently licensed LWRs [11].

12.2 Applicability of Existing Regulations

Essential to future commercialization of HTR technology is the evolution of NGNP licensing into a predictable and stable regulatory environment. However, given the fact that the current licensing requirements set forth in 10CFR were developed for LWRs and that there is very limited regulatory experience with gas cooled reactor technology, there is no set of regulations directly applicable to HTR technology.

The NRC has issued licenses for two HTGRs at Peach Bottom 1 and Ft. St. Vrain and has reviewed several other HTGR designs; however, these license reviews were based on a combination of interpretations of and exceptions to LWR requirements. Such a license-by-exception approach is not compatible with the goals of achieving a predictable and stable regulatory environment for the AREVA-HTR. Consequently, for a license application to be prepared, a new set of regulations, regulatory guides and standard review plans will have to be crafted out of the existing regulatory body to guide both the applicants and NRC in their review.

Screening of Existing Regulations

The first step of the above process is to perform a screening of the existing regulations and supporting documents for applicability to the NGNP. Subsequent steps as described in the previous section will follow. By determining which existing regulations do not apply, wholly or in part, to the NGNP HTR design and which new and unique regulations may be needed, both applicant and the NRC will have a good understanding of how to navigate through the legal and procedural steps to obtaining exemptions or other suitable relief to existing regulations geared to light water reactors.

Applicable Regulatory Documents

The scope of regulatory documents that are currently available to support nuclear power plant licensing includes high level documents such as:

- 10CFR 50, including GDC
- Policy Issues (SECY Papers)
- Staff Requirements Memoranda

Additionally, lower tier documents apply as well, such as:

- NRC Staff Technical Reports (NUREG)
- Topical Reports
- Regulatory Guides
- Standard Review Plans
- Branch Technical Positions
- Industry Codes and Standards
- International Regulatory Guides and Documents

Each of these existing regulatory documents shares the common feature of having been developed primarily for currently licensed light water reactor designs and will have to be reviewed and analyzed for applicability to the NGNP.

In the Exelon PBMR licensing approach [17] and the MHTGR PSID [15] an initial preliminary screening was performed for selected high level regulatory documents in the context of those specific reactor designs.

AREVA, in its work on ANTARES, initiated the development of a regulatory document set for HTRs. The above mentioned preliminary screenings were reviewed for applicability; and, independently, a separate screening of selected high level documents was performed by AREVA personnel with expertise in reactor licensing and HTR design and safety design philosophy.

Regulatory Document Screening Process

The approach AREVA followed was to assemble a panel of experts to perform a screening of selected requirements in the current body of regulations into one of the following categories: applicable, partially applicable, and not applicable. The scope of regulations to be considered in this screening include the entirety of 10CFR50, "Domestic Licensing of Production and Utilization Facilities," including the GDC (10CFR50, Appendix A), all other appendices to 10CFR50, plus other selected parts of 10CFR. The logic that followed in the screening is similar to that developed during the screening for the Exelon PBMR submittal and is presented in Figure 12-3.

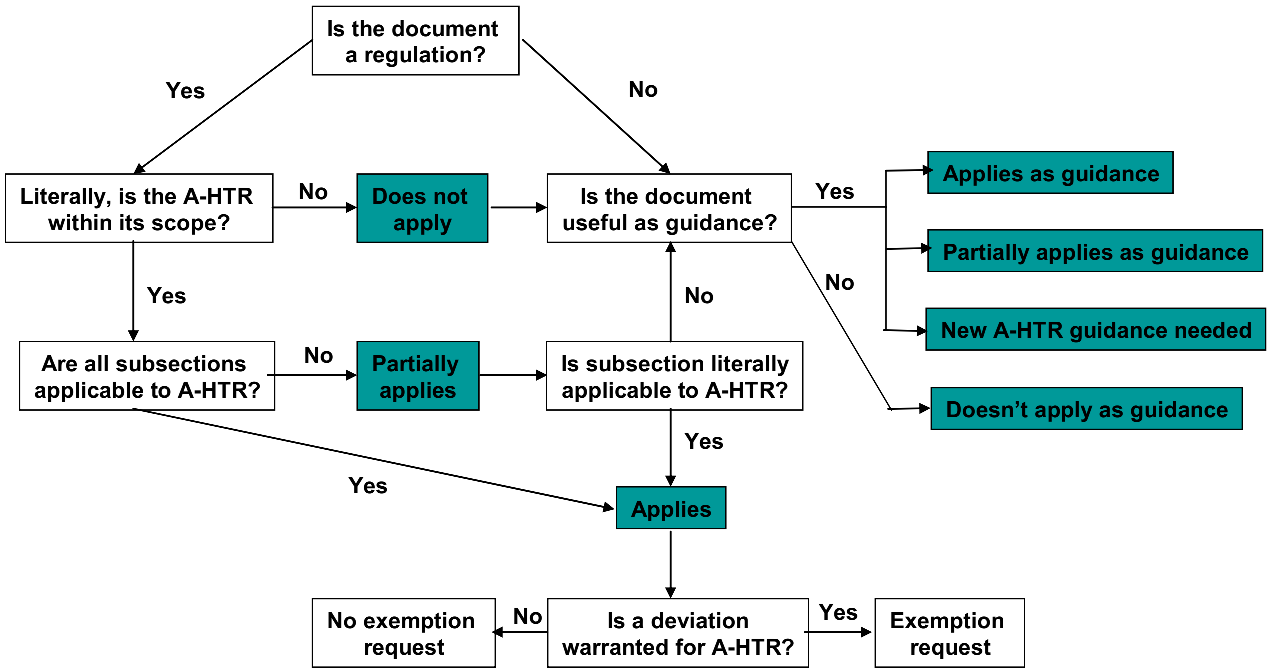


Figure 12-3: Logic for Pilot Screening of Regulations by AREVA for ANTARES

A Delphi process was utilized (i.e., subject matter experts to screen the sample set of regulations and develop the common logic from the group process). The expert panel consisted of a group of individuals with diverse backgrounds. The panel members combined nuclear industry experience was more than 200 years. Backgrounds included extensive experience in LWR and HTGR design, operations, maintenance, construction, licensing, reactor regulation and probabilistic risk assessment. Two of the panel members were involved in similar regulation screening workshops for the MHTGR and PBMR.

Results of Regulation Screening

Repetitive patterns were recognized as the regulations in 10CFR50 and other selected regulations were screened by the AREVA expert panel into the three broad categories of applicable, partially applicable, and not applicable. These patterns were used to define a total of 6 subcategories with two subcategories for each of the 3 broad categories. All of the screened regulations were found to fit into one of these six categories, as defined in Table 12-1.

Table 12-1: AREVA Definition of Regulation Screening Categories for HTRs

Screening Categories	Sub-Categories	Definition
Applicable	A1	Administrative regulation: Fully and literally applies to HTRs
	A2	Technical requirement: Fully and literally applies to HTRs
Partially Applicable	P1	Technical requirement: Partially applies to HTRs; includes subparts not applicable to the HTRs, otherwise fully and literally applicable
	P2	Technical requirement: Partially applies to HTRs; refers to LWR specific design features, LBEs, operating conditions, and safety functions; but provides guidance for HTR specific requirements after accounting for differences in design, materials, operating conditions, LBEs, and HTR specific safety functions
Not Applicable	N1	Not applicable to the HTRs: Does not apply to either current or future power reactors
	N2	Not applicable to the HTRs: Applies safety principles for light water reactor specific SSC, LBEs, or safety functions with no HTR analogue

The results of AREVA's pilot screening are summarized in Table 12-2 Of the 166 regulations that were screened, 107 were found to be applicable, 29 were partially applicable, and 30 were not applicable. Of the 107 that were found to be fully applicable, 69 were of an administrative nature while the remaining 38 were considered technical requirements. All the partially applicable requirements in this pilot study were of subcategory P2. Of the 30 not applicable requirements, 23 were of subcategory N2. Of the 59 regulations that were either partially or not applicable, 33 were found to provide useful guidance in the formulation of AREVA-HTR licensing requirements. These 33 included all the items in category P2 as well as three of the not applicable requirements.

Table 12-2: Results of Regulation Screening Categories

Screening Categories	Number (%)	Sub-Categories	Number (%)
Applicable	107 (64%)	A1	69 (42%)
		A2	38 (38%)
Partially Applicable	29 (17%)	P1	0 (0%)
		P2	29 (17%)
Not Applicable	30 (18%)	N1	7 (4%)
		N2	23 (14%)
Total	166 (100%)		166 (100%)

Follow-on Steps

The above described screening of regulations only considered a limit set of current licensing requirements for LWRs and did not include the supporting regulatory guidance, standard review plans, policies, and industry codes and standards. This process needs to be completed by the applicant and the NRC in support of licensing the NGNP.

It is important to obtain agreement during the pre-application period on the use and utility of the logic to be used in screening the current LWR requirements. This will enable a properly focused application that can be efficiently and effectively reviewed and address the necessary and sufficient requirements to provide reasonable assurance of public safety and security for the NGNP design.

Once agreement is reached with the NRC on the process and logic to be used, the applicant will screen the entire set of NRC regulations to determine applicability, partial applicability and inapplicability. During the expanded screening effort, applicant will continue to validate the logic to be used and necessary refinements to Figure 12-1 will be formulated. Also, actual NGNP design information will be used as it becomes available as well as the results and risk insights from PRA work as it becomes available to refine if needed the decisions on specific partial or not applicable documents. At the completion of the regulation screening, the results will be reviewed with NRC. The final logic chart will be confirmed for use to examine sub-tier regulatory documents that stem from the regulations such as regulatory guides and the standard review plan.

It is the intent of the process that NRC and applicant will use the pre-application period to assure that the full set of regulatory documents that must be considered in whole or in part for the NGNP design are identified in advance of completing the application.

The process described above is an iterative process. For example, the determination of whether a LWR regulation is useful in whole or part as guidance for the NGNP depends upon the design of the NGNP. However, the design of the NGNP will be affected by the requirements as well as guidance applied during the design process. Thus, it may be that both the design and the identification of applicable regulations will evolve over time, with finality being achieved at the time when the license is issued. At the pre-application stage, the applicant is only seeking a tentative agreement with the NRC staff on what regulations are applicable or might be applicable, realizing that the NRC and the applicant cannot make a final determination until both have had the opportunity to review the design and the supporting PRA and the design itself is final.

12.3 Regulatory Agencies and Primary Requirements

The NGNP, which will be located on the INL site, will fall under the jurisdiction of many regulatory agencies. The NRC is the primary licensing authority; however, the DOE and others also have significant requirements that must be met. A listing of these regulatory agencies has been compiled in Table 12-3 below along with the primary regulatory requirement(s).

Table 12-3: Listing of Regulatory Agencies and Primary Requirements

Agency/Primary Requirements	Comments
Nuclear Regulatory Commission (NRC): 10 CFR Part 50 Requirements 10 CFR Part 52 Certification will be obtained for the commercial successor plants.	NGNP will be licensed under 10 CFR Part 50. Ref. 1: 2nd International Topical Meeting on High Temperature Reactor Technology, Beijing, China, September 22-24, 2004, "ANTARES: The HTR/VHTR Project at Framatome ANP." From this reference, AREVA intends to license the NGNP Demonstration Plant in accordance with 10 CFR 50. Based on safety demonstrations of the nuclear heat source, which is common to the commercial plant designs, AREVA intends to certify the commercial plant designs for both electricity generation and hydrogen production/process heat under 10 CFR 52. Link: http://www.iaea.org/inis/aws/htgr/fulltext/htr2004_a10.pdf Ref. 2: AREVA NGNP Status Meeting (2/20/2007) Slides – "NGNP Project 50% Design Review – Licensing" presented by F. Shahrokhi and W.Szymczak. From this reference, the strategy for the NGNP Licensing is application for a Class 104c License Under 10 CFR 50.21. The basis being that there is no operating history for HTRs and the NGNP is a FOAK Prototype. (Also see *Note on next page under the Department of Energy and Bureau of Land Management)
Department of Energy (DOE) and Bureau of Land Management (BLM): Environmental Impact Statement (EIS)	National Environmental Policy Act (NEPA) Requirement under Environmental Protection Agency (EPA) Council on Environmental Quality, 40 CFR Part 1502. The purpose is to inform the public, Federal, State, and local agencies, Tribes, and other interested parties of the proposed action. In this case it would be the conveyance of DOE property to the nuclear plant applicant. *(Note: The requirements of the EPA in the preparation of an EIS as well as the Environmental Report/Early Site Permit/Environmental Monitoring program for a plant licensed under 10 CFR Part 50/52 above would include the requirements of numerous other Government Acts such as: (1) Emergency Planning and Community Right-to-Know Act, (2) Endangered Species Act, (3) Resource Conservation and Recovery Act, (4) Toxic Substances Control Act, (5) Clean Air Act, (6) Clean Water Act, (7) Safe Drinking Water Act as well as Executive Orders such as 11988 – Floodplain Management and 11990 – Protection of Wetlands (if applicable). Other consideration would be required to be given to the National Historic Preservation Act and the Native American Graves Protection and Repatriation Act.)

Agency/Primary Requirements	Comments
	Ref: INL Site Environmental Reports. Link: http://www.stoller-eser.com/Publications.htm
DOE; Department of Homeland Security (DHS) (which includes the Federal Emergency Management Agency – FEMA); and the Department of Defense (DOD): Security	The National Nuclear Security Administration is a semi-autonomous agency within the DOE. The Office of Health, Safety and Security is also another DOE office. The Department of Homeland Security as well as the Department of Defense also maintains work projects at the INL (specifically the Specific Manufacturing Capability, SMC, facility for DOD). All of the preceding organizations have security responsibilities at the INL. Due to the nature of the ongoing activities at the INL including the type of material that is handled there, the DOE/DHS/DOD Security requirements may be more restrictive than those required by the NRC. Additionally, the people gathering classified information and processing it will have to be probably at least Q-cleared. And, appropriate facilities and computer systems for handling and storing this classification of information will have to be provided. Ref: Link http://www.inl.gov/
DOE and Department of Transportation (DOT): Road Access/Permits Rail Access/Permits	Separate roadway access will need to be conveyed from DOE to the nuclear plant applicant. Railroad access will need to be provided to the nuclear plant applicant. Note: The Union Pacific Railroad's Arco Branch crosses the southern portion of the INL and provides rail services to the INL. This branch connects at the Scoville Siding with a DOE spur line, which links with developed areas within the INL. The Settlement Agreement/Consent Order (U.S. v. Battelle 1995) limits the shipment of naval spent fuel to the INL and since this would be competing for rail use, it could impact any shipment to the nuclear plant site. Ref.: Link http://www.eh.doe.gov/NEPA/eis/eis0290/Ch_4/4_11/411_Traffic.HTML
Federal Energy Regulatory Commission (FERC): Transmission Rights Electric Transmission Construction Permits	Requires Permits to Site Interstate Electric Transmission Facilities. The INL site is located at the intersection of a regional network of 13.8-, 230-, and 345-kV transmission lines. The east-west transmission corridor provides interconnection between coal-fired resources in Montana, hydro resources in western Idaho, and the Western market. The north-south transmission corridor connects Utah to Montana through a 230- kV line. From the INL area, the network allows access to Montana, Wyoming, Utah, Nevada, Oregon, and Washington. Ownership of the network is split between IdaCorp (Idaho Power), BPA, PacifiCorp (Utah Power), and Montana Power. Transmission capacity in

Agency/Primary Requirements	Comments
	both the east-west and north-south directions is fully subscribed at times; during peaks, the area is transmission constrained. Ref: "Final Report - Site Selection Evaluation for Pebble Bed Modular Reactor (PBMR) Generating Unit," Prepared for the DOE by Exelon Generation Company, LLC, August, 2002. Link - http://www.ne.doe.gov/np2010/espStudy/espStudyExelon.pdf Ref: Link http://www.ferc.gov
State of Idaho: Idaho Department of Water Resources – Water Rights - Permit	The right to divert the public waters of the state of Idaho requires the obtaining of "Water Rights." Ref: Link http://www.idwr.state.id.us/water/rights/ For Example: Based on a Federal Reserve Water Right, the Department of Energy (DOE) and the State of Idaho negotiated a State water right for the INL. The INL is permitted a water pumping capacity of 80 cubic feet per second and a maximum water consumption of 35,000 acre feet per year. Ref: "Final Report - Site Selection Evaluation for Pebble Bed Modular Reactor (PBMR) Generating Unit," Prepared for the DOE by Exelon Generation Company, LLC, August, 2002. Link - http://www.ne.doe.gov/np2010/espStudy/espStudyExelon.pdf
Idaho Power: Customer (Purchaser Contracts)	Idaho Power is investigating additional generation resources within the southern Idaho region to serve native load and free some transmission capacity. Idaho Power Plans to increase its power supply by about 800 average megawatts over the next ten years. The INL site is served by its own 13.8-kV transmission loop through its substation at Scoville. The Scoville substation is in turn fed from the Idaho Power substation at Antelope with 13.8- kV and 230-kV backup on site from Utah Power. Current additional capacity available at the Antelope substation is approximately 400 MW. Generating capacity in excess of 50 MW at INL would require line upgrades or a dedicated transmission line to the edge of the site and interconnection to the substation. Ref: (1) "Final Report - Site Selection Evaluation for Pebble Bed Modular Reactor (PBMR) Generating Unit," Prepared for the DOE by Exelon Generation Company, LLC, August, 2002. Link - http://www.ne.doe.gov/np2010/espStudy/espStudyExelon.pdf (2) Idaho Public Utilities Commission Order No. 29762, April 2005. Link - http://www.puc.idaho.gov/internet/press/042205_IPCoIRP.htm

12.4 Role of PRA

Probabilistic Risk Assessment (PRA) provides a logical and structured method to evaluate the overall safety characteristics of the NGNP plant. This is accomplished by systematically enumerating a sufficiently complete set of accident scenarios and by assessing the frequencies and consequences of the scenarios individually and in the aggregate to predict the overall risk profile. It is the only available safety analysis method that captures the dependencies and interactions among SSC, human operators and the internal and external plant hazards that may perturb the operation of the plant. The quantification of both frequencies and consequences must address uncertainties because it is understood that the calculation of risk is affected by uncertainties, especially those associated with the potential occurrence of rare events. The treatment of uncertainties is especially important in this application because of the relative lack of relevant operating experience with HTGRs as well as the lack of experience in the performance and review of HTGR safety analyses. The quantification of frequencies and consequences of event sequences and the associated quantification of uncertainties provide an objective means of comparing the likelihood and consequence of different scenarios and of comparing the assessed level of safety against the top level regulatory criteria.

The use of PRA in the licensing approach for the NGNP is not a unique concept and builds upon the approaches used for the MHTGR[15]and PBMR [17] that have been reviewed by the NRC [16,18].

12.4.1 Objectives of NGNP PRA

In order to determine the scope and necessary characteristics of the PRA that will be required for the development of licensing bases for the NGNP, it is important to list the objectives of the evaluation. The objectives are:

- Provide risk insights into the design of the NGNP SSCs that perform safety functions
- Provide a reasonably complete set of event sequences and event frequencies from which to select the LBEs
- Confirm that the TLRC, including the safety goal QHOs for individual and societal risks, are met at a U.S. site or sites
- Provide a primary technical basis for the development of reactor-specific RDC for the plant
- Support the determination of safety classification and special treatment requirements of SSCs
- Support the identification of emergency planning specifications including the location of the site boundary
- Support the development of technical specifications
- Provide insight on the available defense-in-depth in the design

12.4.2 Scope of NGNP PRA

Since the PRA is the primary source of candidate event sequences for the selection of LBEs, completeness in the enumeration of event sequences is an essential PRA attribute. As such the scope of the PRA needs to be as complete as possible consistent with the state of the art of PRA technology. The PRA for the NGNP needs to include the following aspects of a full-scope PRA:

- All sources of radioactive material including the sources in the reactor core and primary heat transport system (HTS), process systems, spent fuel storage, and any other licensable source of radioactive material.
- All envisioned operating and shutdown modes including plant configurations expected for planned and unplanned maintenance, refueling, and inspections.

- All potential causes of initiating events including internal plant hardware failures, human operator and staff errors, internal plant hazards such as internal fires and floods, and external plant hazards such as seismic events and transportation accidents.
- Event sequences that define a reasonably complete set of combinations of failures and successes of SSCs and operator actions in the performance of safety functions and with sufficient detail to characterize mechanistic source terms and offsite radiological consequences comparable to a LWR Level 3 PRA as defined by NUREG/CR-2300 [20].
- Quantification of the frequencies and radiological consequences of each accident family defined to group the event sequences in the PRA including mean estimates and a quantification of uncertainty in the form of uncertainty probability distributions that account for all quantifiable sources of uncertainty. Additional non-quantified sources of uncertainty shall be identified and examine with the use of an appropriate set of sensitivity analyses.
- For NGNP plants comprised of multiple reactor modules, the PRA shall define event sequences that impact reactor modules independently as well as those that impact two or reactor modules concurrently.
- In order to support the evaluation of RDC, the PRA will be capable of evaluating the cause and effect relationships between design characteristics and risk as well as be able to support a structured evaluation of sensitivities to examine the risk impact of adding and removing selected design characteristics.

The scope of the NGNP PRA needed to support the risk-informed approach to licensing will need to be as comprehensive and sufficiently complete as would be covered in a full-scope, all modes, Level 3 PRA covering a full set of LWR internal and external events. However, there will be differences in the PRA due to inherent differences between LWR and HTR technologies.

12.5 Coordination with New Regulatory Activities

NGNP licensing activities must be closely coordinated with the NRC's risk-informed, performance-based licensing framework initiative currently in-progress. This initiative is expected to result in the codification of the proposed risk-informed and performance-based alternative to Part 50 regulations. This is, by far, the most crucial regulatory activity that will play out in parallel with NGNP licensing, construction and, ultimately, operation.

The technical basis for the development this framework established for the licensing of future commercial non-LWR nuclear power plants has been outlined by the NRC staff in draft NUREG-1860. The need to develop such a risk-informed, performance-based framework (technology-neutral or technology-specific) for future reactors is based on the following considerations:

1. The regulatory structure for current LWRs has evolved over five decades. Most of this evolution occurred without the benefit of insights from probabilistic risk assessments (PRAs) and severe accident research.
2. While the NRC has over 30 years experience with licensing and regulating nuclear power plants, this experience (as reflected in regulations, regulatory guidance, policies and practices) has been focused on current light-water-cooled reactors (LWRs) and may have limited applicability to future reactors.
3. The provision of a framework that is technology-neutral with respect to important probabilistic and deterministic criteria governing risk acceptance and performance will facilitate the development of a consistent, stable, and predictable set of requirements that are both risk-informed and performance-based.

The purpose of the framework is to provide the technical basis to support the development of a technology-neutral, risk-informed and performance-based process for the licensing of new nuclear power plants (NPP). As such, it documents an approach that can be used to create a 'level playing field' for all future reactor technologies in terms of the safety criteria to be met.

Within the framework, design criteria and guidance are established for the identification and selection of licensing events and for the classification of risk significant components. High level requirements are established for PRA scope and acceptability to support the use of risk results and insights in the development of risk-informed requirements.

This framework approach, scope and criteria may eventually be used by the NRC staff to develop a set of regulations (i.e., potentially the future 10 CFR 5x) that would serve as an alternative to 10 CFR 50 for licensing future NPPs. The regulations developed from the framework approach could still be used in conjunction with 10 CFR 52 for carrying out the licensing process, i.e., obtaining a combined operating license and/or design certification.

Thus, it is crucial that NGNP licensing process and the risk-informed performanced based licensing initiative be closely coupled to ensure the successful transition from NGNP licensing to the commercial licensing framework.

12.6 NGNP and Hydrogen Plant Collocation Issues

The purpose of this section is to discuss some of the considerations and requirements associated with collocating the NGNP prototype with a hydrogen production process plant, including overall siting requirements for both the nuclear island and the hydrogen facility. Although the hydrogen plant design is not within the AREVA team scope of work, the co-location licensing issues are an important part of the NGNP prototype facility licensing. Similarly, these licensing considerations would also be relevant to the use of a commercial VHTR as process heat facility to support a chemical or petrochemical facility. However, the focus of the discussions in this section is on collocation with hydrogen production as this is currently a functional and design requirement for the NGNP prototype. Furthermore, it is expected that hydrogen production demonstrations of two processes with be part of NGNP: (1) the sulfur-iodine (SI) thermochemical process, and (2) high temperature electrolysis (HTE).

As discussed in the Secondary Cycle Concept Study [21] safety is a critical consideration in the development of the NGNP. While the specific NGNP design and operating capabilities will ultimately have an impact on the detailed development and licensing of NGNP, a number of specific aspects of the design and operations are assumed to have a secondary effect on these topics. This would include final selections of power level and reactor inlet and outlet temperature, as well as operational flexibility for all-electric, cogeneration or all-hydrogen operations. Other considerations that would have a secondary impact on the safety and siting discussions in this section are the number of primary loops for the PCS and hydrogen production process. This discussion also assumes an all gaseous heat transfer strategy (i.e., no use of liquid metal or molten salt in the heat transfer loops).

12.6.1 General Siting

Regulatory requirements applicable to the siting of NGNP at INL will likely include regulations that are: (1) federal requirements (e.g., 10 CFR 100), (2) DOE Orders (e.g., DOE O 420.1), and (3) INL site-specific (e.g., BEA contract requirements, site-specific security issues, etc.). In addition, there will be other requirements that govern the design and permitting of the hydrogen production facilities, including but not limited to: (1) EPA requirements, (2) federal regulations and permits under 40 CFR Part 70 (e.g., Title V permits may be required, especially if a fossil fired energy source is used for cold commissioning of the hydrogen production facilities), and (3) local air and water quality control boards. One study concluded that there may be as many as 4,500 codes, standards and guidance documents applicable to the design of a nuclear facility such as NGNP [22] . Commercial Codes and Standards will also apply. For hydrogen generation, storage and transportation issues, up to 20 separate organizations have or are planning to issue guidelines or mandate codes and standards for hydrogen facilities. The DOE Hydrogen Codes and Standards Coordinating Committee (HC&SCC) is integrating these efforts.

Figure 12-4 below summarizes some of the licensing rules, guidelines and criteria that will apply to NGNP that are particularly relevant to the collocation of the hydrogen facility (given the hydrogen production processes may introduce new environmental or dynamic events to be considered in the NGNP design basis).

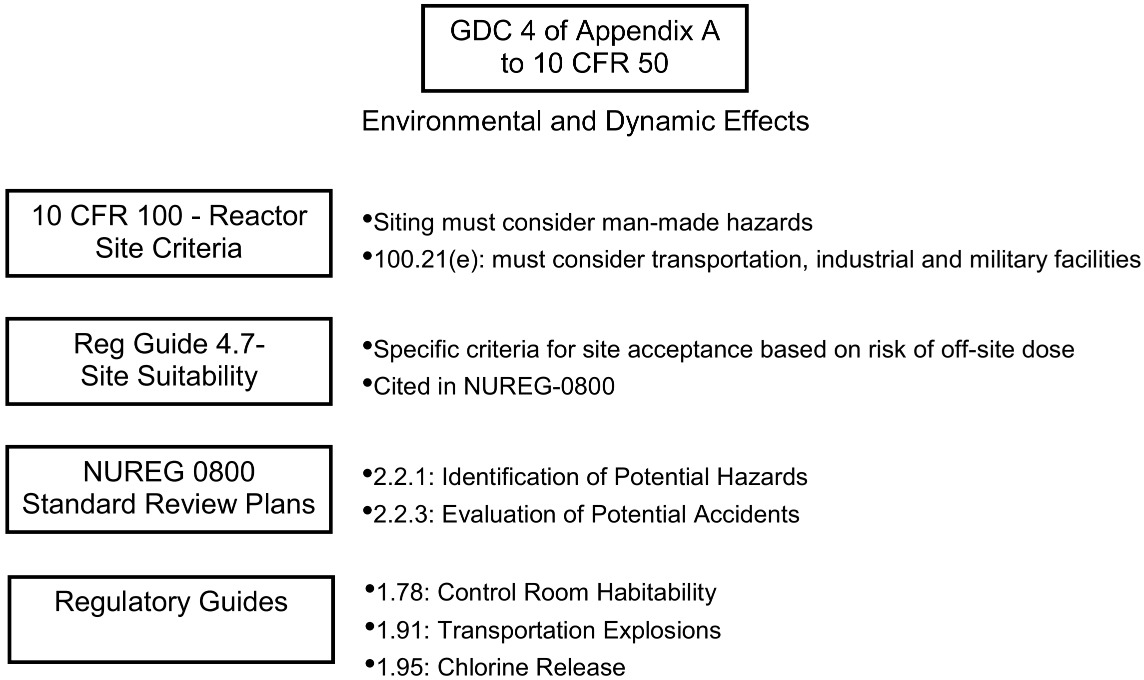


Figure 12-4: RUS Licensing Rules, Guidelines and Criteria for NGNP

General Design Criteria (GDC) of Appendix A of 10 CFR 50 summarizes the high level requirements for considering the effects of external events. 10 CFR 100 (Reactor Site Criteria) discusses how potential man-made hazards must be considered. Specifically, 100.21 (e) includes transportation, industrial (e.g., chemical production), military facilities. More specifically:

"Potential hazards associated with nearby transportation routes, industrial and medical facilities must be evaluated and site parameters established such that potential hazards from such routes and facilities will pose no undue risk …to the facility"

Specific attention is focused on chemical plants, refineries, gas or petroleum product storage that may produce:

- Missiles
- Shock waves
- Flammable vapor clouds
- Toxic chemicals/corrosive gases, or
- Incendiary fragments

All of the above events are potentially relevant to collocation of NGNP with either thermochemical or HTE based hydrogen production. Furthermore, if a VHTR for hydrogen production were to be located near a petrochemical facility or other chemical process plant, consideration of events initiating at these facilities must be considered.

More specifically, Regulatory Guide 4.7 mandates that the licensee must demonstrate the following:

- Any accidents associated with external events from the hydrogen facility or other chemical facility would not result in a radiological consequence that exceeds dose guidance of 10 CFR 50.34 (a) (1), or
- The accident poses no undue risk because it is unlikely to occur, i.e.,. 10E-7 per year conditional probability (for LWRs) or a comparable likelihood of risk measure for HTRs), or
- The NGNP plant can be designed so that it is not affected by the accident.

Any facility within a 5-mile radius must be identified in the licensing process. This would obviously include the hydrogen production facility as it is desirable to locate the hydrogen facility as close as possible to minimize heat transfer losses and pumping losses. In some cases, facilities beyond the 5 mile zone have been included in the Regulatory Guide 4.7 type analyses (e.g., military facilities or some transportation routes). Also, a conditional probability limit of 10E-6 per year is sometimes considered acceptable if qualitative engineering judgment suggests that the industrial accident risk is low, often based on actuarial data.

Other applicable guidance can be found in NUREG-0800 and include:

- SRP 2.2.1-2.2.2 (Potential Hazards in Site Vicinity) – This plan defines requirements for identifying potential hazards, such as:
  - Existence of hazardous materials (e.g., explosive or toxic materials which would include hydrogen, sulfur dioxide, sulfur trioxide, etc. used in the hydrogen plant)
  - All manufacturing and transportation routes (including rail spurs or lines within 350 to 500 m)
  - Pipelines (likely to exist near any petrochemical or chemical facility which employs a VHTR for process heat)
  - Cryogenic fuels.
- SRP 2.2.3 (Evaluation of Potential Accidents) – This plan provides guidance for:
  - Performing calculations to quantify blast loadings or toxicity of chemicals.
  - Probabilistic modeling
  - Control room habitability

Regulatory Guide 1.78 – Control Room Habitability - is likely to be particularly relevant to the siting and licensing of NGNP, and especially in defining the distance between the nuclear island ad the hydrogen production facilities. This Guide contains:

- Assessment guidelines for toxins and asphyxiants
- Rules for evaluating the effects of any chemical < 500 meters from the NGNP control room that exists in a quantity greater than 100 pounds.
- Guidelines for toxicity limits.

While helium and sulfur dioxide are included as examples in Regulatory Guide 1.78, other chemical in the hydrogen production facilities are not. These include HI, I2, sulfuric acid, SO3, phosphoric acid and hydrogen. These species will still need to be subject to review. In the event that the hazard associated with the release of one or more of the chemicals used at the hydrogen production facility exceeds allowable levels in the control room after an event, compensatory measures can be employed in the NGNP design which could include:

- Detection capability (to allow time for control room isolation)
- An isolation or ventilation capability, or
- Operator protection, Self Contained Breathing Apparatus (SCBAs).

Finally, regulatory Guide 1.91- Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Plants - will need to be considered for both the transportation of hydrogen if performed by tanker in gaseous or liquid form, as well as for the delivery of initial and replenishment chemicals for the hydrogen production facility (sulfuric acid, sulfur dioxide, iodine etc.). Hydrogen may or may not need to be provided to the site for electrical generator cooling. Other chemicals that will likely be required at the site from time to time will be required for maintenance such as chemical cleaning of the hydrogen production facility units (acids, oxidants), as well as water treatment for cooling towers or make-up water (e.g., chlorine).

While the above discussion focuses on NRC-type requirements, the DOE has generally followed the NRC lead in the evaluation of safety concerns through DOE Orders. Since NGNP is expected to be licensed through an NRC process, these Orders such as 5480.6 “Siting” or DOE Facility Safety Requirements (DOE O 420.1A) are not discussed specifically herein. They do however require similar consideration of external events such as explosions, fires, hazardous material releases from nearby military/industrial facilities as well as transportation accidents.

12.6.2 General Licensing Issues for Locating NGNP Adjacent to the Hydrogen Plant

Whether NGNP is licensed under 10 CFR 50 or 10 CFR 52, the licensing process will require justifying that collocation of the nuclear island and the chemical plant is acceptable and that NGNP serves as the framework for the licensing of future commercial VHTRs that would deployed for nuclear process heat applications. It is clear however the nuclear island and the hydrogen facilities will be subject to a range of licensing requirements. Each facility may require licensing or permitting by more than one regulatory authority. Some aspects of the licensing and permitting would likely be performed for both the NGNP and the hydrogen production facilities on a site basis, such as discharge permits under a NPDES (or state equivalent). The potential that single license submittal for atmospheric discharges for the site cannot be ruled out (Air Quality Management board submittals).

Whether or not NGNP pursues a Part 50 or 52 strategy, it is highly likely that the criteria for deign and siting of the nuclear plant and consequent parameters for the hydrogen plant, such as distances between the facilities, will need to meet not only the nuclear aspects requirements of the previously discussed rules, regulations and guidelines, but also safety regulations for the hydrogen production facility. Some of these aspects are discussed in the next section.

12.7 NGNP Licensing Impact on DC and Licensing of Commercial Plants

There are several aspects to consider regarding NGNP licensing and the design certification of commercial plants.

First, the NGNP may not be the lead plant at the forefront driving the risk-informed performanced-based licensing framework. It is likely that other commercial ventures, such as PBMR, may lead. In this case, NGNP licensing must maintain a close liaison with the NRC on PBMR developments.

Second, the design certification of the NGNP-commercial version relies on the successful development of the licensing infrastructure under 10 CFR 53 and the successful testing and demonstration of the NGNP itself. Without success in both areas, design certification cannot be secured and future commercialization of the technology is jeopardized.

This is an obvious high-level licensing linkage between demonstrator and commercial plants; however, there is the impact of the resulting licensing requirements themselves that impact both design certification and licensing. Resolution of key regulatory issues, such as containment versus confinement, or level of fuel performance, can have dire consequences not so much with respect to licenseability but more so with respect to commercial viability. Hence, the licensing requirements that evolve from NGNP will make or break the future of the commercial NGNP.

Third, there is the question of "Who" secures design certification? It is certainly not within the purview of DOE unless the DOE intends to develop a generic NGNP commercial offering that others can reference. More likely, a member or member(s) of the public-private partnership chartered to design, license, construct and operate the NGNP will elect to take the NGNP design (or at least their share of the design) forward for commercial development. To what extent the partnership members have exclusive technology rights will bear heavily on the attractiveness of being involved in the partnership. The answer to these questions is not within the scope this document.

12.8 Licensing Impact of NGNP Organizational Structure

The NGNP Alliance, of which DOE is a member, has proposed that the NGNP be designed, constructed and operated as a commercial entity under the public-private partnership organization as shown in Figure 12-5 below.

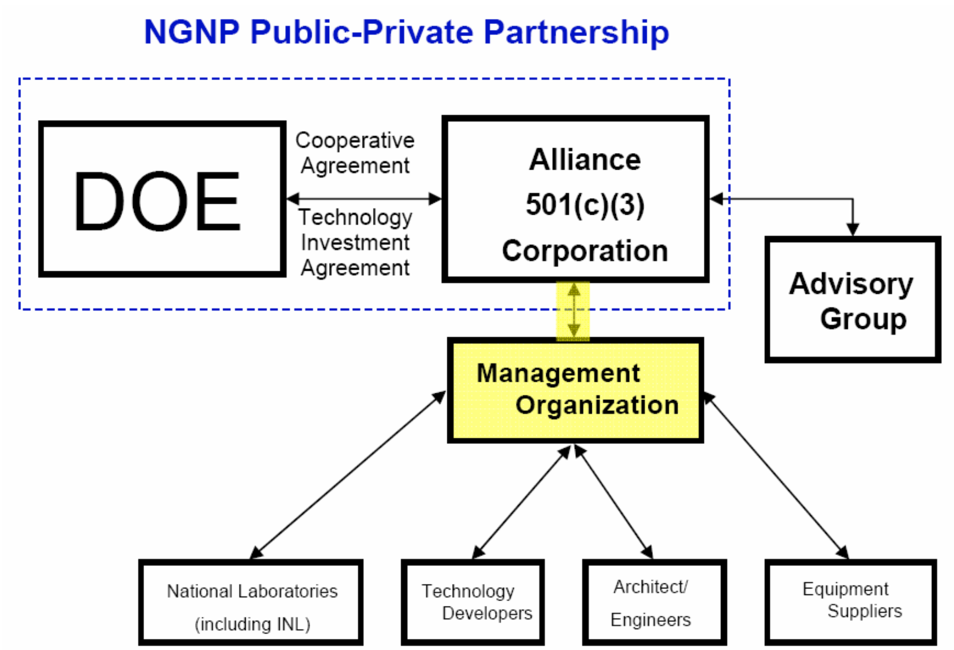


Figure 12-5: NGNP Public Private Partnership

How the above organizational structure is treated in licensing space is critical to a timely and successful NGNP licensing process. The situation is unique in a licensing sense because it is quite different from the normal situation of a single applicant seeking licensure from the NRC. Further complicating the situation is the fact that the NGNP is to be located on federal land at DOE's INL site.

To proceed most efficiently, the "split-license" strategy described below should be proposed and negotiated with the NRC:

1. NGNP Alliance will be the primary licensee dealing with NRC, State of Idaho, and local regulations and requirements.
2. The operational portion of the Operating License (OL) will be held by the "NGNP Alliance" Corporation or by an operating subsidiary company. The Alliance becomes the responsible operating party and is responsible for all safety.

3. The financial portion of the OL belongs solely to the DOE, which as a separate entity and a member of the partnership alliance, will hold long term financial liability for the NGNP and have to file the certificate for such.

The real portion of the OL is also held by the DOE by means of a long term (100-year) leasing arrangement with the Alliance for use of INL site. This keeps the issue of land use and ownership fully within DOE's jurisdiction and avoids complications by third parties that could potentially occur in a land sale/transfer arrangement.

This above strategy is not without precedent as evidenced NRC's approval of the split Part 70 License for the MOX facility located at the Savannah River Site in South Carolina. Further development of the split licensing strategy should receive top priority in the NGNP conceptual design phase.

12.9 Recommendation for NGNP Licensing and Permitting

AREVA recommends that the NGNP be licensed to the public-private partnership arrangement previously described under the conventional two step 10CFR50 licensing process:

1. Secure a construction permit based on the review of a preliminary safety analysis report (PSAR); and,
2. Secure an operating license based on the review of a final safety analysis report (FSAR). The license will initially be for a non-LWR "Prototype" plant that will be converted to a commercial Class 103 license following a successful demonstration period.

Additionally, AREVA recommends that elements of 10 CFR 52 be carried out in parallel with the Part 50 license, in particular, maintaining close liaison with the NRC through pre-application technical exchanges and interactions that will be necessary to develop technology neutral and HTR-specific licensing bases for the NGNP demonstrator and commercial plants

This is essentially a hybrid approach that satisfies the need to initiate construction activities as early as possible but also minimizes the risk of construction prior to obtaining an operating license. However, the opportunity that the Part 50 approach offers in terms of an early construction start can only be seized if the key issue of containment versus confinement is resolved early in the licensing process.

Experience from the operation and prototype testing of the demonstration NGNP will be needed to support the design certification of future commercial versions because, unlike the NGNP, the commercial plan will follow the Design Certification Process provided in 10CFR52 [19]. Therefore, the experience gained during the NGNP commercial licensing process will support the future application for and receipt of a design certification for a similar high temperature reactor technology application.

13.0 MAINTAINABILITY

The development of the NGNP maintenance requirements and maintainability were excluded from AREVA scope of work. However, AREVA considers plant maintenance and maintainability are highly important considerations in the design selection of the NGNP prototype facility and the subsequent commercial plant. Therefore, this section was included to highlight AREVA views on the plant maintenance and ISI requirements. Moreover, the multi-loop indirect cycle NGNP design proposed by AREVA is unique for its maintainability; therefore a section was added to highlight these features of the AREVA design.

13.1 Plant Maintenance Requirements

13.1.1. The NGNP plant shall be designed to permit replacement of life-limited and/or failed components over its life time. The time required to effect such replacements shall be consistent with the plant availability requirements.

13.1.2. The plant designer shall develop a design life classification system and listing which categorizes items (i.e., components and subsystems) according to design life capability and shall develop the strategy to be employed to support the overall plant design life requirement. This design life classification shall be incorporated in the planning of the preventive maintenance and inspection programs.

13.1.3. The plant designer shall recommend, by the end of final design, a comprehensive program for obtaining data for evaluating the actual remaining life capability of long life components, based upon their actual operating history and measurement of their life limiting characteristics.

13.1.4. Building configuration, access route designation, and shielding shall be designed to control personnel radiation exposure.

13.1.5. The number of inaccessible areas due to high radiation levels during reactor operation shall be minimized to facilitate routine operation and maintenance activities.

13.1.6. The capacity factor loss due to planned outages averaged over the lifetime of the plant shall not exceed 8%.

13.1.7. The design shall allow all planned inspection and maintenance activities that must be accomplished with the reactor shut down to be accomplished within the period required for refueling.

13.1.8. Building design and equipment layout shall include features (e.g., cranes, hoists, monorails) to facilitate removal and replacement of major equipment items.

13.1.9. Headroom, pull space, lay down areas, and work space for component and equipment maintenance shall be provided. Permanently installed lifting devices and beams, rails, etc. for temporary attachment of lifting devices shall be included where required in the design. To facilitate access for in place maintenance or component replacement, obstructing building structural members shall be removable without cutting. Consideration shall be given to space requirements (e.g., layout of aisles, sizing of doorways, elevator(s), etc.) for moving equipment and components from their permanent location to shop facilities.

13.1.10. Equipment and components shall be accessible from normally provided floors and platforms to the greatest degree practicable. Where components are not accessible from floors or platforms, special access, such as permanently installed ladders and local platforms, shall be provided.

13.1.11. Nuclear island systems and components shall be designed to facilitate hands on maintenance, subject to the safety and radiation protection ALARA requirements. Remote maintenance techniques shall be considered in the design where reduced radiation exposure or improved capacity factor may be economically achieved.

13.1.12. The design of special maintenance tools shall be provided by the equipment vendor.

13.1.13. The design and arrangement of plant systems, equipment, and components shall facilitate on-line maintenance.

13.1.14. The NHS module shall be designed to allow all components within the helium pressure boundary to be removed and reinstalled to make possible inspection, repair and replacement. A trade study to determine the method of removal and replacement of components within the primary pressure boundary, based on the degree of difficulty, time and cost and the projected probability of occurrence shall be completed and documented by completion of preliminary design.

13.1.15. To reduce the cost of spare parts inventory, the design shall specify components in a manner that limits the number of different types, sizes, and ratings (e.g., temperature, pressure) of parts.

13.1.16. A preventive maintenance plan shall be developed and documented based on the final plant design. A first draft shall be issued on completion of preliminary design. The plan shall identify the preventive maintenance requirements, tasks, methods, tools, personnel skills, and estimated worker hours (including those for health physics personnel) on a system basis for the categories of mechanical, electrical and control and instrumentation maintenance.

13.1.17. A spare parts list and recommended spare parts inventory shall be provided that is consistent with the preventive maintenance plan, anticipated unscheduled maintenance, and plant capacity factor requirements.

13.1.18. Anticipated tasks, methods, tools, personnel skills, and worker-hour requirements to accomplish unplanned maintenance shall be documented for each plant system. The analysis of maintenance activities shall be based on industrial experience (mean-time-between-failure and mean-time-to-repair data) for like type systems and components. Estimated worker-hours, including those of health physics personnel, shall include the time required to isolate systems and equipment, prepare for and conduct maintenance activities, and return to service.

13.1.19. Overhead cranes shall be designed to lift the heaviest equipment or component-part to be handled during planned operations, maintenance, and inspection activities.

13.1.20. Separate mechanical/machine, welding, electrical, instrument, and electronic maintenance shop facilities shall be provided.

13.1.21 The need for on-site mechanical maintenance/machine shop facilities suitable for handling radioactive (contaminated or activated) components shall be determined by the end of preliminary design. Such special process facilities should include decontamination areas with appropriate water supply, drainage and storage/purification facilities.

13.1.22 The NGNP plant design shall enhance maintainability by including human factors considerations. Lighting levels, heating, ventilation and air conditioning and plant services, such as communications and compressed air supply, shall be provided consistent with anticipated operation and maintenance activities and in compliance with applicable regulations, code, and general industrial practice.

13.1.23 Plant piping design shall minimize the need for snubbers and restraints and shall ensure inspectability.

13.2 In Service Inspection

1. The NGNP design shall provide access to the helium pressure boundary to permit in service inspection as required by appropriate sections of the ASME B&PV Code.
2. Where cost effective, the design of systems and components shall incorporate those features required to implement on-line in service inspection. If the unit or major component must be removed from service, design features shall be included to accomplish the inspection during the power unit allotted planned outage time.
3. Plant piping design shall minimize the need for snubbers and restraints and shall ensure inspectability.
4. Design documentation shall include plans and procedures for conducting in-service inspection and shall identify equipment necessary to conduct the inspection. The equipment vendor shall furnish the ISI equipment not commercially available.
5. An in-service inspection program shall be developed and maintained throughout the design process. The program shall include anticipated durations and worker-hours, including health physics, for isolating the equipment/system, preparing for and performing the inspections, and returning the equipment/ system to service. Physical and/or computer models shall be used to assess inspectability.
6. The plant design shall include those facilities and features required to set up the in-core fuel handling equipment for periodic inspections, maintenance, testing, and demonstrations of integrated equipment operation. Such inspection, maintenance, testing, and demonstrations shall not interfere with core refueling operations nor have an adverse effect on plant operation.

13.3 Maintainability

Capability to perform maintenance, ISI and repair/replacement operations is developed in the following systems or components which impose the long durations, i.e., the main primary systems (Vessel System and Primary Heat Transfer System) and the Power Generation System.

The general principles and duration of the maintenance, repair and replacement operations of systems or components will be optimized during the Conceptual Phase, according to cost penalty and to frequency of these operations. The following sub-sections provide a working basis for performing the required work. The needs for maintenance operations shall be defined on risk-informed bases.

Maintenance is defined as the operations necessary for achieving the required performance of equipment. The goal of maintenance is to detect any disturbance in the equipment which could degrade its capability to perform its function. The maintenance operations include the detection of off normal conditions and the capability to repair the anomaly. Compared to in-service inspection and periodic tests, maintenance operations have to correct the disturbances before the occurrence of the failure of the functions achieved by the equipment. If detection cannot be achieved sufficiently soon, the maintenance operations might be replacing of the equipment or parts of it.

For the conceptual phase, this requirement means that the maintenance needs of equipment have to be assessed considering:

- The function of the equipment with regard to the plant availability and safety (i.e., the consequences of the failure of the equipment).
- The operating conditions of the equipment.

- The probability of failure of the equipment.

As a working basis during the preconceptual design phase, the maintainability of any equipment is considered as followed: where this could be advantageous, the consequences on the design have to be assessed.

The maintenance operations shall minimize:

- The doses to operators
- The intervention durations
- The risks associated with the operations
- The consequences on environment.

13.3.1 Main Primary Systems

Vessel System:

Maintenance activities are limited to verify the isolation valves operability (automatic or manual according to design choice). Maintenance of PHTS, SCS and core equipments needs to open the Vessel System.

The repair/replacement operations of the Vessel System are limited to nozzles and secondary/auxiliary isolation valves of circuits connected to it.

The Vessel System equipments or functions to be inspected are mainly:

- Welds (vessels and nozzles)
- Isolation valves leak-tightness,
- Reactor vessel emissivity,
- Vessel System behavior (ageing) in the required time.

Adequate pressure tests of the whole Vessel System are mandated. The consequences of those tests and plant availability will have to be assessed.

The Vessel System will provide in a limited number, accesses for remote inspection inside it of loaded structures as core and PHTS supports. The capability of visual inspection from within the vessels using remote techniques will be considered. But to cope with availability requirements and taking into account difficulty of remote inspection in some internal zones, the periodic examination could be conceived from outside if potentially achievable and if acceptable from licensing point of view.

Primary Heat Transfer System (PHTS):

Maintenance is mainly required for the Main Primary Gas Circulator (MPGC) electrical motor, according to the manufacturer requirements.

The PHTS components are either repairable or replaceable.

For repair, the two potential options are still considered:

- Repair inside the vessel
- Repair outside the vessel. This option requires an easy removing ability of the component or structure to be inspected.

For replacement, the lifetime required for the main PHTS components is:

- MPGC electrical parts: frequent maintenance, see above; replacement according to manufacturer requirements.
- MPGC other parts: 10 years
- IHX: 20 years
- Other PHTS components: 60 years: no replacement is expected, but should nevertheless be possible.

The identified main needs of PHTS ISI are related to the following concerns:

- Cold primary helium to hot primary helium leak
- Cold secondary helium to hot secondary helium leak
- MPGC shaft or blade break
- Flap valve malfunction

A complete ISI of IHX is not required. This is justified by the low radiological releases occurring in case of IHX leak. The limitation of those radiological releases is achieved by the low radioactive helium content in normal operations and by the isolation valves.

Maintenance, ISI, repair and replacement of ducts, especially the hot duct, will have to be assessed.

13.3.2 Power Conversion System (PCS)

No precise information on the PCS maintainability has been produced during the preconceptual phase and this task will be performed in the next phases of the project. However, because of the NGNP indirect cycle design, radionuclides contamination of the PCS components are expected to low to nonexistent, therefore, PCS maintainability would be similar to the standard industry practice for non-contaminated turbomachinery and combined cycle components.

13.3.3 Durations of Maintenance

Refueling Outage

The annular active core is composed of 1020 fuel elements (FE) in 102 columns of 10 FE each. Replaceable reflector columns are also arranged in the inner (61 columns) and outer (102 columns) zones of the core annulus. A numbering system uniquely identifies each fuel or reflector element location in the core.

The fuel element residence time is about 900 Equivalent Full Power Days (EFPD) considering an average fuel burn-up at discharge of 120 GWd/t. With a uranium enrichment of 14% in mass, the fuel management is foreseen either in two or three batches. For the Pre-conceptual Phase, a two batch fuel management has been considered. The equilibrium cycle length is so 450 EFPD.

The duration of refueling outage (from full power to full power) in a 2 batches configuration is about 21 days.

Maintenance Durations for Large Components

Maintenance activities shall be included during refueling outages.

In Service Inspection (ISI) of the PHTS and PCS components and structures should be performed during refueling outages.

ISI of the Vessel System component or structure is also expected to be done during fuel handling periods (about one month). Radiological exposure to staff could create some concerns, to be further clarified.

Preliminary estimate of duration of the foreseen replacements:

- Main Primary Gas Circulator: one week
- Intermediate Heat eXchanger (IHX): one month.

An objective is to perform those replacements during a refueling operation. Given the expected durations, they would have no, or only a limited, impact on the plant availability.

Duration of unforeseen repair or replacement: cumulated duration shall be limited to 16 months for the plant lifetime. Failure frequency design value shall be lower than 0.01 per year.

The rationale for those durations is the following:

- The loss of capacity factor of 10% of plant lifetime corresponds to a period of 6 years. The refueling time is about 2.5 years, so the remaining time for unforeseen repair and replacements is 3.5 years, so 42 months. The 16 months duration above is deduced from a PWR Steam Generator replacement cumulated duration fitted for 60 years and applied to big components as IHX of the NGNP. This is compatible with that 42 months remaining duration.
- For a 20 years lifetime component as a plate-type IHX, the number of foreseen IHX replacements is 2. With the above failure frequency of 0.01, it can be postulated than one supplementary replacement will be necessary during the 60 years lifetime. With single replacement duration of one month, the total duration would be 3 months, which is lower than the 16 months duration required above. Therefore some margin exists either on failure frequency or on replacement duration.

Concerning the main primary and secondary (those included within the PHTS system) ducts, the current objectives are:

- Failure frequency significantly lower than 0.01 per year
- Repair or replacement duration limited to 12 months.
- The available duration for unforeseen replacement is 26 months (42 months – 16 months [for IHX and MPGC unforeseen repair/replacements] = 26 months]; this is compatible with the 12 months mentioned above, keeping in mind that the probability of such an event is low.

The potentiality to reach the above objectives will be confirmed in the next phases of the project.

14.0 AVAILABILITY

The development of the NGNP availability and availability allocation requirements were excluded from AREVA scope of work. However, AREVA considers issues associated with plant availability are highly important in the design selection of the NGNP prototype facility and the subsequent commercial plant. Therefore, this section was included to highlight AREVA views on the plant availability. Moreover, the prismatic reactor design proposed by AREVA has unique availability characteristics that must be considered; therefore this section was added to highlight these features of the AREVA design.

14.1 Availability Requirements

The NGNP as the demonstrator of the commercial viability of the nuclear heat source (H2 plant is outside of AREVA scope of this contract) shall be designed to the commercial plant availability targets. The starting point for defining the availability requirements and indicators of availability are the usual reliability and performance indicators for the power plant such as capacity factor, system and equipment reliability, planned outage and forced outage rate.

This section provides the NGNP prototype facility availability requirements for preconceptual design considerations:

14.1.1 The design capacity factor for electrical generation averaged over the plant's lifetime shall be at least 90% when modeled with equipment mean time to failure and mean time to repair data for the same or similar systems and/or components.

14.1.2 The design of structures, systems and components shared by two or more power units (for multi unit commercial plant design) shall incorporate features, such as redundancy and mutual support, to minimize the occurrence of multiple power unit outages resulting from failures or maintenance of the shared structures, systems and components.

14.1.3 The capacity factor loss due to planned outages averaged over the lifetime of the plant shall not exceed 8%.

14.1.4 The design shall allow all planned inspection and maintenance activities that must be accomplished with the reactor shut down to be accomplished within the period required for refueling.

14.1.5 The calculated capacity factor loss due to unplanned outages averaged over the lifetime of the plant shall not exceed 2% (average 175 full-power hours per year).

14.1.6 Unplanned outages of six months or greater shall not contribute more than 10% of the capacity factor loss from all unplanned outages, including those not expected to occur in an individual plant's lifetime.

14.1.7 The mean likelihood of exceeding the design limits associated with safety related design conditions, and which could, therefore, lead to the regulatory shutdown of this and other similar class of plants shall be less than 10 5 per plant year.

14.1.8 Radiological sabotage protection shall be provided by the physical barriers and features needed to satisfy the other plant functional and safety requirements, and by controlling access to vital areas in accordance with industry mandated security provisions.

14.2 Availability - Considerations Relevant to Pre-Conceptual Design

A first objective of availability has to be provided for commercial reasons and for starting the design process of the NGNP plant. For that purpose, a target value is chosen taking into account the competitors, the utilities requirements (EPRI, EUR) and also of what is considered for current generation of power reactors. Therefore availability target value of 90% was chosen for the Pre-Conceptual Phase. This objective has to be distributed to each system by allocating specific values for planned, unplanned (short and long durations) outages and reduced power operations on the basis of other plants experience feedback or of previous projects. Next phases of the NGNP project, more particularly the Conceptual Phase, will have the task to verify the appropriateness of keeping these objectives and to identify the impacts on the more mature design. A preliminary analysis has been performed for the ANTARES design of which the AREVA NGNP is an adaptation. The relevant results are discussed below.

For the electricity producing commercial plant, the allocations process takes account of the Nuclear Heat Source (NHS) and of the Power Conversion System (PCS) which includes the secondary circuit and auxiliary systems up to the grid connection. For the process heat production commercial plant, it takes account of the NHS and of the PCS only (includes the last circuit providing heat necessary to the process).

Input to the availability assessment is developed by performing a review of relevant availability analyses and availability oriented performance requirements for high temperature gas-cooled reactors and the competing concepts.

The first step in the availability and investment protection (AIP) evaluation is to define availability and reliability performance indicators that are used to express the plant availability and production oriented performance. The starting points for defining these indicators are the usual reliability performance indicators for power plants and process plants, such as availability, capacity factor, forced outage rate, etc. Definitions are provided below for the metrics used for the plant dedicated to commercial electric power production.

Availability Factor

The availability factor is defined as the expected fraction of time at which the plant is capable of producing net power output over a specific time interval of interest. During the preconceptual design phase, the availability factor may be assessed by the ratio of duration during which the plant is not shut down, with the operational phase duration of 60 years (i.e., starting after the construction and the initial commissioning phases and finishing before the decommissioning phase).

The duration of the shutdown phase during the plant life has to include:

- Normal planned shutdown for fuel handling operation and maintenance.
- Expected shutdown, not planned, due to abnormal events, including the delay for restoring the normal plant conditions.
- The shutdown possibly necessitated by particular event occurring on another power unit (e.g., unavailability of common equipment, particular operation during the construction, the commissioning of another power unit).

The availability factor target of 0.90 is related to NGNP power unit or each commercial plant module in case of a multi-module plant.

During operation, the plant availability may be assessed using the capacity factor:

Capacity Factor

The capacity factor is defined as the ratio of the actual energy produced in a given time period to the hypothetical maximum amount of energy which could have been produced. This hypothetical maximum was determined by multiplying the net full power rating of the plant by the time interval of interest. During the preconceptual phase, conservatively, it is assumed that the capacity factor should be very high. This means that, when the power unit is in operation (i.e., not in shutdown state) the generated power is the nominal power (nevertheless, duration and power evolution during start-up and shutdown transients have to be considered).

Therefore, the design capacity factor for electrical generation averaged over the plant's lifetime is at least 90 %.

Reliability

The probability that the plant will operate and produce rated power over the entire mission time, typically over the refueling cycle.

Plant Trip Frequency

The frequency of unexpected plant trips, usually expressed as the number of events per calendar year or per operating year. The plant trip can be an event or event sequence that leads first to a reactor trip and a consequential power conversion system trip, or some other event.

Plant trip frequency is expected to not exceed one per year.

Forced Outage Rate

The fraction of the time that a plant is down in a forced outage when it was scheduled to produce net electrical power; it is not actually a rate but rather a fraction of time.

Capacity Loss Factor

The fraction of energy production not realized due to planned or unplanned power reductions and shutdowns, as compared to that for 100% power operation for a specific period of time, usually expressed as a percentage. The capacity loss factor is sometimes broken down into contributions from planned and unplanned losses in capacity.

For unplanned outages:

- Capacity factor loss averaged over the lifetime of the plant is less than 5 %.
  - Unplanned outages of six months or greater contribute to less than 10% of this percentage.
  - Unplanned outages of two years or greater contribute to:
    - less than 10% of the unplanned outages of six months or greater
    - less than 1% of the unplanned short outages (lower than 6 months).

14.3 Availability - Allocation of Reliability Performance Targets

Allocations of availability and performance requirements to major plant systems for the AREVA NGNP are derived from available user requirements and ANTARES design specifications. Planned outages and forced outages allocations for the Nuclear Heat Source (NHS) and Power Conversion System (PCS) are preliminarily provided.

The events that normally limit the capability to meet availability and reliability performance requirements are:

- high frequency events that involve short to moderate plant outages or production curtailments
- low frequency events that may involve long outages and challenge the capability to protect the investment which do not necessarily involve safety consequences. These events could result in extended outages on the order of months to years and may lead to faulted conditions and an inability to restart the plant. They may pose a substantial risk to the economic viability of the plant even if they pose no significant public health or safety risks and are captured within the AIP evaluation
- other lower frequency events that may not contribute very much to plant availability or capacity factor performance, but are of concern to properly manage the risks of low frequency unplanned events. Events in this category with potential public health and safety implications are covered in the safety evaluation.

The allocation of reliability performance targets for a single reactor module is shown in Table 14-1 below. It has been done for electricity production plant. It should be envelope of the plant dedicated to process heat application for which the secondary circuit is not more complex. The values of Effective Full Power Days (EFP) are given per year, this explains the decimal values.

Table 14-1: Reliability Performance Targets for NOAK[23] Module

Contribution to Capacity Loss Factor (CLF)	Nuclear Heat Source (NHS)		Power Generation System (PGS)		Overall Reactor Module
	Target Effective Full Power days/yr	Capacity Loss Factor (CLF)	Target Effective Full Power (EFP) days/yr	Capacity Loss Factor (CLF)	
Refueling	< 13.7	3.8%	/	/	
Other planned outages	< 2.4	0.7%	< 2.3	0.6%	
Unplanned outages – short duration (< 6 months)	< 5.0	1.4%	< 10.	2.7%	
Unplanned outages – long duration (> 6 months)	< 0.5	0.14%	< 1.0	0.3%	
Reduced Power Operations	< 2.0	0.6%	< 4.0	1.1%	
Unplanned Capacity Loss Factor		2.0%		4.1%	6.0%
Forced Outage Rate		1.5%		3.0%	4.5%
Capacity Factor		93.6%		95.3%	89%
Availability Factor		94.2%		96.4%	91%

This allocating first exercise for the Nth of a kind plant nearly leads to a 90% **capacity factor** and **availability factor** as required in the System Requirements Manual [REF. 1]; the objective in that preliminary assessment is to achieve 90% of capacity factor. This is deduced from a review of service experience with currently licensed LWRs and specifying requirements for advanced reactors.

A review of the advanced reactor sources listed above yielded the comparison in Table 14-2 below between the ANTARES performance goals and those for other advanced reactors.

Table 14-2: Comparison of ANTARES Goals to Advanced Reactor Requirements

	NGNP per Table 14.1	GT-MHR Requirements	EPRI ALWR	GT-MHR Design Assessment
Planned Outages	5.0%	6.0%	11.8%	(1)
Unplanned outage short duration	4.1%	6.3%	1.4%	(1)
Unplanned outage long duration	0.4%	0.7%	(1)	(1)
Reduced Power Operation	1.6%	(1)	(1)	(1)
Total CLF	10.7%	13.0%	13.1%	(1)
Total Capacity Factor	89%	87%	87%	84%

(1) Breakdown or value not available

The above comparison indicates that the performance goals in Table 14-1 may be somewhat ambitious, but yet are in reasonable agreement with comparable goals that have been set. It is reasonable to expect that the AREVA NGNP has the potential of superior performance to the GT-MHR as the former is a direct cycle machine and will need to demonstrate the reliability and maintainability of a new turbo-machine in a radiation field environment. This may be offset somewhat by the higher complexity of the combined cycle concept employed by the NGNP.

15.0 FUEL STRATEGY

The TRISO fuel development, qualification, and production program must ensure the following high-level objectives of the NGNP [24] are met:

1. Develop and implement the technologies important to achieving the functional performance and design requirements determined through close collaboration with commercial industry end-users.
2. Demonstrate the basis for commercialization of the nuclear system, the hydrogen production facility, and the power conversion concept. An essential part of the prototype operations will be demonstrating that the requisite reliability and capacity factor can be achieved over an extended period of operation.
3. Establishing the basis for licensing the commercial version of NGNP by the Nuclear Regulatory Commission. This will be achieved in major part through licensing the prototype by NRC and initiating the process for certification of the nuclear system design.
4. Fostering rebuilding of the US nuclear industrial infrastructure and contributing to making the US industry self-sufficient for our nuclear energy production needs.

For NGNP to be fully successful in achieving the identified high-level objectives, a fuel development strategy has been formulated that utilizes and expands existing commercial fuel facilities, and enables the NGNP program to meet the anticipated reactor performance requirements, which will demonstrate the basis for commercialization of the nuclear island, as well as establish the basis for licensing the reactor and fuel system by the NRC, and developing a domestic fuel supply.

Equally important to meeting these high-level objectives, the strategy developed meets the anticipated delivery requirements to support reactor startup in the 2018 timeframe, assuming the start dates are met and funding profiles identified are provided.

Fuel related requirements and key parameter values have been specified in two top-level NGNP project requirements documents, the System Requirements Manual [25] and the NGNP Prototype Design Baseline [26]. These documents specify that the base fuel for the NGNP will be a TRISO coated particle containing a less than 20% enriched fuel kernel. These particles will be fixed into cylindrical, graphite matrix compacts, which are, in turn, placed into hexagonal graphite blocks.

In addition to these requirements and values, expected fuel performance characteristics will eventually be defined by required plant radionuclide release performance under operational and accident conditions to meet regulatory offsite and worker dose limits. The limiting radionuclide releases associated with the key accident analyses have not yet been determined. As such, the NGNP plant specific required fuel performance characteristics have not yet been defined. Until these are defined, the following general requirements should be utilized.

The ANTARES design assumes a reference fuel form of UO2 SiC TRISO fuel. This was selected to minimize risk of development and best enable new fabrication capability in France. The NGNP has asked for broader fuel performance capability and encourages a domestic source for the fuel, as well as seeking to foster continued development in key HTR areas. Therefore, though the least risk recommendation would be expanding existing US capability to manufacture TRISO fuel using a UO2 kernel, this path might not fully meet all of the NGNP long term goals. Because the time to develop, design, and construct the NGNP allows adequate time to develop UCO based TRISO fuel, we have looked at both UO2 and UCO. There does not appear to be a significant time difference in preparing either, therefore, the baseline schedule shown in this chapter assumes UCO. It is still AREVA's opinion that pursuing UO2 may have lower overall risk, and this fuel type must be included in early fuel irradiation activities along with UCO to provide a robust backup.

Table 15-1: General Fuel Quality Requirements

As-Manufactured Quality Requirements	**Failed Particles/Particle**
Allowable defects* measured at the time of manufacture	$\leq 5.0 \times 10^{-5}$
Contamination of free uranium in fabricated fuel	$\leq 1.0 \times 10^{-5}$
In-Service Operational Requirements	
Allowable failure of fuel particle coatings during normal operation	$\leq 2.0 \times 10^{-4}$
Allowable incremental failure of fuel particle coatings during off-normal events	$\leq 1.0 \times 10^{-6}$

* Fuel defects are particle layer failures or those conditions that would reasonably lead to particle layer failure under normal operation and accident conditions.

15.1 Fuel Particle and Fuel Compact Basis

15.1.1 Fuel Design

At this point in the preconceptual design process it is not considered necessary to specify explicit design details of the fuel particle, fuel compact, and fuel element. However, if the fuel qualification and fabrication schedules presented in this document are to be met, such parameters must be defined fairly early in the conceptual design process.

While it may be possible to meet the NGNP fuel quality and performance requirements utilizing a standard TRISO particle fuel design, that is, a UO_2 kernel surrounded by a low density carbon buffer, an inner pyrolytic carbon layer, a silicon carbide layer and an outer pyrolytic carbon layer; operation to these limits would be at or near the maximum capability of this fuel type. As such, this design may ultimately prove unsuitable for operation under the conditions required to help assure commercial viability. For this reason, use of an advanced fuel design, such as UCO, that would be better able to meet the performance goals should be considered. UCO exhibits significantly lower oxygen release at low burnups, which increases its overall performance envelope, providing margin to failure from under NGNP conditions.

There are some outstanding questions, regarding the ability to implement the UCO design, including all required development, testing, qualification, and licensing activities in a manner consistent with the desired NGNP 2018 deployment schedule. Because of its past operational history and potentially easier licensing path, the standard UO_2 TRISO fuel design should be considered as a fallback fuel choice for early qualification activities.

In order to maximize flexibility, qualification of both the UCO and UO_2 fuel kernel designs should proceed in parallel for as long as budget constraints allow. Given these considerations, for the NGNP, it would be appropriate to consider use of a UCO fuel kernel as the baseline fuel choice, with a UO_2 kernel as an option for early cycle use, only if necessary.

AREVA has selected UCO as the baseline kernel. This selection is due principally to the enhanced burn-up characteristics UCO has over UO_2. This significant increase in burn-up potential improves the economic case for the NGNP on a commercial scale, though at the expense of some additional risk during the developmental phase of the project. Should unforeseen complications arise from the continued development of the UCO kernel, UO_2 would be considered as a backup kernel for the first NGNP fuel load. This backup position should be considered only as a last resort if the performance of the UCO is not as expected.

15.1.2 Fabrication Technology

The NGNP fuel will consist of uranium oxycarbide (UCO) or uranium dioxide (UO_2) kernels coated with a low density graphitic buffer layer, a high density pyrolitic a graphite layer (IPyC), silicon carbide (SiC), and finally another high density pyrolitic graphite layer (OPyC). These fuel kernels are mixed with graphite powder and a resin and pressed into a right cylindrical compact approximately 1 cm in diameter and 4 cm long. Finally, the fuel compacts are inserted into graphite fuel blocks which are shipped from the fuel vendor and placed into the reactor.

15.1.2.1 Fuel Kernel Options

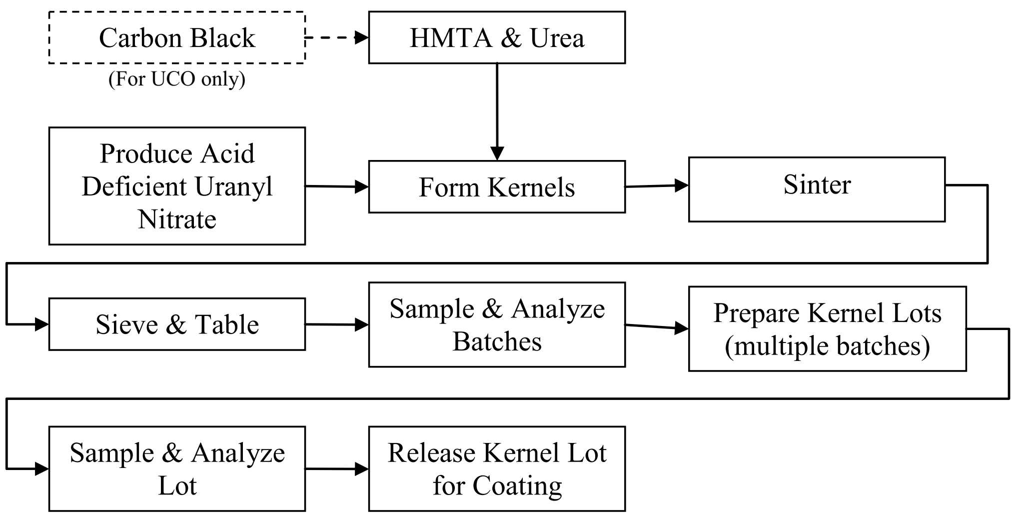


Figure 15-1: NGNP Kernel Process

The kernel fabrication process for the NGNP project is flexible to be able to develop and produce the two potential kernel compositions: uranium oxycarbide (UCO) and uranium dioxide (UO_2). The processing difference between the two compounds is simply the addition of a carbon source to the broth mixture for the UCO kernels. The fabrication process is shown in Figure 15-1.

There are a few processing parameters that differ slightly based on the kernel composition. The most significant of these differences is the sintering profile needed to achieve the desired kernel chemical composition and density. For the UO_2 kernel, the sintering profile is simply a reduction of the as-formed UO_3 kernel to UO2, which nominally occurs between 300oC and 500oC in a reducing atmosphere such as hydrogen. After reduction, the kernels are heated to high a temperature, e.g. 1200oC, to densify the kernel to meet the specification.

The formation of UCO is more complex. As with UO_2, the as-formed UO_3 is reduced to UO_2 at low temperatures. Following reduction, the kernels are heated to a temperature where the carbon will react with some of the UO_2, forming a low dense UCO structure with a byproduct of CO. The low dense structure is heated further to densify the kernels to meet the specification. Figure 15-2 shows a typical UCO kernel, after the sintering operation is complete.

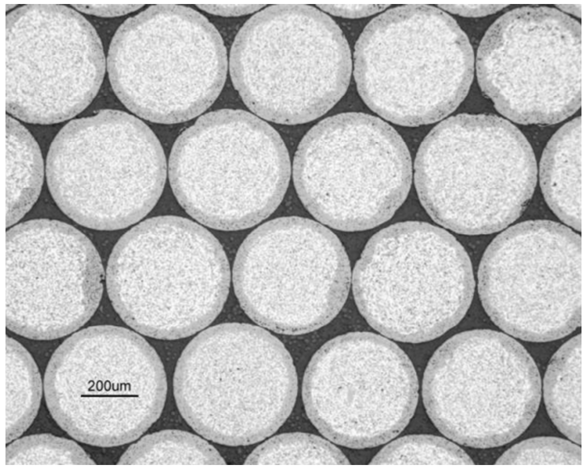


Figure 15-2: UCO Kernels

Each batch of kernels, regardless of composition, is sieved to remove over and under sized and broken material. In addition, the kernels are passed over a "tabler" to remove misshapen or broken material. The batches are sampled and characterized for chemical and physical attributes required to be confirmed at the batch stage.

Several completed batches are blended into a homogeneous kernel lot and sampled and analyzed for chemical and physical attributes that must be confirmed in the kernel lot stage. Once acceptable, the kernel lot is split into uniform batches for the TRISO coating operation.

15.1.2.2 TRISO Coating

The fuel TRISO coating development efforts to date, and those proposed for the NGNP, focus on replicating the performance characteristics of the German HTTR fuel. Over the past several years, a comparison of the as-fabricated German fuel has been made to similar fuel types fabricated in the US. Several differences were noted including the SiC microstructure, pyrolytic graphite anisotropy, and morphology of the interfaces between the layers.

In an attempt to mimic the German fuel characteristics, recent TRISO coating development was focused on a continuous coating operation, where all layers are applied in one furnace run. In addition, process conditions are being replicated to the maximum extent possible, taking into account differences in furnace configuration, etc.

The coating development was initiated with a small laboratory-scale retort having an inside diameter of two inches. Once the parameters were established using the 2 inch coater in a continuous manner, the system was upgraded to accommodate a 6 inch retort and correspondingly higher input charge size. Although the final charge size has not been determined to date, it is expected that between 2,000 grams and 2,500 grams will be able to be processed in a single batch. Once the processing parameters are established using the 6 inch coater, and during the conceptual design of the NGNP reactor, and the design of the fuel fabrication production facility, the determination as to the retort size of the production coater will be made. Several factors will go into making that determination, including economics (both cost to produce a batch of particles as well as capital cost for the furnace of a given size) and overall risk (scale-up effects) of choosing a retort significantly larger than the 6 inch coater demonstrated during the current development efforts.

The multiple layers of the TRISO coatings are sequentially applied to the kernels in a fluidized bed furnace. The furnace temperature and atmosphere are varied, depending on the layer being applied. Table 15-2 identifies the different gases used to achieve the desired coating.

In addition to a robust gas delivery and control system, the furnace must also have an off-gas system to capture undeposited graphite of the carbon deposition steps, and neutralize the HCl byproduct produced during the silicon carbide deposition step.

Table 15-2: Coating Atmosphere

Coating Layer	Atmosphere
Buffer	Acetylene & Argon
Inner Pyrolitic	Acetylene, Propylene, Argon
Silicon Carbide	Methyltrichlorosilane (MTS) & Hydrogen
Outer Pyrolitic	Acetylene, Propylene, Argon

The completed batches are sieved, tabled, sampled and characterized to ensure the particles meet the specifications on a batch basis. Acceptable batches are blended into a homogeneous lot and again sampled and characterized for lot attributes. All data is compared to the fuel specification, statistical analysis performed, and the lot certified as being acceptable for compact fabrication.

15.1.2.3 Compact Fabrication

Compact fabrication is accomplished using the completed, certified particles, mixed and heated with a thermosetting resin and graphite matrix powder.

The matrix and resin are mixed off-line and pulverized to form a homogeneous powder. A prescribed quantity of particles and matrix are mixed together and poured into a mold cavity. Slight pressure is applied to the mixture by inserting the top punch and applying a preset pressure. The mixture is heated to a specified temperature for a predetermined period of time. After the heating cycle, the die is either actively (using water or other heat dissipating coolant) or passively cooled. Once the die reaches approximately room temperature, the compacts are ejected from the die.

As-pressed or “green” compacts receive two heat treatments. First the compacts go through a low-temperature debinder heat treatment to remove the unreacted binder material and any byproducts. Next the compacts are heat treated to ensure adequate performance during irradiation while minimizing the thermal impacts on the contained particles.

A predetermined number of compacts are destructively evaluated to ensure the lot of compacts meet the fuel specification on a statistical basis. Once the chemical and physical attributes of the compacts has been confirmed, the lot will be certified and released for fuel block fabrication.

15.1.2.4 Fuel Block Fabrication

The graphite blocks are fabricated off-line using qualified graphite. The blocks will be inspected and released for use.

The certified compacts will be placed into the fuel holes and end plugs inserted using a qualified method. The loading of the blocks will be verified real-time by quality control over-inspections.

15.2 Fuel Fabrication Strategy

The steps required for development and qualification of UO2 fuel and UCO fuel are essentially identical. The plan discussed in this chapter mentions UCO fuel form, since questions have been raised regarding the development path for UCO. However, the steps would be the same if the selected fuel form is UO2. Similar steps would be followed for an initial UO2 program to be subsequently complemented with UCO, an all UCO program, or a parallel UO2/UCO program.

Until a detailed design for the NGNP is complete, the quantity of fuel needed for the first core and reloads is not clearly defined. For this preconceptual design report, it is assumed that the initial core will contain 5000 kg of uranium in the form of TRISO coated UCO particles. In addition, it is assumed that half of the core will be reloaded every 18 months.

To support the NGNP reactor long-term and minimize capital expenditures, a fuel facility needs to be constructed that can meet the desired reload schedule. For this report, the facility output is assumed to be 2,000 kg uranium per year. This quantity meets the 2,500 kg uranium per 18 months with a 20% excess capacity to accommodate production upsets which may occur during initial commissioning of the facility. Table 15-3 defines the requirements for each processing unit within the facility.

The batches per year and per day are based on 200 operating days per year, three shifts per day. This basis may change as the details of the processing are determined and downtime defined to accommodate nuclear material accountancy, equipment maintenance, process setup, etc.

Table 15-3: NGNP Fuel Facility Process Requirements

Process	Assumed Yield	Batch Size (kg U)	Batches	
			Per Day	Per Year
ADUN	95%	36	.4	69
Kernel Formation	94%	12	1	197
Wash/Dry	99%	12	1	185
Kernel Sinter	98%	4	3	546
Kernel Upgrade	97%	4	.5	535
Coating	94%	2	6	1037
Upgrade	95%	2	6	975

15.2.1 Overall Fuel Supply Options

Several companies around the world have expressed capabilities, or interests in developing the capability to supply fuel for the NGNP project. Based, on open literature, several fuel supply options are identified:

- US (BWXT/AREVA)
  - Pilot-scale facility available with approximately 200 kg’s uranium throughput capacity
  - Any enrichment
  - Demonstrated UO_2 or UCO capability
- Japan (NFI)

    - Existing facility supporting HTTR research reactor
        - 1500 kg U capacity
        - 2 coaters with 3.4 kg U capacity
    - UO_2 only
    - <10% enrichment limitation
    - <5% FIMA demonstrated
- South Africa (PBMR, Pty.)
    - Facility being constructed
    - UO_2 only
    - <10% enrichment limitation
- Russia (VNIINM, SCC, etc.)
    - Unknown capabilities
- China (HTR-10/Tsing Hua Univ.)
    - Relocated German particle line to support HTR-10
    - Fabricating pebbles for HTR-10
    - Unknown capacity or quality

Although several identified facilities have existing capacity to support their needs, there does not appear to be significant excess capacity to take on the additional needs of the NGNP. Therefore, the facility construction, commissioning, licensing, and qualification steps that must be undertaken for a US supplier, must also be undertaken for existing international suppliers as well.

For the NGNP to fully meet the high level objectives, the fuel fabrication facility must:

1. demonstrate the performance of fuel fabricated in existing facilities to meet the enhanced requirements of the NGNP;
2. qualify fuel fabrication capabilities in existing facilities;
3. upgrade or construct a production facility capable of meeting the reload schedule of NGNP;
4. verify fuel performance from the production line;
5. support NRC licensing of production facility; and
6. produce NGNP fuel by the project delivery date.

For these reasons, we believe that the preferred overall fuel supply option is to support development of a BWXT/AREVA domestic fuel fabrication facility.

15.2.2 First Core Supply

To support the programmatic schedule for the NGNP, namely startup in 2018, a fuel development, qualification, and production strategy has been developed as part of the NGNP preconceptual design study. This strategy utilizes the existing infrastructure available at BWX Technologies in Lynchburg, VA to the maximum extent possible.

Two options for the fuel supply strategy have been developed and will be discussed.

15.2.2.1 Option 1

The first option to supply fuel for the NGNP is to utilize the existing pilot facility to produce the performance test fuel. The existing facility is licensed under the regulatory authority of the NRC and has recently successfully completed the tri-annual audit of the NQA-1 quality program.

This fuel will be tested under operating and accident conditions to demonstrate the performance characteristics of the fuel fabricated using current-technology, in near-production sized equipment, including a 6” coating furnace.

While the performance test fuel is being irradiated and after some burnup has accumulated, qualification fuel will be fabricated and pressed into compacts for qualification testing. For this qualification, several more compacts will be fabricated to ensure a statistically significant number of particles are tested to demonstrate performance and qualify the manufacturing and inspection techniques.

During the performance test fuel fabrication and irradiation program, the design and construction of the manufacturing facility will begin. The production facility will utilize the technology employed in the pilot facility, but scaled-up and enhanced for continuous high-volume throughput.

To meet the schedule objectives, the production facility needs to be complete, commissioned, and in production by mid-2012.

Figure 15-3 graphically shows the flow of Option 1.

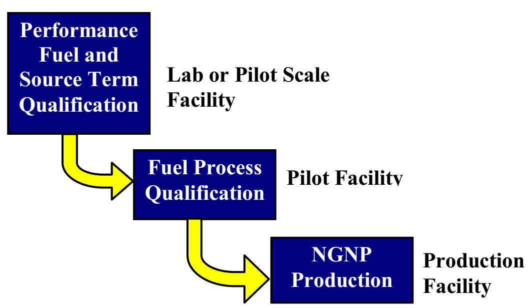


Figure 15-3: Option 1 Process Flow

The key activities needed to be performed to meet the program’s objectives include:

- Complete coater upgrades of the 6” furnace and demonstrate acceptable coating can be applied
- Implement compact fabrication technology in Lynchburg using AREVA and Oak Ridge National Laboratory experience
- Fabricate performance test fuel and begin irradiation testing
- Fabricate qualification fuel and begin irradiation testing
- Design and construct production facility based on pilot-scaled facility
- Validate production facility fuel quality by fabricating fuel particles and compacts and irradiation test to confirm performance

In order to be fully successful using this option, some programmatic risk must be assumed. For example, qualification fuel must be fabricated before the performance test fuel has completed its irradiation and post-irradiation examination. The technical risk to the program, should this strategy be implemented, can be somewhat mitigated by the design of the irradiation test, i.e., the capsule design will permit real time in-core fuel failure monitoring which will give an indication of performance during normal operating conditions.

Another opportunity to accelerate the schedule is to design and construct the fabrication facility based on the fabrication experience gained and the preliminary in-core results of the performance test fuel. The risk to the program is judged to be very small. A significant portion of the schedule acceleration is in the design and construction of the production building itself. The overall size will be unaffected by the performance of the irradiation tests. Several of the long-lead equipment items, likewise, will be unaffected by the preliminary results of the performance test fuel. Minute processing details which could be affected by the irradiation test results have short delivery times and minimal cost. Therefore, these parts of the equipment can be made last-minute if necessary, or remade with minimal cost should the need arise.

The final opportunity to accelerate the schedule under Option 1 is to "qualify" the fuel fabrication process using the pilot-scaled facility, including the 6" coating furnace. Then, once the production facility is completed and commissioned, a "verification" run will be made to prove the performance of the fuel fabricated in the production facility matches the performance of the qualification fuel fabricated in the pilot facility.

Figure 15-4 shows a high-level schedule of what the fuel development/qualification/ production strategy would be should Option 1 be followed. Note that the dashed lines are used to schematically represent data flows between activities and do not represent the expected timing of the data transfer.

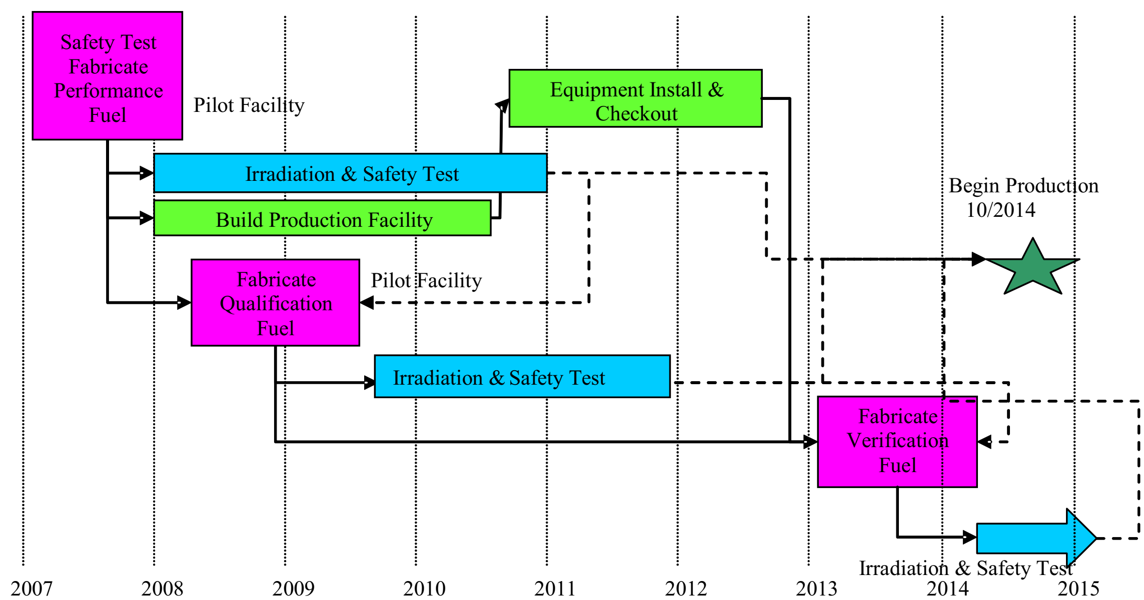


Figure 15-4: Option 1 Program Schedule

Although Option 1 meets the first core delivery milestone of 2018, one significant problem is the four year gap in fabrication experience between the qualification fuel and the verification fuel manufactured in the production facility. During this time, the qualification irradiations are being performed and the production facility is being constructed.

This gap in manufacturing experience is believed to increase the overall risk to the program because:

- Experienced operators will loose their skills if not practiced for four years
- Opportunities to improve the process and develop alternative procedures will not be available
- Inactive facilities may be decommissioned, eliminating the possibility to reproduce materials in pilot-scaled equipment.

15.2.2.2 Option 2

A second fuel strategy option has been developed that mitigates the risks associated with the gap in manufacturing identified in Option 1. This second option utilizes the existing pilot-facility, with some upgraded equipment to achieve a higher nominal throughput, to perform early fuel production until the production facility is available. The process flow associated with this option is shown in Figure 15-5.

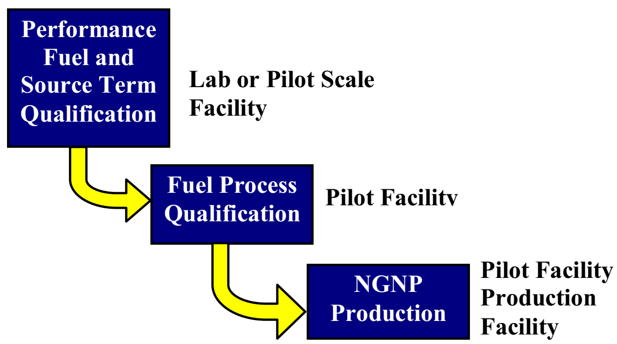


Figure 15-5: Option 2 Process Flow

The current particle fabrication pilot-scaled facility is capable of producing between 150 kg and 200 kg, on a uranium basis. This facility, as currently configured, will produce the performance test fuel in prototypic equipment. In addition, qualification fuel would be fabricated and prepared for irradiation testing.

While the performance test and qualification fuels are fabricated and tested, the pilot facility capabilities will be augmented to increase the throughput. It is envisioned that the kernel wash/dry system will be upgraded to double its capacity. To accomplish this increase, the wash/dry batch system could simply be duplicated, or alternatively, looking forward to the production facility, the general equipment arrangement and processing methods can be modified to be more production-worthy.

In addition to upgrading the kernel wash/dry system, in order to be able to significantly increase throughput, additional coating capability must be installed. To achieve this, an existing fluid bed sintering furnace will be converted to a coating furnace. In order to replicate the current coating furnace, the sintering furnace will need to have the gas injector and retort replaced, as well as the TRISO coating gases plumbed to the furnace. The furnace control system will be upgraded to accommodate the coating process.

Once the sintering furnace is converted to a coating furnace, a new sintering furnace will be procured and installed. The type of furnace will depend on the kernel design selected for the NGNP. If UCO is selected, a fluidized bed furnace will be constructed. If UO2 is selected, a static bed pusher furnace will be utilized. Note that the flexibility of the fluidized bed can accommodate the sintering of UO_2. Type selection is a question of economics. The use of a batch furnace is more costly per kilogram throughput than a continuous pusher-type furnace.

In addition to kernel and particle fabrication capabilities, the pilot facility will require installation of a compact fabrication facility. This facility will require the production of approximately 1000 compacts per day, which will require a multi-cavity mold, matrix production equipment, furnaces for debinding and high temperature heat treatment of the compacts.

This upgraded pilot-facility will be capable of producing approximately 500 kg of uranium in the form of completed compacts. This production, started shortly after the qualification fuel is fabricated and irradiation testing initiated, will significantly augment the production facility throughput. The net affect of using the pilot facility for limited production is to mitigate the risk associated with construction and startup of the production facility. In addition, the engineers and operators will remain trained in the art of fuel fabrication while the production facility is being constructed. This continuation of fuel fabrication will enable minor procedural and equipment adjustments in the production facility prior to and during the startup phase of the construction project.

The Option 2 program schedule is shown in Figure 15-6. As in the previous option diagram, the dashed lines are used to schematically represent data flows between activities and do not represent the expected timing of the data transfer.

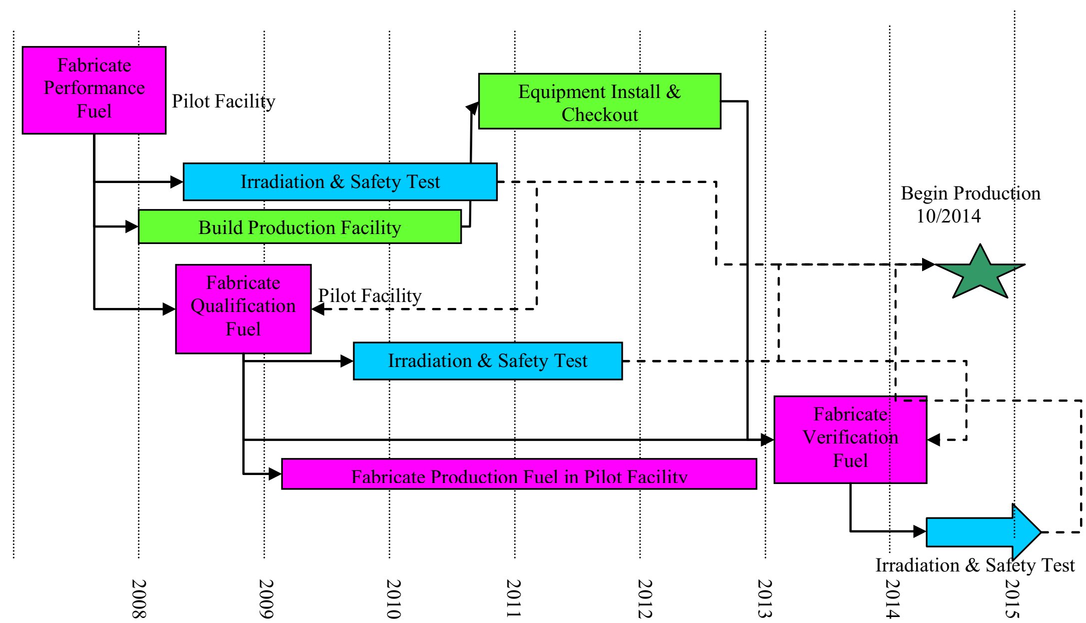


Figure 15-6: Option 2 Program Schedule

Various production scenarios have been developed to assess the impact of using the pilot facility to augment the first core production. These scenarios are shown in Figure 15-7. On an optimistic approach, assuming the pilot facility begins "production" processing in January 2009 and the production facility is commissioned and operational in January 2015, the full core load of fuel will be available in the 4th quarter of 2015, approximately two years ahead of the scheduled startup of the reactor. On a worst case, assuming the pilot facility upgrades are delayed 18 months beyond the optimistic case, and the production facility is delayed approximately one year, the core load is delivered during the fourth quarter of 2017, still in time to support the fuel loading schedule and reactor startup.

In all these scenarios, it is assumed that the pilot facility production will cease once the production facility is running and has demonstrated fabrication capabilities. This assumption is for economical considerations in that fuel produced in the production facility will be cheaper to produce than in the pilot facility, due primarily to the larger batch sizes anticipated will reduce the manhours per kilogram estimated.

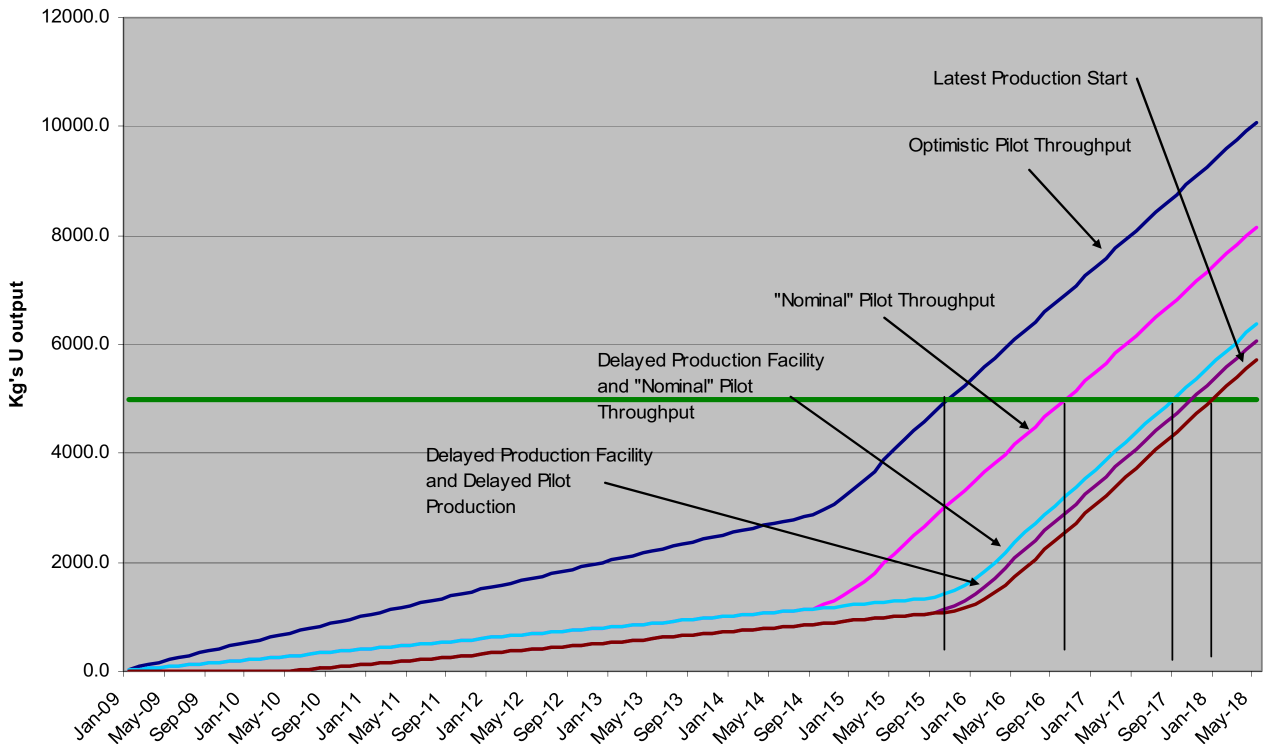


Figure 15-7: Production Throughput Scenarios Using Pilot Facility

Several advantages can be gained by utilizing the pilot facility for the initial production of NGNP fuel:

1. Potential to deliver first core up to two years ahead of the 2018 reactor startup date
2. Engineers and operators remain trained in the state of the art for fuel fabrication
3. Advances in fuel production techniques developed during the time the production facility is being constructed can be implemented in the production facility

Unintended results in early irradiation testing can be worked through in the pilot facility and factored into the production facility with minimal or no schedule impact, i.e., this approach can absorb up to 2 years in schedule slippage due to unforeseen test results

Delays in the commissioning of the production facility can be minimized due to the 500 kg output from the pilot facility

15.3 Fuel Qualification Plan

In order to meet the fuel performance requirements that would allow reliable operation under specified NGNP operating conditions, fuel with fabricated quality and operational characteristics at least as good as past German particle fuel will be required. There are two approaches that can be considered to meet this goal. Past German fabrication processes, practices, and equipment can be replicated, to the extent possible, in the hopes of producing product with similar characteristics. Alternatively, past German experience can be examined and key fabrication concepts coupled with modern fabrication techniques. AREVA/BWXT has chosen the latter approach for its fuel qualification program. Though this approach may require a more rigorous testing and qualification program, it will allow easier extrapolation to operational conditions beyond the German experience.

The NGNP operational date of 2018 presents significant challenges to the development of a fuel qualification program. In order to meet this date and conduct the irradiation and testing necessary to support the plant safety case, the fuel qualification must be success-based. That is, the steps are defined with the assumption that each irradiation will be successful and that acceptable fuel performance will be demonstrated at each step. Based on the successful past particle fuel fabrication experience of BWXT, the recent fabrication research conducted by AREVA and the fairly robust fuel particle design chosen for the NGNP plant, AREVA/BWXT believes that this qualification program will yield acceptable results.

There are three fabrication-irradiation-test sequences envisioned for the fuel qualification program. In each case, coating will be done in a 6" coater to baseline conditions, and a range of certain variables to provide a range of fabrication conditions, using BWXT fabrication processes. Coated particles will be certified to a NGNP fuel specification. Compacts will be fabricated to baseline conditions in the pilot facility in Lynchburg using the AREVA compacting process. Each of the irradiation capsules used during these sequences will be monitored to detect particle failure such that appropriate actions can be taken on a timely basis.

The first test sequence is designed to provide an early indication that the fuel and testing equipment will perform as expected. A smaller quantity of fuel would be irradiated in this sequence, since no statistical inferences will be drawn from the results. It is in this sequence that consideration should be given to irradiation of some UO2 particles as backup to the UCO fuel. Should the UCO fuel fail to perform as expected, the NGNP startup date of 2018 would still be in jeopardy, but the head start provided by the presence of the UO_2 would mitigate the schedule impact. All of the fuel in this sequence would be fabricated in the pilot facility.

The second sequence is designed to provide the data used to qualify the fuel for use in the NGNP plant. A quantity of fuel would be fabricated, irradiated, and inspected that would yield the statistics required to demonstrate that the fuel supports the plant safety case. This fuel would also be fabricated in the pilot line. It is envisioned that several batches would be made and blended to form a homogeneous lot upon which the results would be based. This process will be used to as closely as possible reflect anticipated commercial scale fabrication techniques. The statistical basis and acceptance criteria for the test will reflect this processing technique.

The last sequence will provide the data necessary to verify that the commercial production line is capable of reliably and repeatably producing fuel with the same performance characteristics as that produced on the pilot line. The majority of fuel for this sequence will be fabricated on the commercial production line, though some limited quantity of fuel from the pilot line may be included to provide the opportunity for direct comparisons of particle performance. The quantity of fuel contained in this sequence will be determined by the extent of difference between the pilot and production lines.

15.4 Fuel Feasibility Issues

The NGNP program, as envisioned by AREVA, will establish a production-worthy facility with a robust fuel fabrication process which will be capable of reliably delivering finished fuel to the NGNP on the desired reload schedule. The processes selected, including the internal gelation process for the kernels, continuous coatings, thermosetting resins for the compacts, etc. have been demonstrated to produce high quality fuel. The fuel fabrication system will be able to be expanded as necessary, by adding modules to the production facility, as the demand for high temperature gas cooled reactors, and subsequently the need for fuel increases in the coming years.

When evaluating the options to supply fuel on a long-term basis, the AREVA/BWXT team is uniquely qualified to meet all of the objectives of the NGNP fuel program. Building on the current development and scale-up activities, the team will be able to implement a strategy utilizing the existing pilot fuel facility while constructing and commissioning a production facility capable of meeting the reload needs of the NGNP reactor. The approach identified herein will deliver first core on time with the potential to deliver the core several years early, as well as be in a position to deliver the reload fuel on schedule.

In addition to meeting the scheduler needs of the program, the strategy developed by AREVA and BWXT meet all of the objectives identified in the Preliminary Program Management Plan, namely development of a fuel system that will allow the NGNP to demonstrate all performance aspects of the plant, including economic feasibility of a fuel system capable of higher burn ups.

16.0 ECONOMIC ASSESSMENTS

Per BEA/INL instructions, this section will be provided to BEA/INL separately.

17.0 PROJECT SCHEDULE

This chapter presents the development and initial assessment of the schedule for the NGNP prototype facility envisioned by the AREVA team at the preconceptual design stage. The NGNP project schedule encompasses the total time span beginning with the project conceptual design phase, concurrent supporting research and development activities identified in this report, licensing and permitting activities necessary to obtain an operating license, plant construction activities, long lead component procurement including steps to be taken for key hardware acquisition, testing and initial operation, leading to commercial plant demonstration and operations. Although not shown on the project schedule are the commercial operation years and a final six year decontamination, dismantling, and disposition (DD&D) phase. The NGNP high level project schedule is shown in Figure 17.1. More detailed project schedule is provided separately.

Key elements of the AREVA NGNP project schedule consist of:

- Research and Development
- Plant Design
- Licensing and Permitting
- Procurement
- Construction
- Initial Startup and Commissioning
- Commercial Operations (not shown)
- Decommissioning (not shown)

17.1 General Approach

The following top level project requirements are the prime drivers for the AREVA NGNP project schedule:

- The NGNP project begins in 2008
- The plant shall achieve initial criticality in 2018
- The Plant shall be licensed by the NRC as a prototype test reactor and subsequently the license shall be converted to a commercial nuclear operating license
- The NGNP prototype facility shall be located at Idaho National laboratory old NPR reactor site

In addition to the above top level schedule requirements and based on AREVA's past experience with reactor design and the current design option studied (i.e. NGNP reactor based on ANTARES design) a ten year schedule has been developed that includes focused R&D, Design, Licensing, Construction and Initial Operations and Inspection before the start of commercial operations.

The following sections provide discussions and assessment various elements of NGNP projects and as they impact the project schedule.

17.2 Research and Development

The research and development is the key element of the NGNP prototype facility success. During the preconceptual design phase AREVA NGNP design team has identified several major R&D needs to support the successful design and implementation of the NGNP constrained by the top level project requirements. Chapter 19 of this report details the R&D needs.

As part of the pre-conceptual design effort, R&D needs were identified along with top-level cost and schedule estimates for addressing each need. A detailed R&D task plan with task linkages was beyond the scope of this effort. And without that detail, it is difficult to determine the R&D impact on the overall NGNP schedule objectives.

Nevertheless, some observations are possible:

- Materials development and qualification is estimated to take the longest – up to 72 months.
- Components' testing is estimated to take up to 42 months to complete. Efforts taking the longest require construction of a large He test facility (48 months) followed by a series of component testing currently estimated at 12 months each.
- Fuel development and qualification has a number of efforts requiring from 12 months to 36 months to complete.
- Computer code/methods development and qualification efforts range from 6 months to 60 months, but most efforts are complete by 36 months.
- Power conversion system needs can probably be satisfied within 18 months,

It appears that the above five categories of efforts can be initiated at the same time and proceed in parallel if adequate funding is available. However, without a detailed plan within each category, it is not clear if certain efforts have to be done in series which could lengthen the time estimates.

It is probably most important to initiate the materials, fuel, and computer code efforts first since information coming from these efforts is important to completing design efforts and building components. Construction of the He test facility also needs to be initiated but prototype components can not be tested until they are available which will be later in the project.

17.3 Design, Procurement and Licensing

The NGNP design, procurement, and licensing is highly interlinked as shown in Figure 17.1. The project schedule relies on achieving design goals, action on the long lead procurement, and licensing successes described in this section.

17.3.1 Design

The NGNP design phase is divided in three industry standard phases: 1) Conceptual design, 2) Preliminary design, and 3) Final design. A two year time span is envisioned for each phase to meet the target startup date. The six year of design is considered aggressive but achievable. Its success is dependent on design concept selection, project continuity and funding stability, success of expected performance and data needs driven by the R&D activities, the NRC acceptance of NGNP licensing strategy and expected progress in development of the licensing submittals.

The NGNP schedule provided in this report includes added detail for the conceptual design phase, i.e. the first two years of the design activity. However, all elements of the design phase activities are standard industry practices. In particular, AREVA design phase activities are primarily driven by the well known standard System Engineering principals.

Details of NGNP prototype facility design are provided in Chapters 6, 7 and 8. The schedule impact of the design choices, component design, material qualification, and long lead procurement should not be under-estimated. The current design schedule maintains a reasonable assurance that schedule risks identified in Chapter 18 are manageable.

17.3.2 Long Lead Procurement

The high temperature gas cooled reactor poses unique technical challenges. One of these challenges is the design and procurement of new components. Furthermore nuclear grade component design, qualification and procurement involve many entities that must be managed to achieve the desires outcome.

In nuclear industry as it is so in other industries it s traditional to codify material specification primarily through ASME before it is submitted to the nuclear regulator for acceptance of use. Metallic and ceramic materials that are expected to be used for the AREVA NGNP reactor vessel and internals are being considered for codification by ASME.

The proposed vessel made from Mod 9Cr-1Mo must be manufactured from a combination of welded plates and forgings. The heavy section forging have especially long lead procurement issues that include placement of order in 2008 to reserve a place in a long line of forging procurement.

The ceramic material including graphite must be characterized for nuclear application. This schedule needs for graphite component acquisition among the key milestones that have been identified on the schedule.

Major components with long lead procurement are: the reactor vessel and IHX vessels, compact IHX, tubular IHXs, helium circulators, and the secondary side isolation valves. Also certain parts of the PCS are considered long lead procurement activity.

17.3.3 Licensing and Permitting

The NGNP will be licensed with the conventional two step 10CFR50 two step licensing process:

1. Step 1 is a construction permit based on the review of a preliminary safety analysis report (PSAR);
2. Step 2 is an operating license based on the review of a final safety analysis report (FSAR).

The licensing schedule is based on the above approach, which is discussed in detail in Section 12. The licensing schedule has three critical milestones:

- Limited Work Authorization (LWA) – 3rd quarter 2012
- Construction Permit Approval (CP) – 3rd quarter 2013
- Operating License Approval (OL) – 3rd quarter 2018

LWA/CPA

Receipt of the LWA permits limited site work to begin but at risk. Receipt of the CP is necessary to begin large scale site construction activities. Receipt of the OL is necessary to commence initial plant operations. Any delay in achieving these milestones has a direct impact on the overall project schedule.

The LWA and CP milestones have essentially the same prerequisites that must be satisfied. Conceptual and preliminary plant design work leads to the submittal of the Preliminary Safety Analysis Report (PSAR) and the NRC Construction Permit Application. In parallel, license and permits from other jurisdictional entities must be applied for and/or obtained: EIS, State of Idaho and local permits, FERC, DOE/DOT. Interaction with the NRC and other regulators is required to ensure adequacy of submittals and timely reviews. The LWA is issued based upon the approval of pre-CP site activities, most of which are normally readily reversible with respect to site impact. The CP is issued upon receipt of NRC approval of the CP application, usually after intense interaction between staff and NGNP licensing (e.g., NRC requests for additional information (RAIs), responses to RAI, issue resolutions, etc.).

Of paramount importance to the project for the timely receipt of the LWA and CP milestones is the demonstration of the safety basis for the NGNP, albeit on a preliminary level at that time. In particular, and setting the project at risk, is the resolution of the containment issue. Will a pressure retaining containment structure, similar to that used in LWRs, be required for the NGNP? Or, will a confinement arrangement as proposed for the NGNP Preconceptual design be acceptable? Feeding into this debate is the reliability of the particle fuel for the HTR's safety case and the need for containment are intimately linked through fuel reliability. It is highly likely that CP issuance will be captive by this issue until it is resolve.

There are other issues that will be encountered no doubt along the way to CP approval. One noteworthy item is the development of regulations appropriate to HTR technology. Since 10 CFR 50 is LWR based, development of and agreement with regulations for HTRs will need to be sufficiently advanced to ensure NRC has a regulatory basis on which to formulate its approval.

OL

Receipt of the OL marks the beginning of NGNP operations. Most likely, the OL will be granted in stages (e.g., 2 % power, 10%, 50%, 100%) commensurate with the start up program and attendant plant performance.

The key prerequisites to receiving the OL are the submittal of the final safety analysis report, the operating license application, successful startup program and construction and pre-operational closeout. Similar to the case for LWA/CP approval, OL approval is contingent upon successful demonstration of the NGNP's safety case backed by actual fuel demonstration of fuel reliability. The demonstration of the safety case is also contingent on NRC acceptance/concurrence with analysis methods and results providing the underlying bases to the information contained in the FSAR.

The OL is issued upon receipt of NRC approval of the OL application, usually after intense interaction between staff and NGNP licensing (e.g., NRC requests for additional information (RAIs), responses to RAI, issue resolutions, etc.). Furthermore, the OL will only be issued upon successful clearance of construction issues (i.e., issues stemming from on-site inspections).

Intervention

The approach to licensing the NGNP by 10 CFR 50 does expose the project to more opportunity to delays arising from intervention during the legal processes. In the NRC licensing process, public hearings are required prior to the LWA, CP, and OL approvals. The permissible issues that could be raised are not as constrained as they are under the 10 CFR 52 process. Other jurisdictional entities also offer the opportunity for intervention as well (e.g., EIS, site use etc.).

Implementation of a successful licensing strategy is critical to achieving the milestones for LWA, CP and OL. While intervention is a possibility, the demonstration of the NGNP's safety case is critical to ensuring the licensing schedule can be maintained. Close and continued interaction with NRC and other regulators through the project will be also be a key factor to maintaining schedule.

17.4 Fuel Development and Qualification Schedule

The fuel-related procurement efforts in support of operation of the NGNP plant can be divided into two main areas, fuel qualification and first core fuel fabrication, each with its own series of sub-tasks. The natural progression of these tasks would be that each step would be completed before the next is initiated. This would allow ongoing evaluation and re-focusing of effort as necessary to assure a successful end product capable of meeting all NGNP goals. The current NGNP start date of 2018 does not allow this type of progression.

In the current fuel schedule, these tasks are essentially conducted in parallel, with limited staggering of start dates to allow at least some rudimentary feedback to occur. This introduces some risk into the process. This risk in limited through implementation of a fairly conservative fuel design produced on fabrication equipment based on an existing prototype-scale fabrication line that is currently in use supporting the US AGR fuel program. Lessons learned from past HTGR fuel performance, particularly the successful German fuel program, have been incorporated into the AREVA/BWXT process to further reduce this risk.

The fuel qualification program consists of a series of three fuel production, fuel irradiation, and fuel inspection sequences. The first of these, the Performance Fuel, is designed to verify that the fuel fabrication process, as implemented to produce the NGNP particle design, can reliably produce fuel with acceptable performance. Shortly after this fuel begins its irradiation cycle, the Qualification Fuel will begin production. The short duration of operation before production begins will allow the identification of unforeseen, catastrophic fabrication failures through in-core off-gas monitoring. This batch of fuel will be larger than the Production Fuel to allow a statistically based safety case to be established, and may require irradiation in multiple test reactors, adding to schedule complexity. The last batch of fuel to be fabricated will be the Verification Fuel. This fuel will be fabricated on the commercial fabrication line rather than the prototype line. Though these fabrication lines are expected to be effectively identical, this batch will be used to verify this assumption. Fabrication of this batch of fuel is expected to begin shortly after commissioning of the commercial line. At that time, initial test results from the performance fuel, including safety testing, should be available to provide reasonable assurance of good performance. In addition, in-core off-gas monitoring results for the initial operation of the Qualification Fuel will also be available.

As the fuel qualification process progresses, construction of the commercial scale fuel fabrication line will be completed. This line is to be designed based on the prototype fabrication line with minor upgrades to allow more efficient large-scale production. One of the key decisions to be made, which may impact fabrication schedule and economics, is the size of the coater. Use of a larger diameter coater would allow a more efficient process, but may extend the scope of the Verification Fuel test sequence. Fabrication of the first core fuel would commence upon receipt of NRC licensing of the commercial production line.

One strategy that has been considered to gain schedule flexibility is to fabricate some portion of the first core fuel on the prototype line. Under this plan, fabrication would begin shortly after production of the Qualification Fuel and would be expected to provide approximately ten to twenty percent of the first core requirement. This would also have the benefit of maintaining a trained fabrication and quality control staff once initial NGNP production has begun, eliminating the questions of loss of skilled personnel. This would increase overall risk somewhat in that late failure of the Qualification Fuel would render a significant quantity unusable.

17.5 Construction

The NGNP prototype facility construction phase schedule is also shown on Figure 17-1. Total duration of the construction phase is estimated to be 72 months. The construction phase begins with the receipt of LWA which starts the non-safety related construction activities such as site preparation and excavation of the deep silo for the nuclear island. Grounds preparations and deep excavation operation are expected to last eight to twelve months depending of the characteristics of the underground rock formation.

The nuclear island concrete work will begin in the deep silo following the issuance of the CP. The NGNP required construction techniques would be different from the traditional stick building technique. Modular construction is being extensively used at OL#3 an AREVA Generation III light water reactor construction site. This experience in modular construction techniques will be fully exploited to take full advantage of the unique opportunities that a below ground civil structure design represents.

AREVA anticipates that most NGNP components will be built off-site and shipped to the site in a modular prefabricated form for immediate installation. This includes prefabricated rebar panels and sections for the reactor building reinforcement, prefabricated large helium-nitrogen pipe sections and the reactor cavity cooling system panels.

Certain large components such as the reactor vessel may not be transportable to the Idaho site, therefore, onsite assembly of these components are anticipated. Here again AREVA extensive shop fabrication experience will be used to design an on-site fabrication techniques to conform to the Idaho site requirements.

17.6 Initial Startup Operations and Testing

The initial startup and testing is critical to the overall schedule performance of any nuclear plant. The NGNP prototype facility is no exception. As the prototype demonstration plant for the new generation of high temperature gas cooled reactors the NGNP initial startup operation and testing schedule is developed to achieve the following:

- Component testing and turn-over
- System functional testing and turn-over
- Initial approach to criticality
- Zero power operation
- Power ascension including grid connection
- Normal plant safety system tests (AOO tests)
- Special licensing performance tests (DBA tests)
- Commercial operability endurance tests
- Component dismantling and examination
- Fuel examination

The schedule provides four years for this phase of plant operations. The first two years (2017 and 2018) is dedicated to non-nuclear system testing and turn-over including the standard system turn-over from construction to operations. The second two years (2019 and 2020) includes initial plant criticality. During this phase all safety systems will be examined and tested and several special licensing related tests is planned. This phase of the plant operation includes component dismantling and inspection and fuel examination.

Power generation during this phase of operation is at best intermittent since the main objective is prototype performance testing and inspection.

17.7 Operations and Demonstration

The operation and demonstration phase of the NGNP prototype facility is scheduled to start following the successful completion of the initial tests described in the previous section. During this phase commercial operation of the electric plant and scaled demonstration of the hydrogen production facility will commence.

There are two main objectives of this phase. The first is to gain HTR operational experience as electricity is commercially generated sold to the national electrical. The second and equally important objective is to deliver process heat to a variety of high temperature process heat applications. The 60 MWth loop is ideal for testing of a variety of compact IHXs or high efficient power conversion technologies such as the supercritical CO_2 cycle.

The NGNP initial commercial plant license will be for 40 years; the first two years will be dedicated for initial operations, inspection and testing. The remaining thirty eight years includes commercial operations using an eighteen month refueling cycle.

17.8 Key Milestones

Table 17-1 lists the milestones identified as "key" associated with the AREVA NGNP prototype facility design, construction and operations schedule.

Table 17-1: NGNP Project Key Milestones

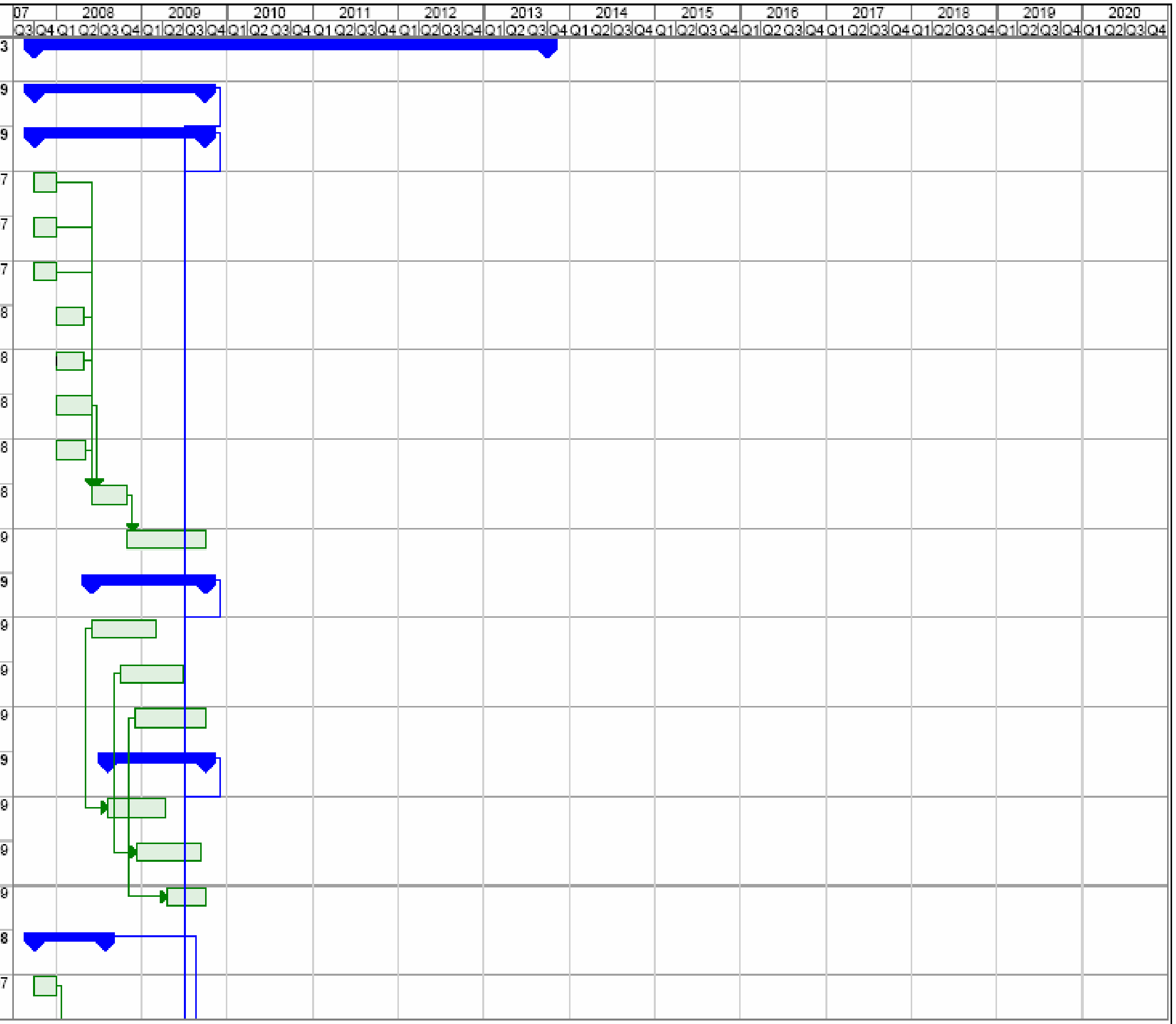
Milestone	Date
Design	
Start of Conceptual Design	Oct-2007
Selection of compact IHX Concept	Mar 2009
Completion of Final Design	Oct-2013
Licensing	
Completion of Environmental Report	Apr 2009
Limited Work Authorization	Oct 2012
Construction Permit	Oct 2013
Operating License	Oct 2018
Materials & Components	
Selection of IHX Material	May 2009
Selection of Vessel Material Grade	Jan 2009
Selection of Graphite Grade	July 2013
End of Forging Qualification Program	Jan 2015
Procurement	
Forging Purchase Order – place holder	May 2008
Reactor Vessel PO -Forgings	Jan 2009
Reactor Vessel PO - Plates	Jan 2012
IHX Purchase Order	Oct 2013
Graphite Internals PO	Oct 2013
Delivery of Vessel Forgings	Jan 2014
Delivery of Vessel Packages on-site	Dec 2015
Delivery of CCGT	Jan 2016

Milestone	Date
Delivery of Circulators	Oct 2015
Delivery of IHX Modules	Jan 2017
Delivery of First Core	Oct 2017
Fuel Development and Qualification	
Begin Performance Fuel Fabrication	Oct-2007
Complete Performance Fuel Inspection	Sept 2010
Begin Qualification Fuel Fabrication	Apr 2008
Complete Qualification Fuel Inspection	Oct 2011
Complete Pilot Line Upgrades	Oct-2009
Begin Commercial Line Construction	Dec-2008
Complete Commercial Line Construction	May-2012
Begin Verification Fuel Fabrication	Feb 2013
Complete Verification Fuel Inspection	Jan 2018
Begin First Core Fabrication	Oct-2014
Complete First Core Fabrication	Oct 2017

17.9 Critical Path Analysis

The critical path for the NGNP prototype facility to plant first criticality on October 1, 2018 is envisioned to be through a sequence of activities, beginning with placing an order (initial purchase order commitment place holder) for the reactor vessel forgings and the start of performance fuel manufacturing and qualification. The critical path continues during fuel acquisition activities as discussed in Chapter 15 and procurement and qualification of the reactor vessel materials. The critical path continues with manufacturing activities associated with qualification and delivery of the first core and the installation of the reactor vessel system, unit helium piping and mechanical/electrical system installation. Following major component installation activities the critical path begins with system tests, fuel loading, and startup testing.

ID	Task Name	Start	Finish
1	**Design**	**10/1/2007**	**9/30/2013**
2	**Conceptual**	**10/1/2007**	**9/30/2009**
3	**Plant/System Design (Main Systems)**	**10/1/2007**	**9/30/2009**
4	Nuclear Heat Source (NHS)	10/1/2007	12/31/2007
5	Primary Heat Transfer System (PHTS)	10/1/2007	12/31/2007
6	Shutdown Cooling System (SCS)	10/1/2007	12/31/2007
7	Power Conversion System (PCS)	1/1/2008	4/30/2008
8	Safety Systems (RCCS, Neutron Control System)	1/1/2008	5/1/2008
9	Control and Protection Systems	1/1/2008	5/30/2008
10	Key Structures (Reactor Building, Turbine Building	1/1/2008	5/2/2008
11	Plant Layout	6/2/2008	10/28/2008
12	Issue SDDs (Main Systems)	10/29/2008	9/30/2009
13	**Analytical Codes & Methods Development**	**6/2/2008**	**9/30/2009**
14	Steady State Codes	6/2/2008	3/3/2009
15	Transient Analysis Code	10/1/2008	6/30/2009
16	PRA	12/1/2008	9/30/2009
17	**System Analysis (Main Systems)**	**8/11/2008**	**9/30/2009**
18	Steady State Analysis	8/11/2008	4/10/2009
19	Transient Analysis	12/10/2008	9/9/2009
20	Probabilistic Risk Assessment	4/20/2009	9/30/2009
21	**Codes and Standards Activities**	**10/1/2007**	**7/30/2008**
22	Evaluate and Select Design Codes	10/1/2007	12/31/2007

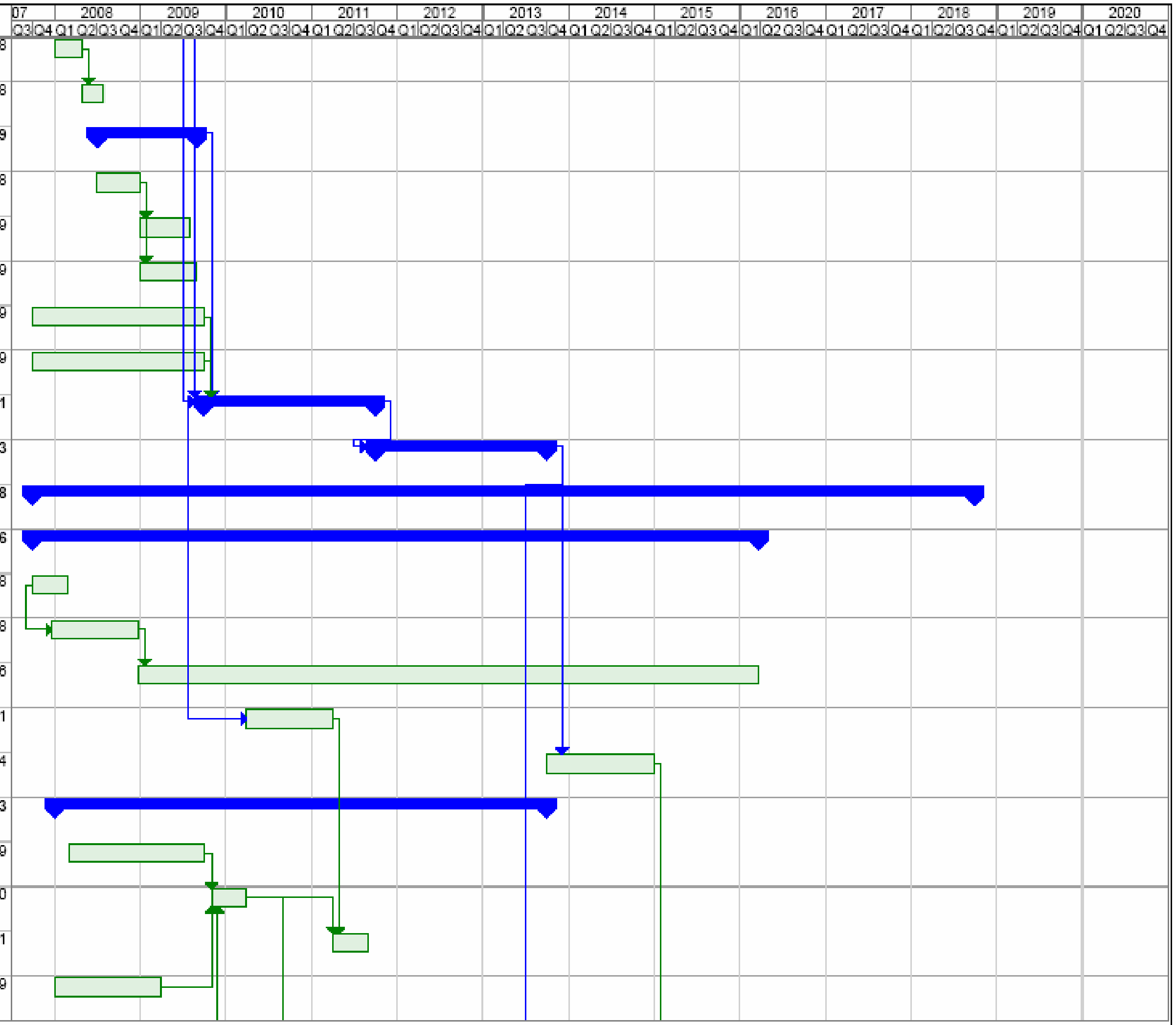


Project: NGNP New Schedule-Rev1 2I
Date: 6/4/2007

Task | Critical Task | Progress | Milestone | Summary | Rolled Up Task | Rolled Up Critical Task | Rolled Up Milestone | Rolled Up Progress | Split | External Tasks | Project Summary | Group By Summary | Deadline

Page 1

ID	Task Name	Start	Finish
23	Identify Needed Code Cases	1/1/2008	4/29/2008
24	Apply for Needed Codes Cases	4/30/2008	7/30/2008
25	**Develop Plant/System Requirements**	**7/1/2008**	**8/31/2009**
26	Main Plant Requirements	7/1/2008	12/30/2008
27	Initial Draft of System Requirements (key systems	12/31/2008	7/30/2009
28	System Interfaces Identification	12/31/2008	8/31/2009
29	**Perform Key Trade Studies**	10/1/2007	9/30/2009
30	**Maintain Reference Design Baseline**	10/1/2007	9/30/2009
31	**Preliminary**	**10/1/2009**	**9/30/2011**
57	**Final Design**	**10/3/2011**	**9/30/2013**
73	**Licensing**	**10/1/2007**	**10/1/2018**
74	**Safety Analysis Reports**	**10/1/2007**	**3/23/2016**
75	Review DOE/NRC Licensing Strategy	10/1/2007	2/29/2008
76	Prepare NGNP INL Site Licensing Stategy	12/24/2007	12/24/2008
77	NRC Pre-application Discussions	12/25/2008	3/23/2016
78	Prepare Preliminary SAR (PSAR)	4/1/2010	3/31/2011
79	Prepare Final SAR Preparation (FSAR)	10/1/2013	12/31/2014
80	**Construction Permit and LWA**	**1/1/2008**	**10/1/2013**
81	Prepare ESP Application	3/3/2008	10/1/2009
82	Prepare LWA application	11/3/2009	3/31/2010
83	Prepare CP Application	4/1/2011	8/31/2011
84	Prepare Environmental Report	1/1/2008	4/2/2009

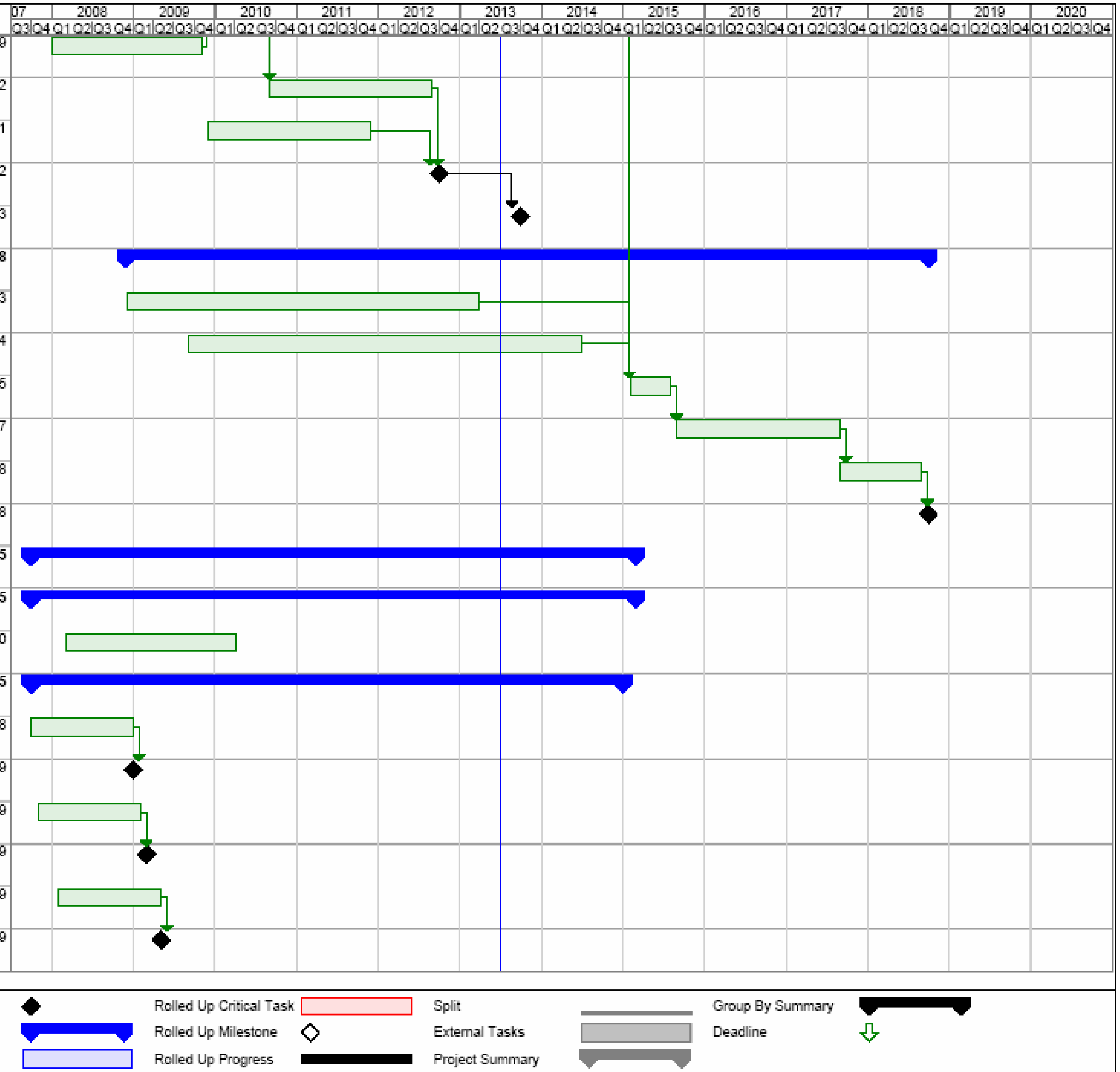


Project: NGNP New Schedule-Rev1 2I
Date: 6/4/2007

Task	Milestone	Rolled Up Critical Task	Split	Group By Summary
Critical Task	Summary	Rolled Up Milestone	External Tasks	Deadline
Progress	Rolled Up Task	Rolled Up Progress	Project Summary	

Page 2

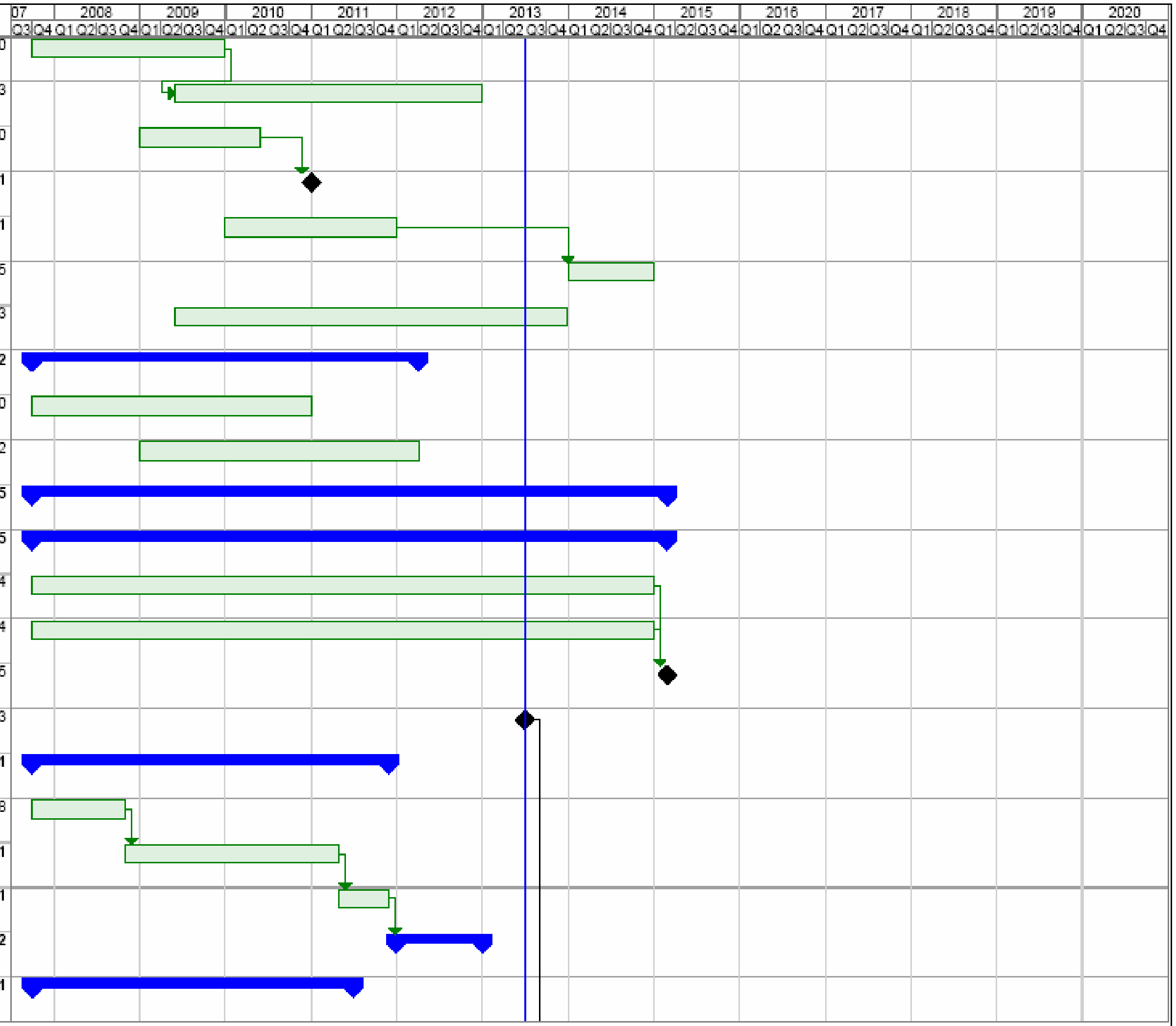
ID	Task Name	Start	Finish
85	State of Idaho/EPA Environmental Permit Applications	1/1/2008	11/2/2009
86	NRC Review LWA and CP	9/1/2010	8/30/2012
87	State and EPA Reviews	12/1/2009	12/1/2011
88	NRC Issue LWA	10/1/2012	10/1/2012
89	NRC issue CP	10/1/2013	10/1/2013
90	**Operating License**	**12/1/2008**	**10/1/2018**
91	Prepare Topical Reports	12/1/2008	3/29/2013
92	NRC Review of Topical Reports	9/1/2009	7/3/2014
93	Prepare OP Application	2/2/2015	7/30/2015
94	NRC Review	9/1/2015	8/31/2017
95	Public Hearing	9/1/2017	8/30/2018
96	NRC Issue OL	10/1/2018	10/1/2018
97	**Research and Development**	**10/1/2007**	**3/1/2015**
98	**Material Development & Qualification**	**10/1/2007**	**3/1/2015**
99	Tribology	3/3/2008	4/2/2010
100	**Metallic Materials**	**10/1/2007**	**1/1/2015**
101	Vessel Materials (High temperature steel)	10/1/2007	12/31/2008
102	Confirmation of selection of vessel material	1/1/2009	1/1/2009
103	Reactor Internals	11/1/2007	2/2/2009
104	Selection of metallic materials	3/2/2009	3/2/2009
105	IHX	2/1/2008	5/5/2009
106	Selection of IHX materials	5/6/2009	5/6/2009



Project: NGNP New Schedule-Rev1 2l
Date: 6/4/2007

Page 3

ID	Task Name	Start	Finish
107	ASME update (IHX and Vessel materials)	10/1/2007	1/1/2010
108	NRC review and approval	6/1/2009	1/1/2013
109	Welding Qualification	1/1/2009	6/1/2010
110	End of welding process qualification	1/3/2011	1/3/2011
111	Qualification of vessel material (including irradiatio	1/1/2010	12/30/2011
112	Final qualification on forgings	1/1/2014	1/1/2015
113	IHX qualification	6/1/2009	12/30/2013
114	**Ceramics**	**10/1/2007**	**4/3/2012**
115	Reactor Internals	10/1/2007	12/31/2010
116	Control rod sheaths (Carbon-Carbon composites)	1/1/2009	4/3/2012
117	**Graphite Materials**	**10/1/2007**	**3/1/2015**
118	**Graphite Qualification**	**10/1/2007**	**3/1/2015**
119	Material Testing	10/1/2007	12/31/2014
120	Irradiation	10/1/2007	12/31/2014
121	End of Qualification program	3/1/2015	3/1/2015
122	Selection of graphite grades	7/1/2013	7/1/2013
123	**Large He Test Loop**	**10/1/2007**	**11/28/2011**
124	Design 10 MW He Test Loop	10/1/2007	10/31/2008
125	Construct Test Loop	11/3/2008	4/29/2011
126	Commissioning	5/2/2011	11/28/2011
127	**Component Testing and Qualification**	**1/1/2012**	**12/31/2012**
141	**Power Conversion System R&D**	**10/1/2007**	**6/30/2011**

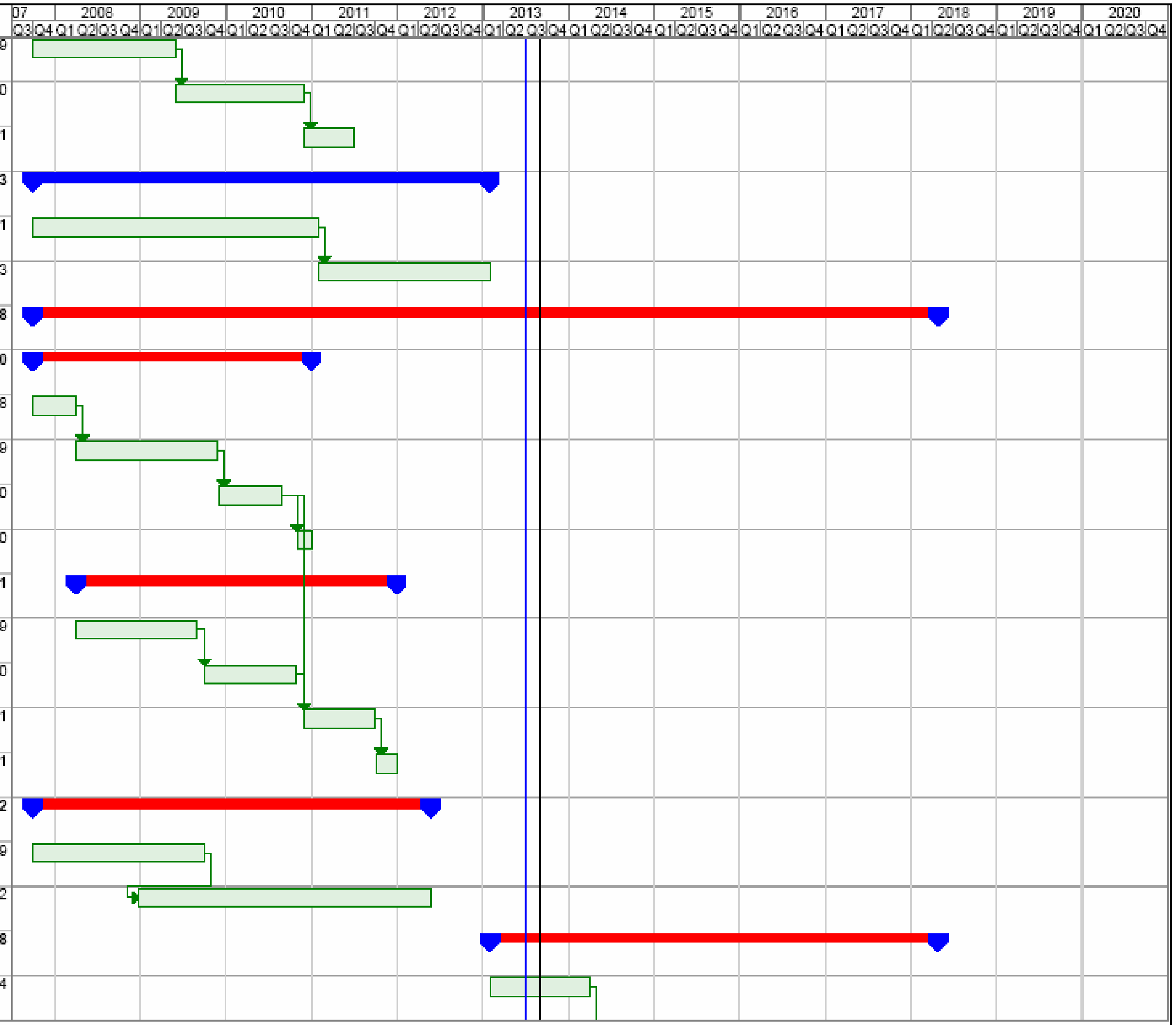


Project: NGNP New Schedule-Rev1 2l
Date: 6/4/2007

Task		Milestone		Rolled Up Critical Task		Split		Group By Summary	
Critical Task		Summary		Rolled Up Milestone		External Tasks		Deadline	
Progress		Rolled Up Task		Rolled Up Progress		Project Summary			

Page 4

ID	Task Name	Start	Finish
142	Nitriding Test for Candidate materials	10/1/2007	5/29/2009
143	Test of Nitriding Protection Methods	6/1/2009	11/30/2010
144	Design Decisions	12/1/2010	6/30/2011
145	**Computer Codes & Methods Development & Qualificati**	**10/1/2007**	**1/30/2013**
146	Code Development	10/1/2007	1/31/2011
147	Computer Code Qualification	2/1/2011	1/30/2013
148	**Fuel**	**10/1/2007**	**4/27/2018**
149	**Performance Fuel**	**10/1/2007**	**12/31/2010**
150	Fabricate Fuel Compacts	10/1/2007	3/31/2008
151	Irradiation Test (1)	4/1/2008	11/27/2009
152	PIE	11/30/2009	8/27/2010
153	Safety Basis Input	11/1/2010	12/31/2010
154	**Qualification Fuel**	**4/1/2008**	**12/30/2011**
155	Fabricate Fuel Compacts	4/1/2008	8/31/2009
156	Irradiation Test (2)	10/1/2009	10/29/2010
157	PIE	12/1/2010	9/28/2011
158	Safety Basis Input	10/3/2011	12/30/2011
159	**Fabrication Facility**	**10/1/2007**	**5/25/2012**
160	Pilot Upgrades	10/1/2007	10/1/2009
161	Fabrication Facility	12/26/2008	5/25/2012
162	**Verification Fuel**	**2/1/2013**	**4/27/2018**
163	Fabricate Fuel Compacts	2/1/2013	3/31/2014

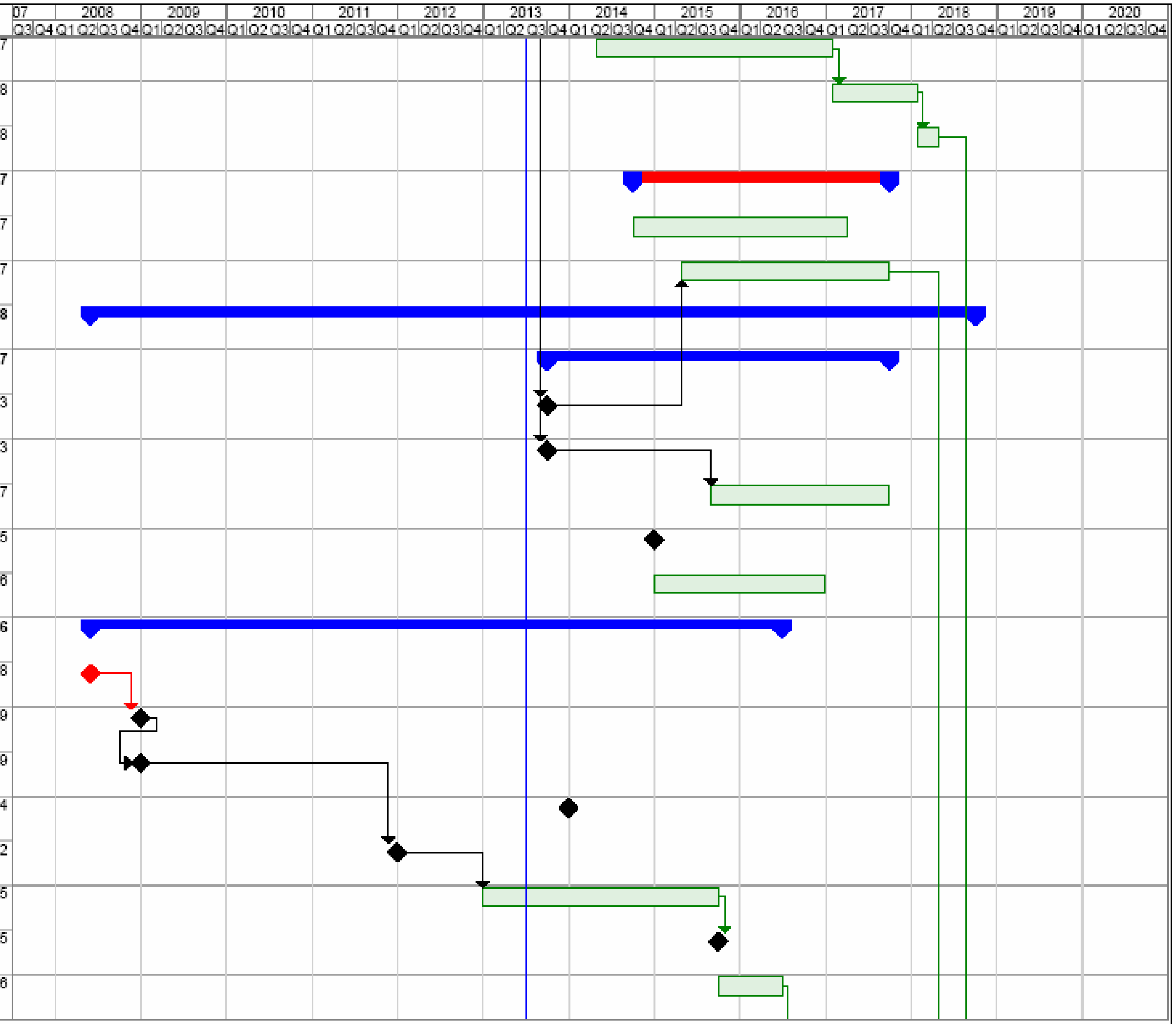


Project: NGNP New Schedule-Rev1 2l
Date: 6/4/2007

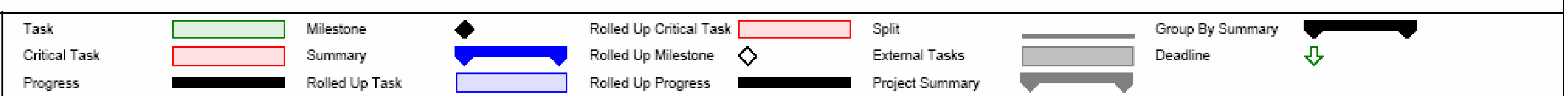
Task | Critical Task | Progress | Milestone | Summary | Rolled Up Task | Rolled Up Critical Task | Rolled Up Milestone | Rolled Up Progress | Split | External Tasks | Project Summary | Group By Summary | Deadline

Page 5

ID	Task Name	Start	Finish
164	Irradiation Test (1)	5/1/2014	1/30/2017
165	PIE	1/31/2017	1/26/2018
166	Safety Basis Input	1/29/2018	4/27/2018
167	**First Core Production**	**10/2/2014**	**10/1/2017**
168	Fabricate Fuel Kernels and Compacts	10/2/2014	3/31/2017
169	Fuel Graphite Blocks	4/27/2015	10/1/2017
170	**Procurement**	**5/30/2008**	**10/1/2018**
171	**Reactor Core and Internals**	**10/1/2013**	**9/29/2017**
172	Procure Graphite Blocks	10/1/2013	10/1/2013
173	Graphite internals purchase order	10/1/2013	10/1/2013
174	Graphite internals fabrication and installation	9/1/2015	9/29/2017
175	Metallic internals purchase order	1/1/2015	1/1/2015
176	Metallic internals fabrication and installation	1/1/2015	12/30/2016
177	**Vessel System**	**5/30/2008**	**6/30/2016**
178	Vessel Forgings	5/30/2008	5/30/2008
179	Confirmation of selection of vessel material	1/1/2009	1/1/2009
180	Purchase order of the forgings	1/1/2009	1/1/2009
181	Delivery of forgings	1/1/2014	1/1/2014
182	Purchase order of plates (Vessel system)	1/1/2012	1/1/2012
183	Vessels fabrication	1/1/2013	10/1/2015
184	Delivery of vessel packages on site	10/1/2015	10/1/2015
185	Fabrication on site and controls	10/2/2015	6/30/2016

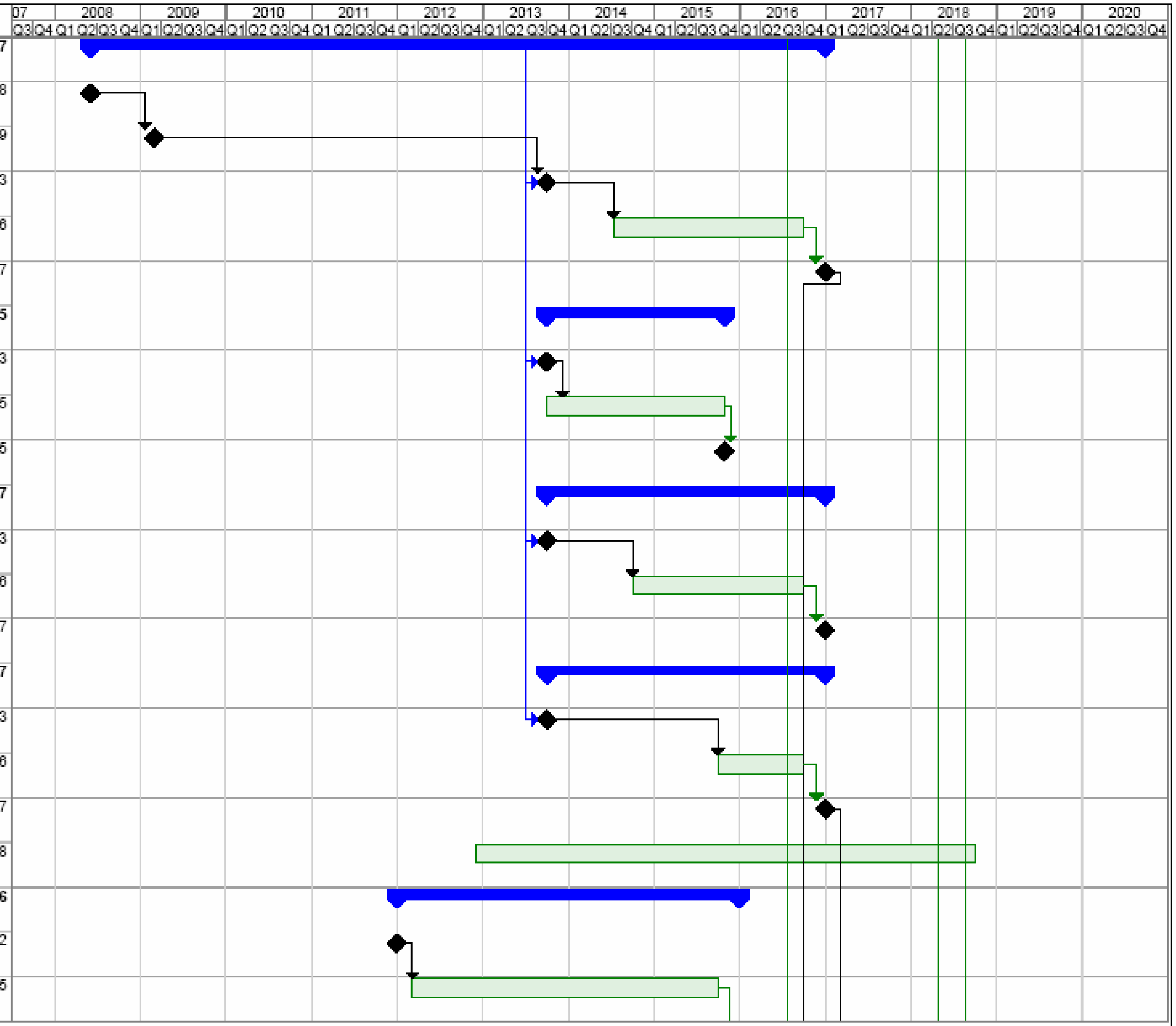


Project: NGNP New Schedule-Rev1 2I
Date: 6/4/2007



Page 6

ID	Task Name	Start	Finish
186	**IHXs**	**6/2/2008**	**1/1/2017**
187	Selection of IHX materials	6/2/2008	6/2/2008
188	Selection of compact IHX concept	3/2/2009	3/2/2009
189	IHXs purchase order	10/1/2013	10/1/2013
190	IHXs fabrication	7/14/2014	9/30/2016
191	Delivery of IHX for Installation	1/1/2017	1/1/2017
192	**Circulators**	**9/30/2013**	**10/28/2015**
193	Circulator Purchase order	9/30/2013	9/30/2013
194	Circulator fabrication	10/1/2013	10/28/2015
195	Delivery of Circulators for Installation	10/28/2015	10/28/2015
196	**Hot Gas Duct**	**9/30/2013**	**1/2/2017**
197	Hot gas duct Purchase Order	9/30/2013	9/30/2013
198	Hot gas duct fabrication	10/2/2014	9/30/2016
199	Hot gas duct on site	1/2/2017	1/2/2017
200	**Isolation Valves**	**9/30/2013**	**1/2/2017**
201	Isolation Valves Purchase Order	9/30/2013	9/30/2013
202	Isolation valve fabrication	10/1/2015	9/30/2016
203	Isolation valve on site	1/2/2017	1/2/2017
204	**General Procurement**	12/4/2012	10/1/2018
205	**Power Conversion System Order**	**1/1/2012**	**1/1/2016**
206	PCS Purchase Order	1/1/2012	1/1/2012
207	Design and Fabrication	3/1/2012	9/30/2015

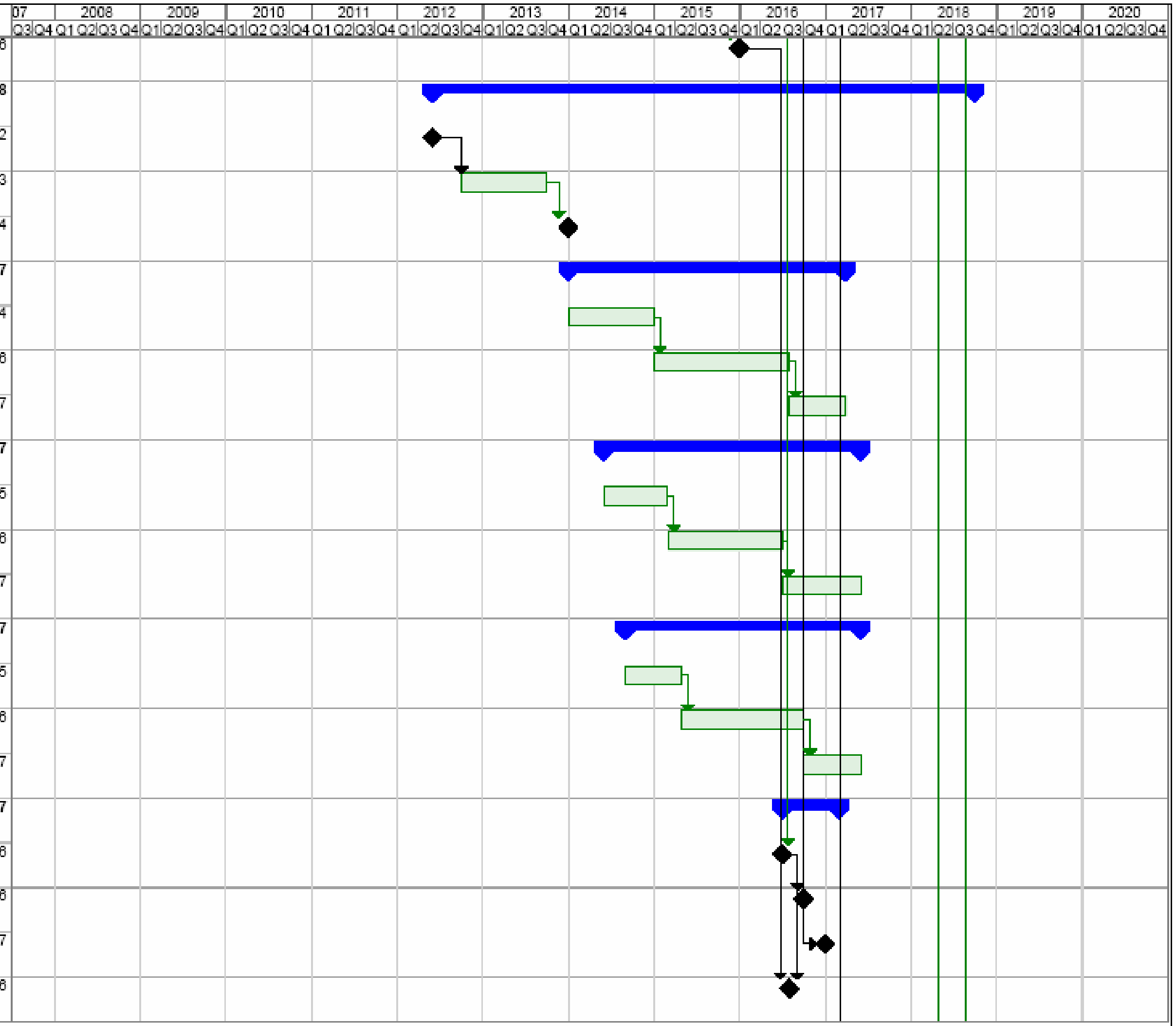


Project: NGNP New Schedule-Rev1 2I
Date: 6/4/2007

Task | Critical Task | Progress | Milestone | Summary | Rolled Up Task | Rolled Up Critical Task | Rolled Up Milestone | Rolled Up Progress | Split | External Tasks | Project Summary | Group By Summary | Deadline

Page 7

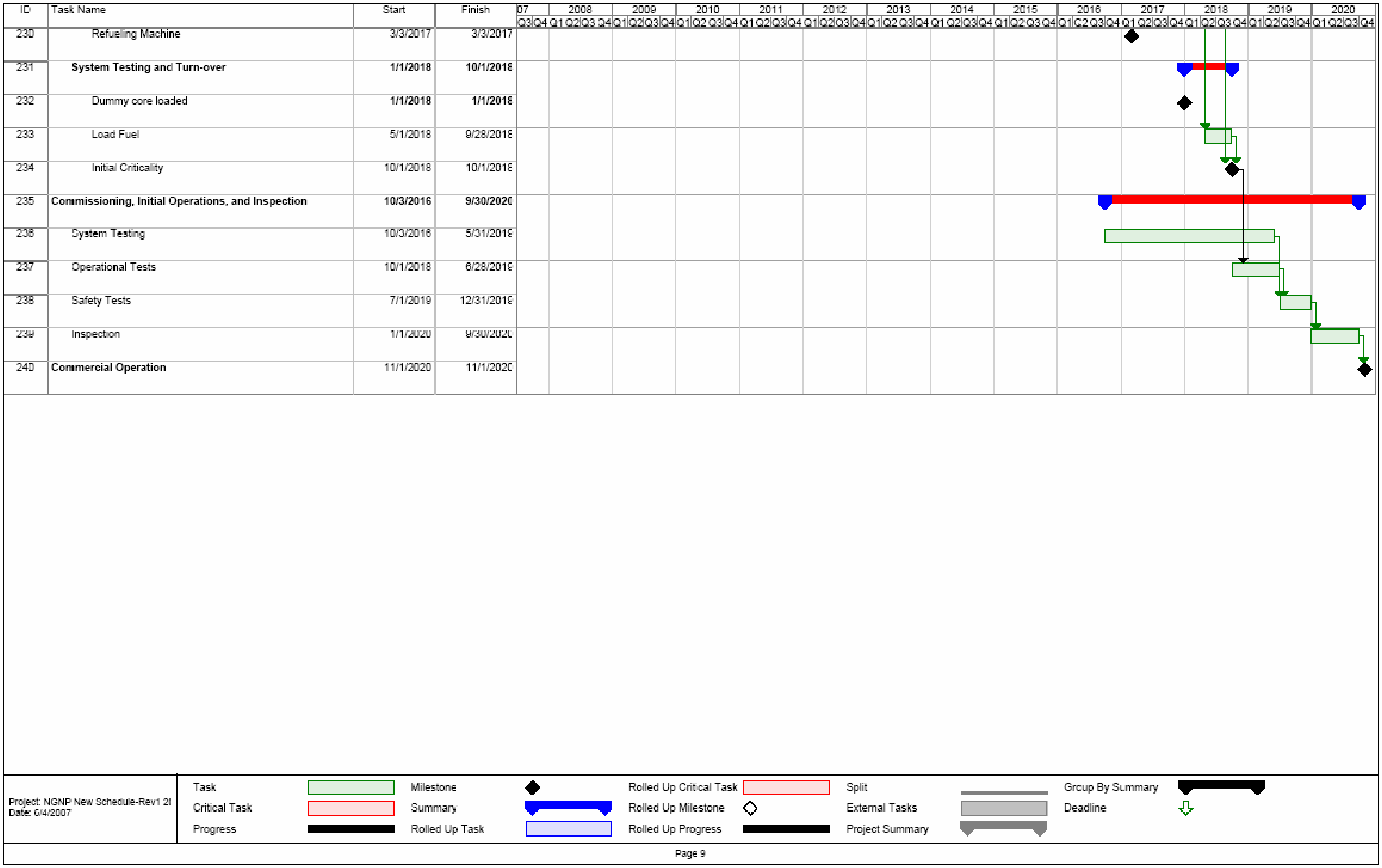
ID	Task Name	Start	Finish
208	PCS Component Delivery	1/1/2016	1/1/2016
209	**Construction**	**6/1/2012**	**10/1/2018**
210	Mobilization	6/1/2012	6/1/2012
211	Excavation of the Reactor Silo	10/1/2012	10/1/2013
212	First Pour of Concrete	1/1/2014	1/1/2014
213	**Nuclear Island**	**1/1/2014**	**3/30/2017**
214	Civil Structural Work	1/1/2014	12/31/2014
215	Mechanical Equipment & Piping	1/1/2015	7/29/2016
216	Electrical and I&C	8/1/2016	3/30/2017
217	**PCS Installation**	**6/2/2014**	**5/31/2017**
218	Civil Structural Work	6/2/2014	2/27/2015
219	Mechanical Equipment & Piping	3/2/2015	6/30/2016
220	Electrical and I&C	7/1/2016	5/31/2017
221	**Balance of Plant**	**9/1/2014**	**6/1/2017**
222	Civil Structural Work	9/1/2014	4/30/2015
223	Mechanical Equipment & Piping	5/1/2015	9/30/2016
224	Electrical and I&C	10/3/2016	6/1/2017
225	**Major Equiment Installation**	**6/30/2016**	**3/3/2017**
226	Reactor Vessel	6/30/2016	6/30/2016
227	Vessel Internals	10/1/2016	10/1/2016
228	IHXs (Vessels and Modules)	1/1/2017	1/1/2017
229	PCS	8/1/2016	8/1/2016



Project: NGNP New Schedule-Rev1 2I
Date: 6/4/2007

Task | Critical Task | Progress | Milestone | Summary | Rolled Up Task | Rolled Up Critical Task | Rolled Up Milestone | Rolled Up Progress | Split | External Tasks | Project Summary | Group By Summary | Deadline

Page 8



ID	Task Name	Start	Finish
230	Refueling Machine	3/3/2017	3/3/2017
231	**System Testing and Turn-over**	**1/1/2018**	**10/1/2018**
232	Dummy core loaded	**1/1/2018**	**1/1/2018**
233	Load Fuel	5/1/2018	9/28/2018
234	Initial Criticality	10/1/2018	10/1/2018
235	**Commissioning, Initial Operations, and Inspection**	**10/3/2016**	**9/30/2020**
236	System Testing	10/3/2016	5/31/2019
237	Operational Tests	10/1/2018	6/28/2019
238	Safety Tests	7/1/2019	12/31/2019
239	Inspection	1/1/2020	9/30/2020
240	**Commercial Operation**	11/1/2020	11/1/2020

Figure 17-1: NGNP Project Schedule

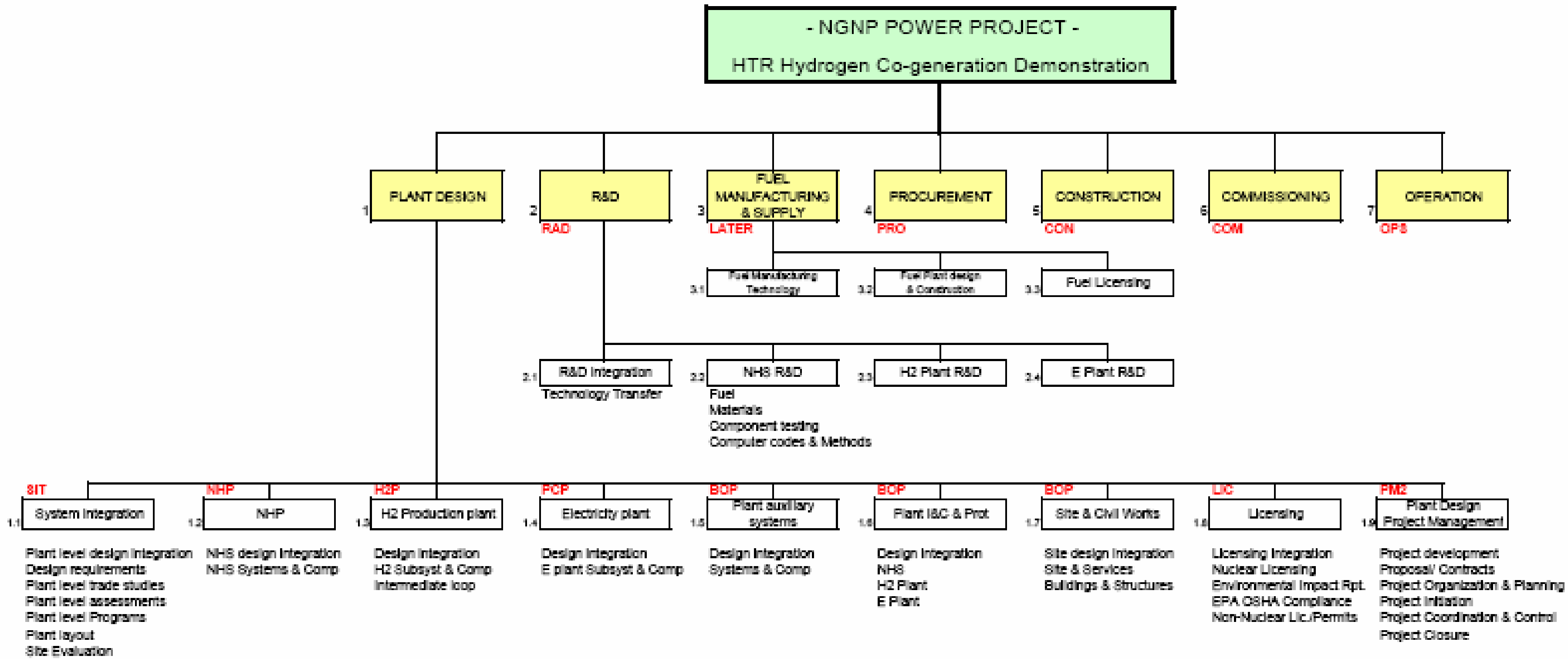


Figure 17-2: NGNP Project Organization

18.0 PROJECT RISK MANAGEMENT

Uncertainty in a project is called risk. It is formally defined as an uncertain event or condition that if it occurs, has a positive or negative effect on a project objective. Risk is an inherent part of all activities, whether the activity is simple and small, or large and complex. Risk Management is a continuous process that identifies, analyzes, mitigates, reports, and tracks risk. The Risk Management process will span the entire NGNP project from its initiation to its successful completion, and includes both internal (within project control) and external (outside project control) risks.

18.1 Risk Management Approach

Risk Management is a formal and disciplined approach focused on identifying, analyzing, and responding to risk. It includes maximizing the results of positive events and minimizing the consequences of adverse events. Its purpose is to enhance the probability of project success by increasing the likelihood of improved project performance, and decreasing the likelihood of unanticipated cost overruns, schedule delays, and compromises in quality and safety.

The Risk Management approach for the NGNP project consists of six major processes, which are briefly described below:

1. Planning,
2. Identification,
3. Assessment,
4. Handling and Response,
5. Impact Determination, and
6. Tracking, Reporting, and Closeout.

Risk Management is essentially an ongoing and iterative process, which applies the best efforts of a knowledgeable project staff to a suite of focused and prioritized issues. The scope of Risk Management applies to the design, construction, modification, and operation of the NGNP facilities.

18.1.1 Planning

Risk Planning for the project is the establishment of the overall Risk Management program and a Risk Management Working Group, assignment of responsibilities, and the documentation of the Risk Management program in a Risk Management Plan (RMP). A detailed RMP will be developed in a subsequent project phase. As the project matures, the RMP will change and additional implementing procedures may be necessary.

18.1.2 Identification

Risk identification is an organized approach for determining and selecting which events are likely to affect the project and documenting the characteristics of the events that may occur with a basis as to why identified events are considered risks.

Two tools were used for this process; the Project Risk Category Screening Checklist, and the Risk Assessment Form (RAF). The Project Risk Category Screening Checklist has been developed as an initial screening tool to assist the project personnel to identify risks to the project. The Checklist attempts to capture broad project categories where risk is likely to occur, such as technical, security, and programmatic areas. A Risk Management Working Group evaluated each item to determine whether anything in the project presents a risk.

Each risk applicable to the project as determined from the Screening Checklist was then documented on a Risk Assessment Form. The form ties the risk to a WBS element and is a formalized way to document each risk for the project. The initial Risk identification for the project relied upon the skill, experience, and insight of project/functional personnel utilizing the forms discussed above.

18.1.3 Assessment

Risk assessment involves determining the probability of the occurrence of a risk, assessing the consequences of this risk, and combining the two to identify a probability/consequence (P/C) correlation score. This P/C score represents a judgment as to the relative risk level to the project as a whole. Based on the risk level, handling strategies are identified to appropriately respond to the risk.

Table 18-1 and Table 18-2 show the criteria for defining probability and consequence of occurrences on the project. Each risk event was analyzed as an independent event (i.e., that the probability of the event occurring is independent of any other event occurring). Although interdependencies are expected among events, these interdependencies will be addressed later in the process. Overall, assessments of probability and consequence are qualitative in nature, based on the RAF initiator's experience and judgment.

Table 18-1: Probability of Occurrence

Qualitative	Quantitative	Criteria	Score
Very Unlikely	< 0.2	Will not likely occur anytime in the life cycle of the project or its facilities. The probability of occurrence is remote and less than 20%.	1
Unlikely	≥ 0.2 but < 0.4	Will not likely occur in the life cycle of the project or its facilities. The probability of occurrence is greater than or equal to 20% but less than 40%.	2
Likely	≥ 0.4 but < 0.6	Will likely occur sometime during the life cycle of the project or its facilities. The probability of occurrence is greater than or equal to 40% but less than 60%.	3
Very Likely	≥ 0.6 but < 0.8	Will likely occur sometime during the life cycle of the project or its facilities. The probability of occurrence is greater than or equal to 60% but less than 80%.	4

Qualitative	Quantitative	Criteria	Score
Most Likely	≥ 0.8 but < 1.0	Will occur sometime during the life cycle of the project or its facilities. The probability of occurrence is imminent and greater than or equal to 80% but less than 100%.	5

Table 18-2: Consequence of Occurrence

Qualitative	Quantitative	Criteria	Score
Negligible	< 0.1	Minimal or no consequences in project performance. No threat to project mission, environment, or personnel. Cost estimates not exceeded. Negligible impact on project mission.	1
Marginal	≥ 0.1 but < 0.3	Small reduction in project performance. Moderate threat to project mission, environment, or personnel. Cost estimates marginally exceed budget proposals. Potential adjustment to schedule/ milestones required that may affect the project mission.	2
Significant	≥ 0.3 but < 0.7	Significant degradation in project performance. Significant threat to project mission, environment, or personnel. Cost estimates significantly exceed budget. Significant adjustment to schedule with resulting milestone changes that may affect the project mission.	3
Critical	≥ 0.7 but < 0.9	Project objectives cannot be achieved. Serious threat to project mission, environment, or personnel. Cost estimates seriously exceed budget. Excessive schedule changes unacceptably affecting the project mission.	4
Crisis	≥ 0.9 but < 1.0	Project objectives cannot be achieved. Catastrophic threat to project mission,	

Qualitative	Quantitative	Criteria	Score
		environment, or personnel. Cost estimates unacceptably exceed budget. Project mission cannot be completed; failure imminent.	5

The Risk probability of occurrence and consequence of occurrence were then considered together to determine the P/C score. This score, obtained from Table 18-3, determines the appropriate risk handling strategy described in the next section. P/C score is color-coded on the Risk Assessment Form to denote relative severity (i.e., red and yellow scores denote a greater project risk that magenta and blue scores).

Table 18-3: Probability/Consequence Correlation Score

	Risk Consequence				
Risk Probability	1	2	3	4	5
5	2	4	6	9	12
4	2	3	5	8	11
3	1	2	4	7	10
2	1	2	3	5	8
1	1	1	2	3	5

18.1.4 Handling and Response

Risk handling is the identification of the course of action or inaction selected for the purpose of effectively managing or mitigating a given risk. Handling strategies are selected that identify the optimum set of steps to balance risk with other factors, such as cost and timeliness.

Once the P/C score is assessed, the response of the project will be directed by the corresponding P/C Action as shown in Table 18-4. These response actions are also color-coded to match the Risk Assessment Forms and are further defined as follows.

- **Acceptance**. This applies to risks with a P/C score of 1 (green). Accepting the risk is essentially a “no action” strategy. Selection of this strategy is based upon the decision that it is more cost effective to continue the project as planned with no resources specifically dedicated to addressing the risk. The cost and duration of implementation is zero, which is documented on the Risk Assessment Form.

  Of the 39 initial risks identified, one had a P/C score of 1. This item addresses the fuel server concept of the fuel handling system.

- **Monitoring**. While risks with P/C scores of 2 or 3 (blue) are still considered “low” risks, they will be monitored by management.

Of the 39 initial risks identified, 13 fall into the monitoring category. These include some design, both operations, and some of the licensing risks. A contingency plan will be developed to deal with these risks should they accelerate or materialize.

- **Mitigate.** This applies to risks with P/C scores of 4 or 5 (magenta). This strategy developed identifies specific steps or planned actions that will mitigate the consequence of a risk should it occur.

  Nine risks have been identified that require a planned action. These include several design risks, one construction risk, one procurement risk, and several licensing risks.

- **Reduce**. This strategy applies to risks with P/C scores of 6, 7, or 8 (yellow). Here, specific steps or actions are taken to reduce the probability of the occurrence of the risk. The expected outcome of a risk event can be reduced by lessening the probability of occurrence, or by reducing the risk outcome by adding specific mitigation actions and any corresponding cost implementation and schedule to the project scope.

  Of the 39 risks, 4 require a reduction strategy. These include two design risks, one procurement risk, and one licensing risk.

  In using the Mitigate or Reduce strategies, the risk remains, but at a reduced level. This reduced level is called the residual risk. By applying risk reduction/mitigation strategies, the goal is to reduce the P/C score from a higher level to a lower level in the chart (move from red to yellow to magenta, etc.). In fact, all risks with scores between 4 and 8 (mitigate/reduce) were successfully reduced to scores below 4 (monitor) except for two items.

- **Avoidance.** This strategy applies to risks with a P/C score of 9 or greater (red). It focuses on completely eliminating the specific threat or risk-driving event usually by eliminating the potential that the risk event can occur. This can be accomplished through total structure, system, or component redesign, or by selecting an alternate design approach that does not include the particular risk. Of course, the project will not be able to eliminate all risks, but specific risk events can often be eliminated with this strategy.

  Two design risks fall into this category; one addressing with fuel development funding, and the other addressing fuel testing resources. These risks cannot be avoided on the NGNP project, so aggressive mitigation strategies are needed, along with a cost and duration for implementing the strategies. The strategies identified for these risk items have reduced the risks to a score of 8 and 2, respectively. The risk having a score of 8 remains high because it is an external risk which is difficult to mitigate.

Table 18-4: Risk Handling Action

P/C Score	Action
≥ 9	Avoid the Risk through alternative approach.
6, 7, or 8	Identify measures to minimize the probability of the Risk occurring (and actively manage those measures) and steps to mitigate the consequences should the Risk occur.
4 or 5	Identify steps to mitigate the consequences should the Risk occur.
2 or 3	Monitor the Risk.
1	Accept the Risk.

18.1.5 Impact Determination

Risk impact determination is the process of evaluating and quantifying the effect of risks on the project. Risk impacts a project in two different ways:

1. Handling strategy implementation, which must be reflected in the project baseline,
2. Residual risk, which must be reflected in project contingency.

The ultimate impact of Risk Management is to increase the probability of project/activity success by focusing attention on problem areas early and reducing the amount of cost and schedule impacts in the future. The impacts of these risks on cost and schedule must be addressed in the project estimates.

18.1.6 Tracking, Reporting, and Closeout

Risk reporting and tracking is the active monitoring of action items developed from risk handling strategies from commencement to successful completion. Good risk monitoring and control processes provide valuable information that assist with making timely decisions in advance of risk materialization. As the project progresses, risk actions will be actively monitored on a continuous basis. The intent of the tracking and reporting process is to:

- Determine if risk mitigation strategies have been implemented as planned,
- Assess if risk response actions are effective,
- Reiterate risk evaluations to identified risks on periodic basis for effecting changes to risk identification and quantification over time due to maturing data and events, and
- Provide performance feedback tools for senior management.

The Risk Assessment Forms will serve as the primary input sources for risk reporting and tracking through the project life cycle.

Risk closeout is the formal documented disposition of an identified project risk event. Risk closure is initiated when the risk event no longer poses a threat to achieving project objectives within defined cost, schedule, technical and programmatic constraints. Risk Closeout requires:

- A residual risk assessment to ascertain the risk event has been reduced to a manageable level (e.g., P/C score ≤ 3).
- A comprehensive review by the Risk Management Working Group to ensure the mitigated risk event has not adversely affected other important aspects of the project.
- The identification and timely completion of pending closeout actions prior to executing final disposition.

18.2 Risk Management Results

18.2.1 Compilation of Identified Risks

The results of the risk identification step produced clear statements of risk with corresponding bases in five main categories:

- Design (17 risks identified),
- Construction (1 risk identified),
- Operations (2 risks identified),

- Procurement (2 risks identified), and
- Licensing (8 risks identified).

These risks are compiled in Table 18-5 along with mitigation strategies or specific tasks, as necessary, to lower probabilities of occurrence and/or reduce consequences. Top or key risks within these categories are discussed in Section 18.2.2.

18.2.2 Key Risks

In applying the Risk Management approach described above, five areas emerge as presenting the greatest risk to the NGNP project. These “key” risk areas are discussed below. These areas have the highest unmitigated probability/consequence (P/C) scores, and as such, need to be targeted for immediate action and monitoring by management.

- Fuel Development and Performance

  With respect to fuel, risks items include insufficient funding to allow a full fuel qualification program (D-005), unavailability of required test reactors (D-006), overall fuel performance during irradiation and safety testing (D-007), and fuel coating challenges (D-009). In the case of fuel performance, the probability of the risk can be reduced, but the potential consequence cannot be significantly minimized. Collective mitigation strategies include:

  - Identify fuel irradiation and inspection needs immediately and reserve required resources.
  - Fuels team is set up to consider a wide range of fuel variables that should result in an acceptable qualification effort.
  - Develop fuel fabrication process based on the use of multiple, proven coaters.

- Nitriding of Materials

  Nitriding of materials (Risk D-001) is a risk because of the high nitrogen weight-percent of in the secondary gas. This could primarily affect the PCS components. Steps that will be taken to mitigate this risk include the following:

  - A verification test will be performed to determine severity of the nitriding effect.
  - Based on results of the test, nitriding protection methods or use of alternate gas may be employed.

- Heavy Component Procurement and Fabrication

  With respect to procurement and fabrication, industrial capacity is limited both in forging size and supply schedule in the current market (Risk P-001). This could affect the major material for the gas and steam turbines. To mitigate this risk, the following action can be taken:

  - Book large forges and cast material at the basic design stage.

- Licensing

  In the area of licensing, one key risk is the NRC may find the radionuclide containment approach unacceptable (L-001). This risk is also tied to fuel performance goals. Mitigation strategies here include:

  - Close interaction with the NRC on this issue.
  - Ensure fuel performance goals are met so that a hard containment is unnecessary.

- Project Funding

- Project funding also affects many other risk and programmatic areas. Consistent funding is required for R&D, design, procurement, and construction in order to achieve the NGNP mission. To mitigate this risk:
    - Closely monitor funding through the DOE – may need to rework schedule and/or scope to accommodate revised funding scenarios.

18.2.3 Overall Project Risk

Overall NGNP project risk includes three main components. The project technical risk looks at the overall risk that the project will fail to achieve its technical performance objectives. The project schedule risk captures the overall risk that the project will not reach completion by required date (e.g., plant startup by 2018). The project cost risk captures the overall risk that the project will exceed the estimated cost.

Ultimately, the overall project risk considers the ability to achieve the top-level project objectives. In the simplest terms, the overall NGNP project risk can be summarized in three key elements:

- Project fails to be completed
- Prototype never meets performance objectives
- Project fails to support commercialization objectives

The overall technical risk for project is judged to be moderate. There are significant technical challenges which remain to be resolved such as fuel, high temperature materials including nitriding, and IHX development. Potential solution paths have been identified, but necessary R&D and design activities must move forward aggressively to resolve these technical risks successfully.

The overall schedule risk is judged to be significant. An initial project schedule has been developed which meets target timeline. However, the schedule is only at the preconceptual design level of detail, and its confidence level has not been assessed. There is large uncertainty due to the initial nature of the preconceptual design on which the schedule is based and because the R&D plan is still being developed. While the schedule is achievable, it requires aggressive action on major milestones in the near future.

The overall cost risk is judged to be moderate. Uncertainty in the current cost estimate stems from the lack of design detail at the preconceptual design phase and the lack of a detailed R&D plan.

The overall technical, schedule, and cost risks are judged to be manageable. The challenges are significant, but they can be adequately mitigated if prompt aggressive action is taken in key areas.

However, the overall project risk is still significant due to external programmatic factors. In terms of the three elements of overall risk to project objectives defined above, the risk is summarized as

- Risk to completion is high. This is due primarily to uncertainty regarding project funding sustainability. It is also affected by lack of clear end-user support. Licensing risk is also beyond the direct control of the project organization. Efforts are underway to mitigate these risks via the Public/Private Partnership and other channels. Success in these areas will significantly reduce these concerns.

- Risk to meeting project performance objectives is moderate. This risk is controlled by the overall project technical risk. Adequate and prompt execution of the required R&D plan is required to mitigate this risk.

- Risk to subsequent commercialization is moderate. There must be a clear alignment between the project and near-term markets. Steps must also be taken to bridge the perceived gap between potential end user time horizons and the anticipated FOAK commercial plant schedule.

Control of overall NGNP project risk requires prompt project execution, the development of a technology roadmap with off-ramps for key technology risks, a strategy to avoid funding and resource constraints, and alignment with commercial market needs. Adequate steps exist to reduce or mitigate the project risk. Nonetheless, the remaining risk is significant, and alternate paths might better reduce overall risk to an acceptable level.

As discussed in Section 21.1.1, one such alternative could be the steam cycle concept which may provide lower overall risk with a lower development cost, less technical and schedule risk, and better alignment with market applications and timing.

Table 18-5: NGNP Project Risk Summary

Risk ID No.	Risk Title	Description of Risk	Initial Risk			Approach	Management Strategy	Residual Risk		
			Prob	Cons	P/C			Prob	Cons	P/C
Design										
D-001	Nitriding	Nitriding of the materials used for PCS components and piping may occur to a large degree because 80 wt% nitrogen is involved in the secondary gas.	3	4	7	Reduce	A nitriding verification test will be performed to determine severity of the effect. Based on this test, a protection method for nitriding may be needed or an alternate mixed gas employed.	1	1	1
D-002	Composites	Architecture of composites influences its mechanical behavior, as well as the matrix and fiber nature. No nuclear component has been built up to now in composite C/C (or SiC/C). The risk is to obtain, at the end of the qualification program, poor properties (virgin, oxidized or irradiated properties) compared to those expected and required.	2	3	3	Monitor	Monitor.	2	3	3

Risk ID No.	Risk Title	Description of Risk	Initial Risk			Approach	Management Strategy	Residual Risk		
			Prob	Cons	P/C			Prob	Cons	P/C
D-003	Corrosion/ Nitriding of IHX materials	HTR past experience highlighted issues associated to oxidation (including internal oxidation), carburization and decarburization of In 617 under He plus impurities. Ni based alloys are also affected by nitriding, whose impact is dependent on pressure and temperature. The alloy is affected in depth, so the consequences of the material degradation are more important when the plate or exchange tubes are thin.	4	3	5	Mitigate	Assess the feasibility of a protective surface treatment of the IHX material (coating). Decrease in-service temperature and pressure conditions to use Hastelloy X or XR instead of 617 or 230 alloy.	2 1	3 3	3 2
D-004	Fuel Design Changes	Changes must be made to the fuel particle design during the qualification process. These changes are assumed to be the result of changing core design needs or fuel fabrication requirements, not mal-performance of the fuel.	1	4	3	Monitor	Monitor.	1	4	3
D-005	Fuel Development Funding	Program funding is insufficient to support fuel development on a schedule required to meet the NGNP operational date of 2018.	3	5	10	Avoid	Work and communicate with DOE to ensure funding is available to allow the fuel qualification program to proceed in a fashion that will meet the 2018 operation date.	2	5	8

Risk ID No.	Risk Title	Description of Risk	Initial Risk			Approach	Management Strategy	Residual Risk		
			Prob	Cons	P/C			Prob	Cons	P/C
							May need to rework schedule and/or scope of work to accommodate revised funding scenarios.			
D-006	Fuel Testing Resources	The required test reactors and fuel inspection facilities are not available to support fuel development on a schedule required to meet the NGNP operational date of 2018.	3	5	10	Avoid	Identify fuel irradiation and inspection needs and reserve the required resources as soon as possible. In addition, back-up resources can also be reserved to cover unforeseen testing needs or resource outages.	1	3	2
D-007	Fuel Performance	Fuel performance during irradiation and safety testing does not meet required limits which support the plant safety case.	1	5	5	Mitigate	The AREVA team fuel qualification strategy is designed to consider a range of fuel variables that would be expected to result in an acceptable qualification effort. The inclusion of additional fuel	1	5	5

Risk ID No.	Risk Title	Description of Risk	Initial Risk			Approach	Management Strategy	Residual Risk		
			Prob	Cons	P/C			Prob	Cons	P/C
							particle variants would not significantly increase the probability of success.			
D-008	Fuel Handling System	Fuel Server concept cannot be developed such that it operates in accordance with current refueling time estimates and with required reliability or results in unacceptable high equipment cost.	1	2	1	Accept	Should the Fuel Server System be determined to be unworkable for any of the reasons described above, the existing Fuel Cask transfer system, demonstrated at Fort Saint Vrain, can be adopted. The designation of “Marginal” was applied to cover the initial development costs, which would be lost.	1	2	1
D-009	Fuel Coater Changes	Testing of verification fuel indicates that extrapolation of the fuel coating parameters and performance results from the 6” coater cannot be successfully extrapolated to a larger production coater.	2	5	8	Reduce	Develop the production fuel fabrication process based on the use of multiple 6” coaters rather than a larger coater design.	1	1	1

Risk ID No.	Risk Title	Description of Risk	Initial Risk			Approach	Management Strategy	Residual Risk		
			Prob	Cons	P/C			Prob	Cons	P/C
D-010	Gas Circulator Feasibility	Feasibility issue due to larger machine (about 5 MW) compared to the state of the art (around 4 MW).	1	3	2	Monitor	Monitor.	1	3	2
D-011	Plate IHX design feasibility (60 MW He/He loop)	Plate IHX feasibility (with a lifetime > 10 years) is a concern due to: temperature level, corrosion, manufacturing, and thermomechanical resistance.	5	2	4	Mitigate	Urgent to launch an R&D program: For corrosion tests on base and coated materials, For thermohydraulic correlations tests, For IHX representative mock-ups, For development of a visco-plastic model (material database to be completed), and For manufacturing (diffusion bonding, brazing, ISIR). Reduce design life of 5 years for compact IHX. Potential alternatives to mitigate the risk: 1) Reduce the design lifetime 2) Fall back option	3 2	2 2	2 2

Risk ID No.	Risk Title	Description of Risk	Initial Risk			Approach	Management Strategy	Residual Risk		
			Prob	Cons	P/C			Prob	Cons	P/C
							with a tubular IHX on the 60 MW loop 3) Fall back option with a tubular IHX at lower temperature in consistency with the steam cycle option to combined cycle.	1	2	1
D-012	Tubular IHX feasibility	Tubular IHX feasibility is a concern due to: size of the module, temperature level, corrosion/nitriding, manufacturing, assembly (which are out of the state of the art)	3	3	4	Mitigate	Urgent to launch R&D program: For tube bending, tube welding, ISIR and assembly, For corrosion and nitriding tests on base and coated materials, For thermohydraulic test to assess more accurately the correlations to be used in order to optimize the design, and For IHX representative mock-ups.			

Risk ID No.	Risk Title	Description of Risk	Initial Risk			Approach	Management Strategy	Residual Risk		
			Prob	Cons	P/C			Prob	Cons	P/C
							Potential alternatives to mitigate the risk: 1) Reduced lifetime target (< 20 years): IHX replacements are planned more frequently.	2	3	3
							2) Reduced size of the IHX module to fit with the state of the art from manufacturing point of view.	2	3	3
							3) Fall back option to combined cycle option with a steam cycle at lower temperature that leads to feasible SG instead of IHX.	1	3	2
D-013	Irradiation Behavior of Graphite Grades	The graphite grade(s) retained for the fuel assemblies and reflectors do not satisfy the prediction, in particular the irradiated data are less satisfying than foreseen.	2	3	3	Monitor	Monitor.	2	3	3

Risk ID No.	Risk Title	Description of Risk	Initial Risk			Approach	Management Strategy	Residual Risk		
			Prob	Cons	P/C			Prob	Cons	P/C
D-014	Oxidation Behavior of Graphite Grades	The graphite grade(s) retained for the core support and fuel assemblies do not satisfy the requirements for important air ingress in the primary circuit.	2	3	3	Monitor	Monitor.	2	3	3
D-015	Large He Test Facility	A large He test facility (about 10 MW) is needed to do prototype testing of NHS components (valves, IHX, hot gas duct). Since such a facility does not exist, it must be built. Failure to complete the facility in time to do prototype testing would delay NGNP startup or could result in unacceptable plant performance if the NGNP was started up without prototype component testing.	2	3	3	Monitor	Monitor.	2	3	3
D-016	Qualification of Graphite	The coke used for the choice and the qualification of graphite is no more available at the time of the procurement order or not available in sufficient volume	2	3	3	Monitor	Monitor.	2	3	3
D-017	Radiation Cooling Issues	Radiation cooling is not correctly evaluated and temperatures achieved during conduction cooldown situations would be higher than foreseen.	2	3	3	Monitor	Monitor.	2	3	3
Construction										

Risk ID No.	Risk Title	Description of Risk	Initial Risk			Approach	Management Strategy	Residual Risk		
			Prob	Cons	P/C			Prob	Cons	P/C
C-001	Feasibility of the welding of mod 9 Cr 1 Mo	Welding of Mod 9 Cr 1 Mo material could show feasibility issue or the welding qualification could be longer than foreseen. If this is the case, this would have an impact on the NGNP schedule.	2	4	5	Mitigate	Continue weldability actions taking into account all the welding processes which will be required for the fabrication of the vessel system. Assess on a regular basis the progress achieved and identify as soon as possible feasibility issue. Secure partnership with welding products suppliers to facilitate and accelerate the availability of new commercial welding products.	1	4	3
Operations										
O-001	Performance of PCS	Net power of PCS cannot satisfy the design value.	3	2	2	Monitor	Monitor.	3	2	2
O-002	Operation and Control	The PCS plant cannot be operated and controlled in planned operation and control mode. We have no experience of PCS operation and control.	1	4	3	Monitor	Monitor.	1	4	3

Risk ID No.	Risk Title	Description of Risk	Initial Risk			Approach	Management Strategy	Residual Risk		
			Prob	Cons	P/C			Prob	Cons	P/C
Procurement										
P-001	Procurement of large-sized forge material, casting material	Major forging and casting suppliers shop lines are almost fully booked for coming 2-3 years due to the enormous demand. Therefore it will take more than 4-5 years at least to receive the major material of Gas turbine, compressor and steam turbine, even if they will be booked at present time.	4	4	8	Reduce	The large-sized forge material, casting material will be booked at basic design stage.	1	1	1
P-002	Feasibility of long lead procurement of Mod 9 Cr 1 Mo forgings	The only supplier of forgings in the dimensions envisioned for the reactor vessel is JSW. JSW present capabilities are not sufficient to produce the type of ingot size which is needed for a 100% forging design with mod 9 Cr 1 Mo and it is unlikely that JSW would expand its capabilities in a time frame compatible with the NGNP. There is however a risk that JSW capabilities would not be compatible with the ingot size required or that JSW would not be willing to provide this type of product.	2	4	5	Mitigate	Check with JSW during the conceptual design phase that they could provide the type of forgings envisioned and send a purchase order as soon as possible to obtain a commitment from them.	2	3	3
Licensing										
L-001	NRC Acceptance of NGNP Approach to	NRC does not accept the NGNP approach proposed for radionuclides containment (i.e., vented confined).	3	4	7	Reduce	Interact with NRC on this issue; ensure fuel performance goals are met in order to	2	4	5

Risk ID No.	Risk Title	Description of Risk	Initial Risk			Approach	Management Strategy	Residual Risk		
			Prob	Cons	P/C			Prob	Cons	P/C
	Radionuclides Containment						substantiate claim that a hard containment is not necessary.			
L-002	Acceptance of Licensing Requirements for Fuel	Fuel radionuclide retention requirements need to be set consistent with NGNP goals for a vented confinement (not accepted by NRC).	2	3	3	Monitor	Monitor	2	3	3
L-003	Acceptance of the NGNP Licensing Approach and Requirements	There are many potential risks that may occur in the adaptation of existing LWR licensing requirements to the NGNP (i.e., HTR) that could significantly impact the design and cost of the NGNP as well as its licenseability.	3	3	4	Mitigate	Proactively involve the NRC and other regulators now, to ensure issues are addressed early and to demonstrate feasibility of approach. Communication is key in order to thrive within the new licensing framework. See specific issues attached below.	3	2	2
L-004	Applicable codes and standards	The PCS cannot be permitted with non-nuclear code ASME Sec. VIII	2	3	3	Monitor	Monitor.	2	3	3

Risk ID No.	Risk Title	Description of Risk	Initial Risk			Approach	Management Strategy	Residual Risk		
			Prob	Cons	P/C			Prob	Cons	P/C
L-005	Codification of internal and IHX materials	ASME update to incorporate internal materials (alloy 800H is the prime candidate) and IHX material (617 or 230) is not developed in time or modifications proposed for these materials are not approved by the NRC.	2	4	5	Mitigate	Increase the presence within ASME Sub-Committees. Interact with ASME and NRC management to encourage the involvement of the NRC as early as possible in the process of modification of the ASME Code. Continue effort on R&D to support the Code modifications.	1	4	3
L-006	Codification of mod 9 Cr 1 Mo	ASME update to incorporate heavy section products of Mod 9 Cr 1 Mo is not developed in time or modifications proposed for this material are not approved by the NRC.	2	4	5	Mitigate	Increase the presence within ASME Sub-Committees. Interact with ASME and NRC management to encourage the involvement of the NRC as early as possible in the process of modification of the ASME Code. Continue effort on R&D to support the	1	4	3

Risk ID No.	Risk Title	Description of Risk	Initial Risk			Approach	Management Strategy	Residual Risk		
			Prob	Cons	P/C			Prob	Cons	P/C
							Code modifications. Adapt the power level and the operating parameters (or implement an active cooling system) so that PWR material could be used for HTR vessel system. In such a case, the PWR material is already codified and approved by the NRC but only for PWR application and further justification should be brought for HTR application.			
L-007	Codification of Graphite	ASME update to incorporate section about graphite is not developed in time or rules proposed for this material are not approved by the NRC.	2	3	3	Monitor	Monitor.	2	3	3

Risk ID No.	Risk Title	Description of Risk	Initial Risk			Approach	Management Strategy	Residual Risk		
			Prob	Cons	P/C			Prob	Cons	P/C
L-008	Operation of IHX vessel in non-negligible creep regime	Regulator may not accept operation of IHX vessel at temperature above the negligible creep regime, or regulator may impose unreasonable safety analysis constraints on design as a result of operation in the non-negligible creep regime. External insulation of the IHX vessel is preferred to minimize heat loss, simplify design, and facilitate ISI. However, this results in operation at 500°C which is slightly above the negligible creep regime for modified 9Cr-1Mo. (Modified 9Cr-1Mo is routinely used at temperatures well above this (e.g., 600°C) in non-nuclear service.) Note that RPV beltline is not affected by this. Part of cross vessel may be affected.	2	2	2	Monitor	Monitor.	2	2	3

Legend	
Prob =	**Probability of Occurrence**
Cons =	**Consequence of Occurrence**
P/C =	**Probability/Consequence Correlation Score**
Risk Score	**Action**
≥ 9	Avoid the risk through alternative approach, if possible.
6, 7 or 8	Identify measures to minimize the probability of the Risk occurring (and actively manage those measures) and steps to mitigate the consequences should the Risk occur.
4 or 5	Identify steps to mitigate the consequences should the Risk occur.
2 or 3	Monitor the Risk.
1	Accept the Risk.

19.0 RESEARCH AND DEVELOPMENT (R&D)

The Very High Temperature Reactor (VHTR) is uniquely suited for producing hydrogen without consuming fossil fuels or emitting greenhouse gases. Successful deployment of an advanced VHTR will depend to a large extent on the research done to anticipate and address the large number of technical risks and, at the same time, to demonstrate the advantages of the VHTR.

Although the VHTR is an unprecedented first of a kind (FOAK) system, the basic technology for the next generation nuclear plant (NGNP) has been established in former high temperature gas-cooled reactor plants (DRAGON [England], Peach Bottom Unit 1 [U.S.], Arbeitsgemeinschaft Versuchsreaktor (AVR) [Germany], Thorium Hochtemperatur Reaktor (THTR) [Germany], Fort St. Vrain (FSV) [Colorado]). In addition, the technologies for the NGNP are being advanced in the Gas Turbine-Modular Helium Reactor (GT-MHR) project, and the Japanese HTTR and Chinese HTR-10 projects are scaled reactors demonstrating the feasibility of some of the planned NGNP technology and materials. Further research and development (R&D) is needed to increase coolant temperature beyond 850 ºC and to develop the interface between the Nuclear Heat Source (NHS) and the heat utilization systems.

The objectives for R&D needs are to identify and characterize the needs for R&D work to mitigate technical risk and to resolve critical issues affecting design, fabrication, testing and operation of the VHTR plant. The systems of principal interest in this effort are the nuclear heat source (NHS) / nuclear island and the power conversion system (PCS). Creation of an R&D development plan is beyond the scope of this effort.

19.1 Approach to Define R&D Needs

The general approach for defining applicable R&D needs for the NGNP is shown pictorially in Figure 19-1. First, the objectives and scope of VHTR hardware and analytical computer codes were determined from the work breakdown structure (WBS). Next, Subject Matter Experts (SMEs) were surveyed to determine current technology maturity, R&D needs to mitigate technical risk and/or resolve critical issues, prioritize R&D needs, estimate cost and schedule, and identify facilities to perform the R&D. An adaptation of the aerospace Technology Readiness Level (TRL) approach was used to define technological maturity. The “Importance” and “Knowledge” parameters of the Phenomena Identification and Ranking Technique (PIRT) were used to prioritize R&D needs. The surveys were then compiled, compared to previous applicable VHTR work, and iterated for completeness and consistency. The risks and risk mitigation approaches identified in Section 18.0 were then used to confirm that all R&D needs have been identified. A similar approach, using a modified survey, was taken to identify R&D needs for computer codes and models.

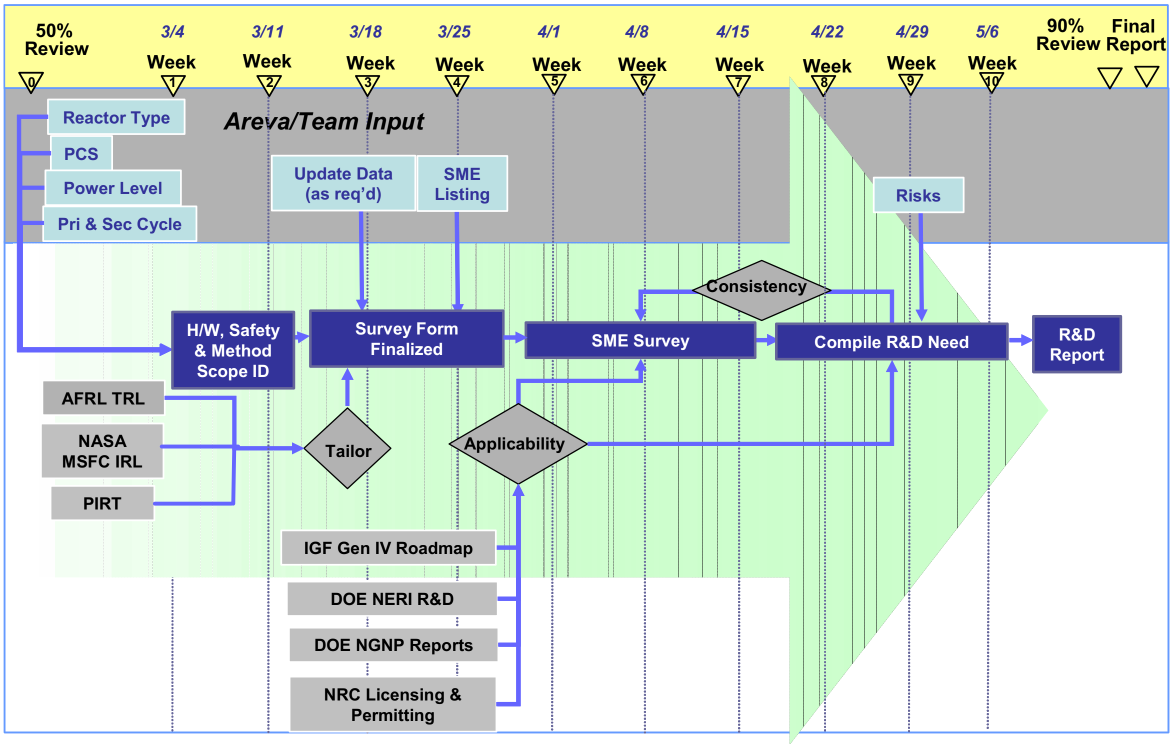


Figure 19-1: Work Flow to Identify R&D Needs.

19.1.1 Technology Readiness Level (TRL)

A technology readiness level (TRL) scale, such as that used in the aerospace industry, was adapted for use in assessing VHTR technical maturity [27], [28] .The original NASA TRL approach offers a method of subjectively quantifying the maturity of certain technologies for use in the space program. It provides a "snap shot" of program maturity at a given time. TRLs range from Level 1: Concept Conceptualized to Level 9: Mission Proven. Efforts to correlate TRL to the Nuclear Technical Maturity Assessment have been adapted [29]. This is illustrated in Table 19-1, where the TRL for the NHS and PCS are mapped to the traditional aerospace TRL. It was further tailored for identification of maturity of computer codes and design methods for the VHTR Table 19-2.

Table 19-1: VHTR NHS and PCS TRL Mapped to Aerospace TRL

TRL	Aerospace Technology
9	Has an identical unit been successful on an operational mission (space or launch) in an identical configuration?
8	Has an identical unit been demonstrated on an operational mission, but in a different configuration/system architecture?
	Has an identical unit been mission (flight) qualified but not operationally demonstrated (space or launch)?
7	Has a prototype unit been demonstrated in the operational environment (space or launch)?
6	Has a prototype been demonstrated in a relevant environment, on the target or surrogate platform?
5	Has a breadboard unit been demonstrated in a relevant (typical; not necessarily stressing) environment?
4	Has a breadboard unit been demonstrated in a laboratory (controlled) environment?
3	Has analytical and experimental proof-of-concept been demonstrated?
2	Has a concept or application been formulated?
1	Have basic principles been observed and reported?

TRL	VHTR PCS Technology
9	Has an identical unit been successful on a commercial operation in an identical configuration?
8	Has an identical unit been demonstrated on a commercial operation, but in a different system/configuration architecture?
	Has an identical unit been successful on a pilot plant?
7	Has a prototype unit been demonstrated in the operational environment with demonstration of safety features?
6	Has a prototype unit been demonstrated in a relevant environment, on the target or surrogate platform?
5	Has component/breadboard been demonstrated in a relevant (typical; not necessarily stressing) environment?
4	Has component/breadboard been demonstrated in a laboratory (controlled) environment?
3	Has analytical and experimental proof-of-concept been demonstrated?
2	Has a concept or application been formulated?
1	Have basic principles been observed and reported?

TRL	VHTR NHS Technology (Process & H/W Equipment)
9	Process integrated into operations /Equipment is commercially available or proven and in use
8	Hot Prototype off-design (for safety) demonstrated
7	Hot Prototype design basis demonstrated
6	Cold Prototype demonstrated
5	End-to-end design (flowsheet / equipment) completed
4	Hot feasibility demonstrated
3	Cold feasibility demonstrated
2	Design concept or technology application formulated
1	Identification of new design

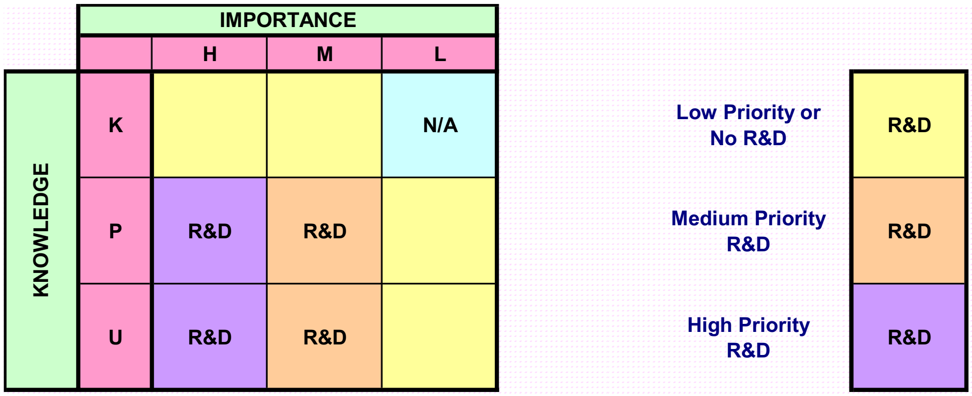
Table 19-2: VHTR Computer Code Maturity TRL Definitions

TRL	VHTR NHS Technology (Computer Code / Modeling)
9	Computer code / model proven & in use for identical applications
8	Computer code / model fully verified and validated for applications; Existing model used for different, but in scope application
7	Computer code / model verified and validated only for AOO and DBA scenarios
6	Integrated modeling (in Prototype) completed
5	Individual module modeling completed and validated at simulated operating environment
4	Individual module modeling at laboratory environment completed
3	Proof-of-concept demonstrated at laboratory environment
2	Modeling concept formulated
1	Basic principles know

19.1.2 Priority of R&D Needs

Ranking of R&D needs ultimately must consider many factors including: required time to complete the R&D, when the results are needed, cost and other resource requirements to perform the R&D, and probability of success. Detailed ranking, considering all these factors, must be performed within the context of developing the overall R&D plan. This is beyond the scope of the current effort. A knowledge-based priority ranking has been developed based on the matrix in Figure 19-2. An additional ranking reflecting R&D urgency should be developed as a next step based on when the R&D results are needed and the estimated duration of the R&D.

The Phenomena Identification and Ranking Technique (PIRT) is a systematic way of gathering information from Subject Matter Experts (SMEs) and ranking the importance of that information, in order to meet some decision-making objective. Among the various PIRT parameters, "Importance" and "Knowledge" were adapted to prioritize R&D needs. This approach is shown in Figure 19-2. From the identified "Importance" and "Knowledge" levels, one can prioritize R&D needs from low to high.



<u>IMPORTANCE</u>

High (H)	Implies that the phenomenon/model/parameter has a controlling impact on Figure of Merit (FOM). Simulation of experiments and/or analytic modeling with a high degree of accuracy is critical.
Medium (M)	Implies that the phenomenon has a moderate impact on the FOM. Only a moderate degree of accuracy is required for analytic modeling or measurements.
Low (L)	Implies that the phenomenon has a minimal or zero impact on the FOM.

<u>KNOWLEDGE</u>

Known (K)	Implies fully or almost fully known (more than 75% of what we could expect to know).
Partially Known (P)	Implies the knowledge base is moderate (25-75% of the knowledge base is established).
Unknown (U)	Implies that the knowledge base is low (less than 25% of the knowledge base is established).

Note: No assessment of knowledge is given if the importance is low (NA, not applicable, is given).

Figure 19-2: PIRT Approach to Determine R&D Needs and Priorities

19.1.3 Survey Forms

The hardware survey form and the software survey form are shown in Table 19-3 and Table 19-4 respectively. The hardware survey form is applicable to VHTR hardware such as fuel, materials, component, and power conversion system development and qualification. The software survey form is applicable to VHTR computer codes and methods development and validation.

Table 19-3: VHTR Hardware Survey Form

VHTGR H/W Survey Form

NGNP VHTGR H/W R&D Survey Form *(Design Baseline is listed in Reference)*

WBS # (Reference)		**WBS Title**			
Subject Matter Expert Name		**Email**		**Phone**	
Organization					

Current Technology Readiness Level
Identify TRL, including rationale

TRL		*TRL Definitions are listed in at Reference* *If can't decide please describe in the rationale block; if more space is required please insert "rows" or adjust "height"*
Rationale & Assumptions		

R&D Needs
Identify and characterize the needs for R&D work to resolve critical issues affecting design, fabrication, testing and operation of the VHTGR plant

Importance and Knowledge definitions are listed in Reference; Estimated Cost or Schedule in a range (e.g., $5 M to $7M or > 12 months) is acceptable
If more rows/space are needs please insert "rows" and/or adjust "height"

1	

Which Program Phase Will This R&D Support? (please check one)

Design	Construction	Initial Operations	Operate Commercially

Importance (H/M/L) *(Reference)*	Knowledge (K/P/U) *(Reference)*	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Fallback Option *(Reference for definition)*

Table 19-4: VHTR Computer Codes and Models Survey Form

VHTGR Computer Code / Modeling Survey Form				
NGNP VHTGR Computer Code R&D Survey Form *(Design Baseline is listed in References)*				
WBS # *(References)*		WBS Title		
Subject Matter Expert Name		Email		Phone
Organization				
Computer Code Usage Category (Please check all applicable) Please check all applicable, if existing code please provide code name with version				
Reactor System Analysis		Autonomous Control		
Neutronics		Materials Analysis		
Thermal-Hydraulic		Structure Analysis		
Severe Accident		PCS Analysis		
FP Transport		Heat Exchanger Analysis		
Containment Analysis		Human Factor Simulation		
PRA		Economic		
Fuel Performance				
(please add-on)				
Objectives of Modeling				
Status of Computer Code Please check applicable				
Current Code Fully Applicable	Current Code Needs Modification	Current Code Needs Major Modification	Current Code Needs Validation	No Existing Code
Current Technology Readiness Level Identify TRL, including rationale				
TRL		*TRL Definitions are listed in at References* *If can't decide please describe in the rationale block; if more space is required please insert "rows" or adjust "height"*		
Rationale & Assumptions				
R&D Needs (Description of Technical Approach) Identify and characterize the needs for R&D work to resolve critical issues affecting design, fabrication, testing and operation of the VHTGR plant				
Importance and Knowledge definitions are listed in References; Estimated Cost or Schedule in a range (e.g., $5 M to $7M or > 12 months) is acceptable *If more rows/space are needs please insert "rows" and/or adjust "height"*				
1				

Importance (H/M/L) *(Reference)*	Knowledge (K/P/U) *(Reference)*	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Validation Data Required (What kind?)	Alternative Model(s)

19.2 R&D Needs

Application of the approach described in Section 19.1 resulted in the identification of the R&D needs that are discussed in this Section. Individual survey forms appear in Appendix C; and a summary of TRL, importance, knowledge, and estimated cost and schedule for each R&D item is listed in Table C.2 of the Appendix.

The R&D needs that are key to the success of the project include:

- Fuel development and qualification, particularly irradiation and testing of compacts and mass production processes. R&D costs in this area are about $210 million.
- Materials development and qualification. This covers certain high-temperature steels, composites, and graphite selection/qualification. The associated R&D costs are estimated at $33 million.

- Components testing. A large (10 MW) helium test loop is required for prototype tests of components. This loop could cost as much as $110 million. An additional $50 million would be needed for actual hardware tests (includes a smaller 1 MW test facility).
- Computer codes & methods development/qualification. Included here are neutronics, fuel performance, heat transfer, and mechanical analysis codes. The total R&D expense is estimated at over $26 million with $8 million associated with neutronics code benchmarking to critical experiment data.
- Power Conversion System. This work covers nitriding tests and improvement of blade performance. The associated R&D costs are estimated at $10M.

In total, the R&D program is expected to cost about $440 million and span 60 months

19.2.1 VHTR Fuel Development and Qualification

Historical programs have demonstrated the feasibility of TRISO fuels containing either UCO or UO2 kernels in gas reactors at reasonable performance temperatures and burnups. Irradiation testing is currently ongoing with preliminary UCO kernels (contained in particles) produced in a pilot-scaled facility. However, the production capabilities do not exist in the U.S. infrastructure. Two risks (Risk D-005 & D-006) have been identified for fuel development. These are insufficient funding and unavailability of a test reactor and fuel inspection facilities to support fuel development on a schedule required to meet the NGNP operational date of 2018.

Fuel Kernel

Currently the VHTR TRISO kernel TRL is 4, and three “High” to “Medium” priority R&D needs for kernel materials and manufacturing have been identified.

1. The first is development of an advanced carbon source for UCO kernel production. It is estimated that it will require $5M to $10M and 12 months to produce a carbon source and test materials in a UCO kernel fabricating pilot-facility.
2. Second is development of an advanced kernel wash and dry system to cost effectively increase throughput of the kernel line without degradation in kernel quality.
3. Third is development of enhanced fuel sintering for either UCO (large fluidized bed sintering) or UO2 (static bed sintering) focusing on increased throughput and reduced cost. Costs and schedule are estimated at $15-$20M and 24 months for this development effort and the preceding effort (2).

Facilities at BWXT have been identified for performing fuel kernel R&D.

Coating

Extrapolation of fuel coating parameters and performance results for the existing six inch coater to production scale has been identified as Risk D-009. Currently the VHTR TRISO coating manufacturing process is at TRL 4 and the R&D priority is ranked as “High”. Coating materials qualification R&D need is included in the following section on compact materials.

This R&D item will examine whether the maximum coating batch capacity of the six inch coating retort that currently exists at BWXT, will economically support VHTR fuel production. Should a larger coater be required, R&D should be performed on the new coater and this would require a facility expansion. Cost estimates range from $5M, if the existing facility will support production, to $20M, if a new facility is required. It may take up to 24 months to complete the new facility.

Compact

Compact fabrication using thermosetting resins has been developed and demonstrated on a laboratory scale. However, currently-available materials have not been irradiated and performance under relevant environment has not been demonstrated. As a result the compact TRL is now at 3. Further, the priority of this R&D need was evaluated as "High".

It has been estimated that it would take 36 months and $60M to select a graphitic matrix, resin, etc. to produce thermosetting compacts and to demonstrate the performance under normal and off-normal accident conditions.

In addition, it is essential to establish compact manufacturing capabilities in the U.S. and it is recommended that it be based on the AREVA process. The other three compact R&D needs include:

1. Testing to confirm compact pressures and temperatures in order to minimize fuel damage.
2. Development of the heat treating process to ensure complete graphitization of the matrix material.
3. Perform irradiation tests on compacts to demonstrate performance for nominal and off-nominal operating conditions.

These four R&D needs will cost $40M and 36 months schedule. Expansion of BWXT fuel line for compacts is recommended.

Inspection and quality control methods

Several inspection techniques are available for fuel kernels, particles, and compacts. However, a strong correlation between as-fabricated and inspected particles and compacts and irradiation performance has not been shown in all cases. It has been identified as an "Avoid" risk with "Likely" occurrence and "Crisis" consequence for unavailability of fuel inspection facilities. Consequently, the first three fuel inspection and quality control R&D needs below have been categorized as "High Priority" while the last is "Low Priority." These R&D needs are:

1. Development of QC inspection techniques that directly relate to irradiation performance.
2. Development of techniques for large-scale production capabilities that minimize the quantity of materials that require destructive evaluation to ensure statistically acceptable fuel is produced. Techniques to be investigated could be: microfocus x-ray of particles (dimensional inspection of particle layers), mercury porosymetry (buffer density), sink-float (IPyC, SiC, and OPyC density), anisotropy measurements of the IPyC and OPyC layers, etc.
3. Irradiation testing of the compacts to ensure that as-measured attributes actually correlate to performance. This would be necessary to ensure the correct attributes are being measured and characterized.
4. Development of highly reliable instrumentation and data acquisition software will be needed to ensure fuel particle quality is built into the fuel.

These R&D needs are estimated to cost $27M and require 36 months.

As for quality control methods, most are currently available for pilot-scale fuel production with a higher TRL of 6.

Fuel mass production

Many areas of the fuel fabrication process have been demonstrated on a pilot-scale. However, some chemical processing areas or the process will require significant scale-up to meet production demands. This scale-up is not expected to be linear and product quality must be demonstrated on the larger scale. As a result, the TRL for fuel production is as low as 3.

Three “High Priority” R&D needs have been identified. These are:

1. The “scale-up” R&D should focus on kernel wash and dry, sintering, coating (assuming larger than 6" coater is required), compact matrix formulation, and compact fabrication.
2. QC techniques need to be developed with mass production in mind (please see the previous paragraph).
3. Irradiation testing will be required to confirm that fuel performance matches performance from the laboratory/pilot facilities.

It is estimated these R&D needs will cost $30M and require 30 months. The existing facilities at BWXT can be modified to develop larger-scaled production.

19.2.2 Materials Development and Qualification

The materials R&D needs will focus on testing and qualification of the key materials commonly used in very high-temperature designs. The materials R&D will address the materials needed for the VHTR reactor, power conversion unit, intermediate heat exchanger (IHX), and associated balance of plant.

There are five risks identified in the area of materials, though none has score greater than 6 (Risk Critical). There is a general understanding that the materials R&D is essential for VHTR success.

In addition, the VHTR design relies on contact conditions between different materials (metal to metal, graphite to ceramics, ceramics to metal, etc.) and R&D actions have to be performed to assess the contact conditions to avoid unexpected situations (bonding, wear, etc). As an example, the core support to reactor vessel interface is currently assumed to be a sliding interface. R&D actions are required to make sure that the helium environment (together with the contact pressure) is not likely to create a bonding effect between the alloy 800H and the 9CR1Mo materials. Tribology tests are needed on expected couples of materials in representative VHTR conditions. Dedicated facilities, for example facilities at AREVA NP and CEA, will be required. These tests were estimated for $0.5M and 18 months.

The materials development and qualification R&D needs discussion is grouped into three areas: metallic, ceramic, graphite materials.

19.2.2.1 Metallic Materials

The primary candidate for vessel materials is Modified 9Cr1Mo steel. Alloy 800H is considered for internal materials. Modified 9Cr1Mo is also a candidate only if the temperature is kept well below 750°C. As for the IHX, superalloys such as In617 or Haynes 230 are candidate materials.

Mod 9Cr1Mo has already been used in conventional power plants and is also supported by significant R&D test results from past Fast Reactor R&D programs. An R&D program has already been launched in the context of HTR ANTARES activities to complete the required input data for the final selection and the qualification program. The TRL for Mod 9Cr1Mo is 6.

For Mod 9Cr1Mo steel the R&D needs, of “High Priority,” include mechanical properties on heavy section products (base and weld metal), effects of aging and radiation, corrosion in helium environment, weldability (Risk C-001), emissivity, negligible creep conditions and creep fatigue. A specific test program on representative plates and forgings (including welded joints) will be required for component qualification. It has been estimated that the qualification of Mod 9Cr1Mo will take approximately 72 months and $4M due to the need of procuring a large forging. Due to the long lead procurement time of Mod 9Cr1Mo forgings a risk (Risk P-002) has been identified.

Mod 9Cr1Mo is covered by the ASME code up to 371ºC in Subsection NB and beyond 371ºC in Subsection NH. Subsection NH does not currently cover heavy section products (Risk L-006) and needs to be updated to cover specific aspects of Mod 9Cr1Mo. Actions have already been launched in the context of the DOE/ASME Gen IV material project to provide basis for code development. R&D efforts to support this codification should be continued.

In view of past experience in gas cooled reactor, alloy 800H is a prime candidate for metallic internals operating in cold helium. Moreover, efforts are in progress to extend its coverage up to 850°C in ASME III-NH (Risk L-005 The TRL for alloy 800H is 8.

For 800H alloy the R&D needs include:

1. Emissivity measurement under likely representative state of surface (as machined and oxidized after machining) and
2. Corrosion behavior under representative primary helium environment.

For extension of alloy 800H coverage in ASME III-NH the following items are needed:

1. Long term tests at temperature higher than 760°C,
2. Tensile tests at temperature higher than 870°C and
3. Extension to cover 60 years lifetime.

Two available nickel-based super alloys (In617 and Haynes 230) have been selected as structural materials for the IHX: In617 (NiCr22Co12Mo), which has been widely studied in the early 80's for HTR application and Haynes 230 (NiCr22W14), which has been developed more recently but it exhibits better corrosion resistance. An extensive research program has been launched in France within the framework of the ANTARES program to evaluate mechanical properties, thermal stability, and corrosion resistance in the temperature range of 700 °C to 1000 °C for extended periods. Currently the TRL of In617 is 6.

In617 and Haynes 230 R&D needs, of "Medium Priority," have been identified to address the following issues:

1. baseline mechanical property data, including creep-fatigue data,
2. long-term thermal stability,
3. effects of helium coolant chemistry on material degradation,
4. effects of 80% nitrogen-20% helium mixture on material degradation and
5. corrosion effects on mechanical properties.

The In617 and In230 R&D efforts will cost $4M and require 30 months to complete.

19.2.2.2 Ceramics

No nuclear components or structures made of composites were used for the past HTRs or for other reactor concepts. The use of composites is driven by their high resistance to high or very high temperatures (Risk D-002). An R&D program has been launched in the frame of ANTARES to explore the possible use of such materials inside the primary circuit. Thermal insulation, using composite materials, will be needed to provide thermal protection of metallic components which would otherwise be subjected to helium at very high temperatures. For the aforementioned applications, the ceramic TRL is currently at 7.

The R&D needs for applied composite materials (C/C or C/SiC composites) emphasizes qualification of material properties such as:

1. thermal-physical properties (thermal conductivity (K), coefficient of thermal expansion (CTE), heat capacity (Cp)),

2. mechanical properties including multiaxial strength,
3. fracture properties,
4. fatigue properties and
5. behavior in an oxidizing atmosphere and oxidation effects on properties.

In addition, for thermal insulation, ceramic materials qualification should be for:

1. thermal-physical properties (K, CTE, Cp) and
2. behavior under oxidation.

No control rods made of composites were used for past HTRs, or for other reactor concepts. The use of composite C/C for control rods has a low TRL of 2. Other composites such as C/SiC are also envisioned. An R&D program has been launched in the frame of ANTARES to explore the possibility of employing such composites for the control rods. SiC/SiC composites are not considered mature enough to meet the NGNP 2018 schedule.

Additional tests for control rod ceramic materials include:

1. irradiation effects on properties including irradiation induced dimensional change and irradiation induced creep and
2. tribology.

The R&D needs for ceramics are of "Medium Priority." Total cost was estimated at $4M and the schedule was 54 months.

19.2.2.3 Graphite Materials

Graphite, an essential structural material for the VHTR, will operate under significant irradiation conditions and requires a characterization in the range of expected temperatures (Risk D-013). Nuclear grade graphite was used in past HTRs programs, amassing a substantial database. These grades are no longer available (Risk D-016). An R&D program has been launched within ANTARES program to select the best candidates among the new available grades or to request the development of a new grade, and to acquire design data. The TRL of graphite materials is 7.

Nuclear graded structural graphite (PCEA, NBG17 and/or NBG18) qualification includes:

1. thermal-physical properties (K, CTE, Cp, emissivity),
2. mechanical properties including multiaxial strength,
3. fracture properties,
4. fatigue properties,
5. irradiation effects on properties including irradiation induced dimensional change and irradiation induced creep,
6. behavior under oxidized atmosphere including oxidation effects on properties (Risk D-014) and
7. tribology.

Due to schedule limits, it is recommended that graphite R&D be performed in two phases: preliminary and detailed. The R&D needs for graphite materials are of "High Priority." Total cost was estimated at $20M ($6M for the preliminary phase and $14M for the detailed phase) and the schedule is greater than 54 months.

Development of ASME and ASTM codes and standards for graphite is essential for timely application graphite for NGNP reactor (Risk L-007).

19.2.3 Components

R&D needs have been identified for the following nuclear heat source / nuclear island subsystems / components: Circulators, IHX (Tube), IHX (Plate), Isolation Valves, Fuel Handling System, Neutron Control System Drive Mechanism, RCCS, Plant and Safety Protection, Hot Gas Duct and Instrumentation.

The components of the Helium Purification System and the Shutdown Cooling System have been evaluated and no R&D needs have been identified due to similar subsystems currently in use, or were used, in various other helium cooled reactors. Qualification of the helium purification charcoal can be performed during the commissioning phase.

Circulators

Circulators up to 4 MWe have already operated in HTR reactors. The test program is dedicated to component qualification during the commissioning phase rather than as an R&D task. Planned tests ("Low Priority") include:

1. Air tests of the impeller (at scale 0.2 to 0.4).
2. Helium tests of magnetic and catcher bearings.
3. Tests of the circulator shutoff valve.
4. Full scale integrated tests.

IHXs

The R&D inputs are based on two IHX concepts: Tubular IHX for 193 MWt power conversion and Plate IHX for 60 MWt loads for hydrogen plant loop.

Small test facilities up to 1 MWt are available. Large test facilities of about 10 MWt will need to be designed and built (Risk D-015). It is estimated that it will require $20M and 30 months to build a 1 MWt test loop and $80 to $120M to build and test a 10 MWt test loop: $72M to $112M for the facility, $1M for the test article, and $7M for the test.

Tubular IHX

The Tubular IHX design is based on the extrapolation of past German experience. NGNP requirements lead to high temperature operation with an innovative secondary fluid mixture of helium and nitrogen. Risk D-012 identifies feasibility concerns on module size, temperature level, corrosion/nitriding, manufacturing and assembly (which are not state of the art).

Tubular IHX R&D needs of "High Priority" include:

1. Tests to confirm fabrication feasibility (tube bending, tube welding, nozzles on hot header, ISIR and assembly, etc).
2. Corrosion and nitriding (Risk D-003) tests on base and coated materials in a representative environment.
3. Fabrication of representative IHX mock-ups from thermo-hydraulic and manufacturing point of views.
4. Testing in representative helium and helium-nitrogen environments is recommended.

The current plan is to use a full scale mock-up for component qualification. The need for intermediate testing on sub-scale mock-ups is deemed unnecessary provided that manufacturing issues are sufficiently addressed.

Plate IHX

The feasibility of the plate IHX is a concern and a reduced lifetime is expected (Risk D-011). Primary concerns are temperature level, corrosion, manufacturing, and thermal mechanical resistance. The current TRL of plate IHX is at 2. The plate IHX R&D needs, which are "Medium Priority"," include:

1. Development of visco-plastic model (material data-base to be completed).
2. Corrosion tests on base and coated materials in a representative environment.
3. Development of manufacturing techniques (fusion welding, diffusion bonding, brazing and forming).
4. Tests on representative IHX mock-ups from both thermo-hydraulic and manufacturing point of views (diffusion bonding, brazing, ISIR).

A three step approach is recommended for component qualification, these are:

1. tests in air with sub-scale mock-ups,
2. tests in helium with sub-scale mock-ups (about 1 MWt test loop). These tests will provide a basis for recommendations on which type of concept should be used for the NGNP, and
3. final qualification on a full scale mock-up (at least for the channels and the plates) on a large test facility (around 10 MWt).

Isolation Valves

A hot gas isolation valve was designed during the German HTR development program and tested in the KVK test facilities. The corresponding valve was designed for operation in helium at 900 °C and is very close to what is envisioned for the VHTR.

Qualification of the isolation valve has a priority of "High." The two qualification steps are:

1. Elementary tests to characterize the fiber conditions, assembly techniques, spacers, etc.
2. Full scale mock-up tests in a relevant helium-nitrogen environment.

These tests should cover:

1. manufacturing parameters,
2. depressurization tests,
3. pressure loss, heat loss, support tube temperature tests in a relevant helium-nitrogen environment,
4. leak tightness tests of the valve,
5. closing and opening and
6. fatigue and creep-fatigue of specific areas.

Fuel Handling System

Currently the Fuel Server portion of the Fuel Handling System requires the most development and is judged to have TRL of 6. Risk D-008 describes the Fuel Server risks and mitigation approaches. The remainder of the Fuel Handling System components, including the Fuel Elevator, Adaptor Plate and Fuel Handling Machine, has been demonstrated at the Fort St. Vrain reactor. In addition, the HTTR reactor utilized a similar set of components. These portions of the system should be considered TRL 8.

Due to its “Low” priority, the Fuel Server system will be designed during the program. Testing of the Fuel Server system, beyond initial component testing, will be incorporated into the Fuel Handling System development testing program.

Reactor Cavity Cooling System (RCCS)

Use of an un-insulated reactor vessel coupled with a water-cooled panel heat exchanger as a core cooling mechanism for accident conditions has not been demonstrated (Risk D-017). The basic components of the system are fairly common and well understood. Proper design and sizing of the system will require a demonstrated understanding of key heat transfer parameters for the vessel wall and panel surfaces. Consequently the RCCS has a TRL of 5.

Determination of the heat transfer characteristics of the proposed surfaces for the reactor vessel and the panel heat exchanger will need to be accomplished. A large scale demonstration of the capability of the RCCS to remove reactor decay heat is recommended. The cost for these tasks (“High Priority”) is approximately $1.0M and the schedule is 24 months. Currently there is facility available at ANL which can accommodate a large scale demonstration of the RCCS.

Hot gas duct

The reference design (TRL of 6) for the primary and secondary hot gas duct is the Vee-shaped metallic concept. This design appears to be compatible with the core expected outlet temperature, subject to demonstrating that no significant hot streaks occur. The ceramic concept will be retained as a fall back option.

The hot gas duct qualification should be performed in three steps:

1. Elementary tests to characterize the fiber conditions, assembly techniques, spacers, etc.
2. Sub-scale mock-up tests, about 1 MWt in helium if possible, to validate fiber specification and ceramic spacer specification.
3. Full scale mock-up tests, around 10 MWt.

These tests should at least cover

1. depressurization tests,
2. pressure loss, heat loss, temperature of the support tube (in helium),
3. leak tightness tests of connections
4. fatigue and creep-fatigue tests (e.g., bellows, Vee-shape spacers, etc).

The cost for these tasks is approximately $4.5M, not including cost of 10 MWt test facility which is currently not available, and the schedule is 24 months. In the first stages of the design, tests should cover both the metallic and ceramic concepts. Priority is “High.”

Instrumentation

NGNP will be the test bed for testing and validating HTR technology and specific instrumentation might be required for operation at high temperature. The detail of this instrumentation (in particular the operating conditions) will be a function of the type of testing and experiments envisioned and will depend also on the monitoring strategy.

For neutron flux detectors some R&D and qualification efforts may be desirable to select detector technology and verify adequate sensitivity and lifetime.

For temperature measurements the standard thermocouples used in nuclear plants today are capable of measuring operating temperatures up to 1200 ºC. Monitoring accident conditions may require the use of Pt-Rh thermocouples for operation at higher temperatures. These types of thermocouples are not used today and limited data about their reliability in nuclear environments exists. R&D will be needed to qualify Pt-Rh thermocouples for use in the NGNP, particularly if measurement of temperatures within the core is desired. It will cost $2M to develop detector technology.

19.2.4 Computer Codes and Methods Development and Validation

A summary of computer codes that may potentially be used for VHTR analysis is provided in Table 19-5. The applicability and status of the subject codes are also shown in the table. In the following sections R&D needs for development and qualification of each code are presented.

Table 19-5: Summary of Potential Computer Codes for VHTR Applications

Codes	Categories	Objectives	Code Status
MANTA	Reactor System Analysis	Calculation of main system parameters (temperature, pressure, flow rate) of the HTR plant during all transient (normal, abnormal) when the primary coolant flows in forced convection, in order to define plant operation and control and to provide load data for primary components. Possibility to calculate generalized natural convection.	Fully Applicable Needs Validation
RELAP5-3D	Reactor Systems Analysis	Best-estimate system analysis coupled to CFD models for Generation IV, including gas reactor concepts. RELAP currently is a principal tool for LWR safety analyses.	Needs modification Needs validation
MCNP	Neutronics	Reference steady state core calculation for all type of cores.	Fully Applicable Needs Validation
NEPHTYS	Neutronics	1) Reactivity effects (first criticality, moderator and doppler coefficients, control rod worth, reactivity loss versus depletion) 2) 3D neutron flux and nuclear power distribution within the reactor core 3) 3D burnup distribution and nuclide inventory for back-end cycle and decay heat issues.	Needs Modification Needs Validation
MONTEBU RNS	Neutronics	Reference depletion and decay heat core calculations for all types of cores.	Fully Applicable Needs Validation
CABERNET	Coupled Neutronics/ Thermal	Reactivity, power and temperature, burnup and fluence distribution calculation in steady state and transient conditions for block type cores (input to fuel performance assessment).	Fully Applicable Needs Modification Needs Validation
STAR-CD	Thermal-Hydraulic	Determination of: 1) thermal loadings on the components (vessels, internals, fuel…) during normal or upset conditions, 2) the thermal behavior of the core, 3) the mixing inside the primary system, 4) heat losses and performances of components, 5) flow repartition across the components and 6) pressure shock waves.	Fully Applicable Needs Validation
FP Transport	FP Transport	Transport of radio contaminant species from the fuel block graphite walls into the primary coolant in normal operation and up to the environment in case of accidents.	Needs Major Modification

Codes	Categories	Objectives	Code Status
Structural Mechanics	Structural	Assessment of component behavior under normal operation and accident mechanical and thermal loadings.	Needs Modification
ATLAS	Fuel Performance	Assessment of coated particles performance during normal operation and accidental conditions. Calculation of the failure fraction and fission product release rate from a fuel load in normal operation or accidental conditions	Needs Modification Needs Validation

19.2.4.1 Reactor System Analysis Code

MANTA

Global validation of MANTA currently consists of code-to-code benchmarking: comparisons with CATHARE from CEA (France), LEDA from EDF (France), ASURA from MHI (Japan), REALY2 from GA (USA) and RELAP5-3D from INL (USA) have already shown good agreement. Qualification against experimental data is also progressing (EVO loop, HE-FUS3 loop and PBMM). Nevertheless additional benchmarks against experimental data are required. Some facilities that could provide valuable data have been identified: namely, HTTR reactor in Japan, HTR10 reactor in China, SBL-30 loop in the USA (SNL). The qualification of component models will follow from the qualification tests of the components. The core model qualification follows from comparison with other codes and with experimental results. Further, experimental data from HTTR and HTR-10 safety tests and from SBL-30 loop is required.

RELAP

The U.S. DOE sponsors RELAP5 code development at the INL. It is expected that this support will continue. Development needs are highlighted in the report INEEL/EXT-04-02993. Validation beyond that identified in this report and consistent with that planned for MANTA should be pursued.

19.2.4.2 Neutronics Codes

All neutronics codes identified for VHTR application have a TRL of 6. The R&D needs are mainly for code qualification against experimental data.

MCNP and NEPHTYS

The R&D needs for both MCNP and NEPHTYS are of "High Priority."

1. The approach for qualification consists of comparing results against Monte-Carlo reference calculations and benchmarking against the few available experimental data (FSV, HTTR). Thus new dedicated critical experiments, with an asymptotic spectrum representative of the expected prismatic fuel assembly and core, with full access to pin-by-pin power distributions, and control rod and burnable poisons worths are needed.
2. Experimental data of neutronic characteristics (spectrum, fission and capture rates) at the interface between a prismatic fuel assembly and a graphite reflector assembly.

Data from FSV and HTTR first criticality testing can be applicable to MCNP and NEPHTYS code qualification.

The cost for neutronics code qualification has been estimated at $8M and schedule of 36 months, which includes the procurement of the fuel assemblies for the critical mock-up. The modification of the existing neutronics code for VHTR will cost $0.5M and 12 months.

MONTEBURNS

The R&D needs for MONTEBURNS are "High Priority."

1. Experimental results of fuel irradiation experiments (compacts or pebbles) at representative burnups, temperatures and fluences.
2. Experimental results of decay heat at short term (<100 hours) for representative fuel composition and burnup.

The cost to qualify MONTEBURNS is estimated at $2M with a schedule of 24 months.

CABERNET (=NEPHTYS / STAR-CD)

The TRL of the CABERNET code is 6 and the R&D needs are

1. Enhancement of capabilities for the calculation of transient analyses and
2. Experimental data of coupled power and temperature distributions obtained on representative fuel assembly geometry. If not achievable before NGNP: (a) Partial qualification data (e.g., burn-up measurements on fuel columns after irradiation in HTTR, which can provide a code/experiment comparison on the axial power distribution of a cycle); (b) Additional power margins will be necessary for initial operation of NGNP, to account for the uncertainty on the coupled neutronics-thermo-fluid dynamics calculation; (c) Need to provide in-core measurements of power and temperature distributions in NGNP for qualification of coupled calculations; (d) R&D needs for developing appropriate sensors for in-core measurements (never performed in HTRs). This code qualification can be performed during commissioning phase.

19.2.4.3 Thermal/ Hydraulics/ Pneumatics Codes

STAR-CD

The current TRL of the STAR-CD code is 6. Code development and qualification R&D needs are evaluated at a "High" Priority.

- Development of graphite oxidation model for air ingress transients on reactor internal structures.
- Qualification of:
  - conduction cooldown models on representative geometry, materials and temperature,
  - turbulence and mixing on representative mock-ups in critical areas (lower and upper reactor plena, hot gas duct, core bypass, IHX collectors) and
  - graphite oxidation models with selected graphite grades in representative operating conditions.

Several predecessor tests performed with different graphite grades at CEA and FZJ. NACOK experiments within the European RAPHAEL project (coupling of graphite models with thermo-fluid dynamic behavior) can be applied for STAR-CD qualification.

The qualification of STAR-CD code will cost $1.8M and 18 months and for development is $0.2M and 12 months.

19.2.4.4 Fuel Performance Models and Codes

ATLAS

The R&D need for ATLAS development/modification is to improve the diffusion and the coatings corrosion modeling. For the ATLAS code it has been estimated to cost $1.5M and 24 months for code development/modification.

For code qualification the heat-up experiments of irradiated fuel particles at relevant operating conditions (burnup, temperature, and fluence) are required to anchor the developed code. The estimated cost for qualification of ATLAS is $5M and schedule of 30 months, which includes two irradiation and heat up tests. In addition, there is an R&D need to develop the UCO models. Both development and qualification efforts for ATLAS have a "High" priority.

19.2.4.5 Other Codes

Fission Product (FP) Transport

The FP code has a TRL of 4. The R&D needs of the FP Transport code include development of models for:

- assessment of product activation in the primary circuit (in particular tritium and 14C),
- radio-contamination distribution in the primary circuit, making distinction between circulating activity, plated out / deposited activity and purification system, for both normal operation and accidental situations,
- radio-contamination releases outside the primary pressure boundary and
- radio-contamination releases in the environment during accident scenarios.

This FP transport code development along with the experimental work will cost $6M and 60 months. The experimental aspects are not included in this estimate.

It is also recommended to develop a mechanical analysis code for the NHS.

Structural Mechanics

Among all applicable computer codes the structural mechanics code has the lowest TRL of 3. The main tools for structural analysis exist, but specific modeling and correlations for NGNP geometry and materials have to be developed. This work (priority of "High") includes:

1) incorporation of constitutive laws for materials and developing numerical models
2) seismic modeling of a block-type core
3) fluid structure interaction and flow-induced-vibration methodology, and
4) leak-before-break methodology.

This effort is estimated to cost $1M and take 18 months to complete.

19.2.5 Power Conversion System

The major components of the PCS, including He/N_2 Cycle Control and Ducting, Heat Recovery Steam Generator, Steam Cycle and Generator and Electrical Equipments have a very high TRL (8 to 9). No R&D needs have been identified. However the turbo-machinery in the Brayton Cycle has been evaluated at TRL of 7.

Nitriding of metals will occur when exposed to hot nitrogen (Risk D-001). This nitriding process tends to embrittle metals which could lead to failures of turbine blades and pressure boundaries such as boiler tubes, gas shells, etc. The need to experimentally determine the degree of nitriding that occurs in potential PCS materials, and to quantify the effects of temperature on nitriding, has been identified. This R&D need is not only for turbo-machinery, but also for IHX (Tube) and Brayton cycle gas duct.

In addition, R&D is also needed for compressor blade performance in order to ensure high efficiency, mitigating the risk of lower than expected PCS efficiency (Risk O-001).

The total R&D efforts for PCS will cost $10M and take 18 months.

19.3 Possible Sources for R&D Tests

Using the R&D needs identified in Section 19.2, existing sources for performing some of the test programs were identified. These sources appear in Table 19-6. This table should not be regarded as an exhaustive list of test facilities, but as a starting point for detailed R&D planning.

Table 19-6: NGNP Test Facilities

R&D need	Description of the experimental means required	Sources of R&D	Comments
FUEL			
Fabrication			
Laboratory scale fabrication	To test the industrial fabrication process with limited flow of fuel material in order to be able to sreen a large number of fabrication parameters in order to improve the understanding of phenomena and optimize the process		
Coated particles		CEA Cadarache BWXT, ORNL	
Compacts		CERCA	
Industrial pilot line	Main components at the industrial scale in order to • finalize the definition of the industrial process • qualify a fuel representative of industrial production		
Coated particles		BWXT	6¨ diameter coater. Industrial coater might have larger diameter
Compacts			
Characterisation	Measurement of the key geometrical, physical, mechanical, thermal, chemical & micro-structural parameters of the fuel & of its materials & of the evolution of these parameters under irradiation		
Before irradiation		BWXT, Under development in CEA & AREVA	
After irradiation			
Fuel qualification			
Irradiation	• Online monitoring of fission gas releases in order to be able to count the number or failed particles • Test rig with sufficient capacity to have a statistically sufficient number of particles for proving the target performance of the fuel	SIROCCO (OSIRIS, CEA Saclay)	Cannot receive a sufficient number of compacts for acceptable statistics for qualification. Sufficient for lab. scale process validation
		HFR (Petten, Netherlands) ATR (INL), HFIR (ORNL)	
Safety testing	Out-of-pile heat-up of an irradiated fuel element with online fission gas monitoring & periodic solide fission product measurement	KÜFA (ITU, Karlsruhe), HFEF (INL), ORNL	

R&D need	Description of the experimental means required	Sources of R&D	Comments
MATERIALS			
Apart from standard laboratory means for material characterisation, the following specific means are necessary for studying the impact of HTR environment on materials:			
Corrosion studies			
In impure He atmosphere	Vessel material, high temperature metallic materials, graphite, composites		
Atmospheric pressure	High temperature (up to 1000°C), precise control of impurity content		Taking into account the number of materials to be examined (including the variability of composition of materials within the range of their specifications), the number of temperatures & atmosphere compositions to be screened & the length of each test, several loops must be operated in parallel, in order to be able to finalize the selection of the most appropriate materials for HTR operating conditions & to qualify the selected material within a reasonable period of time.
Uniform corrosion		CORALLINE, CORINTH (CEA Saclay) ESTEREL (EDF Les Renardières) AREVA Le Creusot	
Corrosion + creep		CORSAIRE, FLAMENCO (CEA)	
Corrosion + low cycle fatigue		CEA Pierrelatte	
Normal operating pressure			Verification of the absence of influence of the absolute pressure performed in the component qualification loops
In air ingress situations	Graphite & composite oxidation	OXYGRAPH (CEA Cadarache) THERA, INDEX (FZ Julich)	
Tribology	Friction and wear in high temperature representative He atmosphere	He tribometers in AREVA Le Creusot & CEA Cadarache	
Irradiation • Vessel material • Graphite • Composites (control rod cladding, possibly internals)	Irradiation test rigs • In a reactor providing a sufficient level of fast flux in order to obtain a representative neutron damage in a reasonable period of time • Maintaining high temperature conditions (between 400 & 1000°C) on material samples • Sufficiently large to accept a large number of samples in order to take into account the variability of the material, possibly also to irradiate together different material grades and to accept samples which are large enough to satisfy ASTM requirements for mechanical testing	An irradiation of vessel material have been performed and an irradiation of graphite is ongoing in HFR (Petten, Netherlands). Graphite irradiation is planned in OSIRIS (CEA, Saclay), including a test with in-situ irradiation creep measurement. Also, testing could be done in ATR (INL), HFIR (ORNL), and/or the MIT test reactor..	

R&D need	Description of the experimental means required	Sources of R&D	Comments
COMPONENTS AND SYSTEMS			
Plate IHX: selection of design concept, validation of the design and qualification by a step by step approach			
1st step: separate effect tests:impact of different design options on the performance			
Thermo-mechanical behavior	Representative operating temperature of the IHX and representative temperature transient. Air atmosphere is acceptable	CLAIRE loop being presently upgraded in CEA Grenoble (900°C, cool-down transients of 300°C in 5 sec., and heat-up transients of 300°C in 120 sec).	
Heat transfer performance	Representative fluids Representative geometry Representative temperature	• Tube geometry, He and He and He+N_2, 900°C (AREVA Le Creusot) • Representative geometry, air (CLAIRETTE loop, CEA Grenoble) • Representative geometry, He, 500°C (HE-FUS3 loop, ENEA Brasimone)	
Homogeneity of flow distribution	Large flow, room temperature & air acceptable	PAT loop available in EdF Chatou	
Corrosion	Representative atmosphere with controled impurities	See material corrosion loops	
2nd step: validation of the selected design	He loop providing at the same time most of the representative operating conditions		Detailed design finalized for a 1 MW He loop, 950°C, full pressure, controlled impurities, HELITE, to be built in CEA Cadarache. The flow is too small to have a representative flow distribution in the headers
3rd step: IHX qualification	Large He loop providing at the same time all the representative operating conditions allowing the qualification of an IHX module (10-20 MW)		
Tube IHX	Only final qualification necessary on the large He loop		
Valves, hot gas duct	Qualification on He loops		
Circulator	Air tests of the impeller (at scale 0.2 to 0.4)	Loops available in manufacturer facilities	
	He high temperature test of magnetic and catcher bearings	FLP 500 bench, Zittau University (Germany)	
	Integrated scale 1 test		Either dedicated loop at operating temperature (400°C) with very large He flow or during NGNP commissioning tests

R&D need	Description of the experimental means required	Sources of R&D	Comments
COMPUTER CODE QUALIFICATION			
Neutronics	Critical experiment with the possibility of getting an asymptotic neutron spectrum as well as transition spectrum at the interface of the core and the graphite reflector	Feasibility of a representative critical experiment in the MASURCA zero power reactor in CEA Cadarache assessed	
	Isotopic analysis of fuel irradiated to very high burn-up	Undertaken by ITU (JRC Karlsruhe) for a pebble irradiated to 15%FIMA in HFR, also INL	
Thermo-fluid dynamics	Representative mock-ups for critical areas, for instance		
	• Mixing of cold and hot streaks in core outlet plenum	MIR facility (INL)	Validation of turbulent mixing but not of the thermal diffusion between hot and cold streaks
	• Conduction cool-down: validation of modelling		
	- Decay heat release through internal structures to the vessel outer wall		
	- Radiative heat transfer in the reactor cavity and natural circulation in the RCCS	NSTF (ANL)	
	• Flow in IHX headers (see IHX)		
Neutronics / Thermo-fluid dynamics coupling	In core flux and temperature instrumentation in NGNP		To be developed
System analysis	Comparison with data on gas loop transient operation	Many data exist on transient operation of gas systems in which same type of phenomena as in NGNP occur (EVO loop (Germany), HE-FUS3 (ENEA), Micro-model (South Africa), HTR-10, HTTR.	

20.0 SPECIAL STUDIES

The DOE-AREVA contract governing NGNP preconceptual work specified that the design be an adaptation of ANTARES, the AREVA HTR design. ANTARES is an indirect cycle, 600 MWth prismatic graphite block reactor that, via an intermediate heat exchanger, is coupled to a combined cycle gas turbine (CCGT) power conversion system.

In this context, the following special studies were performed as part of the preconceptual design scope of work:

1. Reactor Type Comparison Study
2. Prototype Power Level Study
3. Power Conversion System Study
4. Primary and Secondary Cycle Concept Study

Thus, the results of two studies did not affect the adaptation of the ANTARES design to NGNP because the key design features are set by contract; however, the results of two studies did affect the adaptation and their results have been integrated into the NGNP preconceptual design. The results of the reactor type comparison study and the power conversion system study, nevertheless, will be subsequently used by the DOE as input to their overall selection process for NGNP technology.

The reports for these special studies are stand alone documents and are contained in their entirety as appendices to this report. Key results from these studies are presented in the following sections.

20.1 Reactor Type Comparison Study

The Reactor Type Comparison Study compared the prismatic reactor concept to the pebble bed reactor concept. The report identifies the most important discriminating criteria between the two concepts and provides an assessment of the important technical, operational and maintenance differences and the important developmental risks for each. The report concludes that the prismatic reactor concept best fulfills the needs of the NGNP Program and should be selected as the NGNP demonstration reactor. (This conclusion is strictly limited to reactor type and does not endorse any particular power conversion system to which the reactor may be coupled.)

20.1.1 Study Objectives and Assumptions

The main study objective was to provide an answer to the main question posed, namely:

What type of reactor should NGNP be?

The main question was only answered after appropriate comparisons have been made for each option with respect to the relevant NGNP functions and requirements and a detailed assessment of the key discriminating criteria.

Furthermore, a consideration was given to the future commercialization aspects of the chosen type of reactor. Commercialization of HTR technology is the real NGNP success criterion that can only be measured by the extent of HTR deployment in the decade following NGNP startup and operation.

For the purposes of the study, it was assumed that the NGNP plant is a full-sized demonstration plant. This was consistent with the finding of the Power Level Trade Study Report.

Furthermore, this study does not assume a fixed power level; rather, it assumes that each reactor pebble bed or prismatic – has been optimized for its mission at its maximum achievable power level.

Also, the comparison between pebble bed and prismatic options in this study does not assume a given design for each technology. Granted, the pebble bed offering of PBMR and AREVA's prismatic offering (ANTARES) offer good starting points and information, nevertheless, they should be viewed as examples of available HTR technology. Hence, the comparison should be more accurately considered as more a "generic" comparison of pebble and prismatic technology. Furthermore, this assessment is limited to reactor type and confined to the envelope defined by the reactor vessel. Hence, the reactor as viewed herein is considered a "universal" heat source that can be connected to the application of choice.

20.1.2 Key Discriminator Review Summary

In summary, based upon the results of the key discriminator review, the prismatic reactor offers the following key advantages over the pebble reactor alternative:

- Greater economic potential
- Higher power level and passive safety
- More useable power
    - i.e., less parasitic power loss
- Greater design flexibility
- Higher degree of license-ability
    - Concept previously licensed in the USA (FSV)
- Higher degree of predictability
    - Core performance
    - Scheduled outages
    - Less chance of forced outages

20.1.3 Reactor Type Study Conclusion

The report concludes that the prismatic reactor should be selected for the NGNP because it represents the best technological foundation for a commercially attractive, multi-use high temperature reactor concept. Of all the benefits the prismatic reactor offers, it is its superior power level capability that makes it most attractive.

20.2 Prototype Power Level Study

This special study was conducted to answer the following questions:

- What should be the rated power level of the Nth of a Kind (NOAK) commercial VHTR module?
- Given the desired power level of the commercial VHTR module, what should be the rated power level of the NGNP prototype plant?
- In order to demonstrate commercial scalability of an associated hydrogen production plant, what is the power requirement for a demonstration plant to be associated with the NGNP reactor?

The study examined criteria that were selected because of the provide insight they would provide relative to these many faceted questions. The criteria were separated into two groups:

Key Discriminating Criteria	Subordinate Criteria
Market View	Core Neutronics
Economic Considerations	Fabrication Issues
Plant Safety Limits	Component Feasibility
Licensing Issues	Plant Flexibility and Operability
Demonstration of Passive Safety Features	Research and Development
Hydrogen Plant Process Heat Requirements	

20.2.1 Commercial Plant Power Level

What should be the rated power level of the Nth of a Kind (NOAK) commercial VHTR module?

The most likely commercial applications for the VHTR will entail the use of process heat and electricity in various combinations. A principal anticipated commercial use of the energy from the VHTR will be the production of hydrogen. However, AREVA internal studies have also identified a number of other potential industrial applications for the VHTR. All considered applications for the VHTR are listed below in Table 20-1.

Table 20-1: Potential Industrial Applications of the VHTR

1. Coal to Liquids	8. Industrial Process Heat Applications
2. Oil Sands	8.1 Steel
3. Oil Shale	8.2 Alumina and Aluminum
4. Coal Gasification – "Clean Coal"	8.3 Chlorine VCM and PVC
5. Hydrogen Production	8.4 Ammonia and Fertilizers
6. Petroleum Refineries	8.5 Chemical Platforms
7. Electricity Production	9. Biomass
	10. Water Desalination

The power requirements for the above commercial applications were examined in detail and the result of that examination concludes that, in general, there is no apparent constraint on reactor size relative to the application. In most cases, the optimal size for a commercial VHTR is the largest capacity permitted within the design constraint of passive safety.

Modular VHTR's rely on conduction and thermal radiation in their passive safety features for decay heat removal. The thermal performance of the plant during a loss of active cooling is dominated by four items: the geometry of the plant, the thermal energy stored in the core at the beginning of the event, and energy (the decay heat) that is generated inside the core, and the heat transfer properties of the core (graphite).

AREVA performed parametric studies to evaluate the sensitivity of the results of the limiting design basis accident (depressurized conduction cool down) to the key influencing parameters; namely, core power level, core inlet temperature, and graphite conductivity in terms of an equivalent change in reactor power. The results of the study support the conclusion that a maximum reactor thermal power rating of 565 MWth should be acceptable while allowing some margin for uncertainties.

Based on the above, the commercial VHTR module should be designed to operate at the maximum safe power level; and, based on the AREVA's evaluation of plant safety limits, that maximum power level is 565 MWth.

20.2.2 NGNP Power level

Given the desired power level of the commercial HTGR module, what should be the rated power level of the NGNP prototype plant?

The NGNP prototype plant should be designed and operated at 100% of the planned commercial power level, 565 MWth for the following reasons:

- It demonstrates at full scale the VHTR passive safety features
- The licensing experience is directly transferable to the commercial VHTR
- First-of-a-kind engineering costs can be shared
- Lessons learned directly transferable to the commercial VHTR

From a reactor vendor perspective, participation in the NGNP project, particularly the latter stages where significant sharing of the costs is anticipated, depends on a favorable balance of the cost with the benefits gained through such participation. Some of these benefits are difficult to assess from an economics standpoint, such as the perception of industry leadership gained through participation. Others are easier, particularly those related to the applicability of costs incurred for the NGNP that are directly transferable to the commercial fleet. Chief amongst these benefits is the ability to complete first-of-a-kind engineering tasks in a cost share manner. This benefit is maximized if the power level of the NGNP reactor is equal to the power level of the eventual commercial plant. Any difference in power level will reduce this benefit. The reduction in benefit will increase dramatically as the power levels diverge. Thus, from a reactor vendor standpoint, there is considerable incentive to have the NGNP built as a full size demonstration plant.

Deployment of the NGNP provides an opportunity for eventual end-users to benchmark key cost data that will aid the decision making process. These costs may include capital cost data, construction costs, costs of operation and maintenance, and fuel cycle costs. Design of the NGNP at any power level other than 100 percent of the commercial plant will make these benchmarks less directly applicable, thus less useful.

Table 20-2 presents a graphic summary of the NGNP power level requirement evaluation. For each study criteria, an indication of the recommended commercial and/or NGNP reactor powers is stated based on the evaluation results. In addition, a qualitative assessment is provided of the potential impacts of a reduction in the NGNP rated power as a fraction of the projected commercial plant. These assessments are based largely on the expertise of the personnel involved in the evaluations. Comments are also provided which summarize the key findings of the individual evaluations.

Table 20-2: NGNP Power Level Evaluation Results

Study Criteria	Recommended Power Level	Impact of NGNP as a Fraction of Full Size				
		100%	75%	50%	25%	20MW
Market View	**Max. Practical**	Base	Moderate	Major	Major	Major
Economic Considerations	**100%**	Base	Major	Major	Major	Major
Plant Safety Limits	**Max. of 565 MWth**	Base	Minor	Minor	Minor	Minor
Licensing Issues	**100%**	Base	Moderate	Major	Major	Major
Demonstration of Passive Safety Features	**100%**	Base	Major	Major	Major	Major
Core Neutronics	No Limitations	Base	Minor	Minor	Minor	Minor
Fabrication Issues	>50%	Base	Minor	Minor	Moderate	Moderate
Component Feasibility	No Limitations	Base	Minor	Minor	Minor	Moderate
Plant Flexibility and Operability	No Limitations	Base	Moderate	Major	Major	Major
Research and Development	100%	Base	Minor	Moderate	Moderate	Moderate
Hydrogen Plant Process Heat Requirements	No Limitations	Base	Minor	Minor	Moderate	Major

Based on the foregoing arguments, it follows that the NGNP prototype should be a full-rated design, though it may initially be licensed to some fraction of that design capability. While distortions in pricing may result from a first of a kind vs. Nth of a kind installation, this approach will nonetheless present the best opportunity of achieving these cost benchmarks. These benchmarks will play a key role in assuring the eventual commercial acceptability of this reactor technology.

In conclusion, summary, economic considerations from both the reactor vendor and end user standpoints support the construction of the NGNP as a full sized demonstration plant.

20.2.3 Hydrogen Plant

In order to demonstrate commercial scalability of an associated hydrogen production plant, what is the power requirement for a demonstration plant to be associated with the NGNP reactor?

The hydrogen process requirements impact the NGNP power level by establishing the thermal and electrical needs for the hydrogen engineering demonstration. In order to determine these requirements, an assessment was made of the expected power requirements for a commercial-scale hydrogen process.

For the Sulfur-Iodine Process the most developmental and challenging portion of the process is the HI Decomposition and a full-size HI decomposition process train should be demonstrated. Sizing the balance of the SI Process to this capacity would result in a hydrogen process energy requirement of 60 MWth and 20 MWe. For the high temperature electrolysis process, the demonstration facility would consist of one train of electrolyzer modules, requiring 4 MWe and 1 MWth of process heat.

20.2.4 Recommendations

The evaluations of the key discriminating criteria resulted in the following answers to the three study questions:

1. The commercial VHTR module should be designed to operate at 565MWth.
2. The NGNP prototype plant should be designed and operated at 100% of the planned commercial power level, that is, 565MWth.
3. The hydrogen generation demonstration loop using the SI process will require 60MWth of process heat and 20MWe from the power conversion system while the hydrogen demonstration loop using the HTE process will require only 1.2 MWth of process heat and 5MWe from the power conversion system.

20.3 Primary and Secondary Cycle Concept Study

The Primary and Secondary Cycle Concept Study establishes the basic NGNP operating parameters for the primary and secondary cycle and establishes the reference configuration for NGNP preconceptual design adaptation. Furthermore, it enhances the basis for NGNP Design Baseline.

The main objective of the study is to answer the following questions:

- What is the recommended reactor Tout?
- What is the recommended reactor Tin?
- What should the system configuration be? And,
  - Should the heat supply to the hydrogen process be in parallel or in series with power generating system?
  - How many loops should the system have?
- What is the secondary side Thot and Tcold?
- What are the primary and secondary system pressures?

The answers to the above questions are driven by the following high level NGNP objectives; namely:

- The demonstration of scalability to commercial scale electricity and hydrogen production
- The demonstration of advanced hydrogen production processes (i.e., SI and HTE), and
- Achieving initial NGNP operation by 2018

Additionally, a hierarchal decision process was followed that addressed the need to make multiple, sequential, and interdependent decisions about parameter selection based upon the governing design considerations: feasibility and risk, safety, performance, flexibility, cost and schedule.

20.3.1 Reactor Outlet Temperature

The reactor outlet temperature selection is based on a tradeoff between H2 process performance and NI/H2 plant feasibility. The nature of the SI process for hydrogen generation favors higher process temperatures (900ºC or higher) which governs process efficiency and affects equipment sizing and selection. The HTE process requires a somewhat lower process temperature (800ºC) for optimum efficiency. Conversely, nuclear heat source feasibility is very temperature dependent and, above 900ºC, feasibility issues become dominant. Hence, it was determined that 900°C reactor outlet temperature best balances NHS and H2 process concerns.

20.3.2 Reactor Inlet Temperature

Given the reactor outlet temperature of 900ºC, what is the recommended reactor Tin?

Tin was selected based on an assessment of inlet temperatures ranging from 400-600°C with emphasis on 500°C, 525°C, and 550°C. The key drivers in its selection are addressed below while the impact of other lesser drivers is found in the main report.

The reactor inlet temperature is a tradeoff between core design, vessel and internals materials considerations. An increased reactor outlet temperature requires larger inlet temperature increase in order to maintain normal fuel temperature. Normal operating fuel temperatures are inversely affected by the temperature rise across core and, to counteract this effect, an increase in inlet temperature relative to the ANTARES base design (400ºC to 500ºC) becomes necessary to maintain margin.

More importantly, the reactor inlet temperature directly affects the reactor vessel operating temperature. Of prime concern is the temperature of the reactor vessel metal because of the potential to operate in the creep regime. For reactor inlet temperatures > 450°C, thermal protection for the vessel may be required and modest thermal protection is expected to be adequate at 500°C.

Based on the above, a reactor inlet of 500°C was selected because it provided greater margin for core design while still being acceptable relative to reactor vessel metal temperature.

20.3.3 System Configuration

20.3.3.1 Should the H2 and PCS heat loads be in series or parallel?

Two basic options were considered: an arrangement with the H2 IHX and PCS in series; and, an arrangement with them in parallel. Major considerations that were factored into the decision are the impact on H2 process and PCS performance, plant efficiency, maneuverability, availability, maintainability and NHS feasibility. Other discriminators are operational flexibility and flexibility for component testing.

Operational flexibility is greater for parallel loops because H2 and PCS flow rates and temperatures can be varied, startup and shutdown are not tightly coupled, H2 load rejection does not impact the PCS, and PCS load changes do not significantly affect the H2 process. Furthermore, testing flexibility is hampered by series configuration because of tight interface requirements between H2 process heat supply and the PCS. Parallel configurations can easily support the testing of different process IHX designs and circulators. Again, flow rates and temperatures can be controlled independently in parallel configuration

Based upon these considerations, a parallel configuration selected because it is more compatible with energy needs of H2 process and PCS, it is most feasible technically, and it affords increased operational flexibility, and, increased testing and demonstration capability

20.3.3.2 How many loops should the system have?

The question of "How many loops should the system have?" is complicated and is driven primarily by the capability/feasibility of IHX technology. At a minimum, the system needs two loops – one for the PCS and one for the H2 loop. The major consideration is feasibility and other discriminators, such as cost, performance, and flexibility, are subordinate.

At issue is the feasibility of a single IHX to transfer 100% of the plant thermal load to the secondary plant. AREVA work on ANTARES has considered compact heat exchanger design technology as a potential solution; but, the technology has higher temperature impacts, requires major development and the time would be required is not consistent with the needs of the project. A tubular IHX is feasible, but a single IHX is impractical, hence multiple loops will be required.

On this basis, it was concluded that a 3 loop configuration (3 IHXs for PCS + 1 IHX for H2 process loop) was the best option because it represented maximum feasibility and much reduced schedule risk. It also incurred other benefits as well as in gaining operational flexibility, and more maintainability. Furthermore, it is fully compatible with two loop commercial H2 plant and the increased plant cost compensated by reduced risk

20.3.4 Establish Secondary Temperatures

What is the secondary side T_{hot} and T_{cold}?

Given that the reactor side temperatures are set (T_{in} = 500ºC, T_{out} = 900ºC), the secondary side temperatures become set based on the selected approach temperature for the IHX design. The major impacts considered are those on PCS performance and NHS feasibility for the main loops and H2 process performance for the H2 loop. Of course, cost is universal concern for IHX design.

For the IHXs serving the PCS, a range of IHX approach temperatures between 25-50ºC was considered, however, it was concluded to remain with the current baseline of a 50ºC approach temperature on the basis that it results in good PCS performance, a reasonable effectiveness goal for large IHX (89%), and, as such, a large IHX necessitates controlling cost.

For the IHX serving the H2 process loop, the design challenge is somewhat lessened because of its duty size (60 MWth) which is significantly less than that of a main loop IHX. Given that the

H2 process performance needs higher temperature, a more aggressive approach temperature of 25C was selected, resulting in an effectiveness goal that is more reasonable for smaller IHX (94%). Furthermore, compact IHX technology was selected for the hydrogen plant due to it being more feasible at the level of service.

20.3.5 What are the Primary and Secondary System Pressures?

Primary pressures from 4.0-8.0 MPa were considered with emphasis on 5.0 MPa, 5.5 MPa, 6.0 MPa, and 6.5 MPa. The major considerations are operating cost and plant cost. Other discriminators are NHS feasibility and secondary system performance.

Based upon the evaluation, a primary system pressure of 5.0 MPa was selected since it offered good balance between vessel cost and pumping power. The pressure of the secondary system nominally balanced with the primary to minimize stress in IHX's with a slight bias (~+0.1 MPa) to optimize IHX design and operation.

20.3.6 Overall Results

Summarized below are the results that answer the main questions posed at the beginning of this section:

Reactor outlet temperature 900°C

Reactor inlet temperature 500°C

System configuration Parallel heat supply to electricity generating system (PCS) and hydrogen plant

3 loops with tubular IHXs for PCS

1 loop with compact IHX for hydrogen plant

Secondary temperatures	450-850°C for PCS
	475-875°C for hydrogen plant heat transport loop
Primary system pressure	5.0 MPa primary circuit
	Secondary balanced with primary (bias of 0.1 MPa)

20.4 Power Conversion System Trade Study

The Power Conversion System trade study is an independent study performed within the frame of the DOE-AREVA contract. Furthermore, the results of the study do not influence the selection of the PCS for the NGNP preconceptual design because the PCS concept is fixed by the DOE-AREVA contract specifying the adaptation of the ANTARES design.

The PCS study examines two closely related questions; namely:

- What type of PCS should be used?
  - Brayton cycle
  - Rankine cycle
  - Combined cycle gas turbine
  - Supercritical CO_2 (SCCO2)
  - Cascaded Supercritical CO_2 (SCCO2)
- How should it be coupled to the reactor?
  - Direct
  - Indirect

Considerations driving the selection are system performance; flexibility and operability; adaptability of existing technology; technology maturity; deployment schedule, system costs including development, capital, and operation and maintenance; reliability; availability; and maintainability.

Further, the relationship between the NGNP and a commercial plant must be considered. The NGNP must serve both electricity (PCS) and the hydrogen plant. The NGNP conditions are driven largely by hydrogen process. However, the commercial electricity plant would likely have different conditions. Also, the optimum PCS for the commercial plant may not be the same as the optimum PCS for the NGNP.

20.4.1 Approach

The general approach taken for the study was to develop heat balances for each of the candidate power conversion systems using either of two software packages: ChemCad for gas cycles and GateCycle for steam cycles. To validate the software, results for the different cycles were compared satisfactorily to results published by MIT [30]

Each cycle's heat balance was developed based on the following common assumptions:

- 565 MWt helium gas cooled reactor power

- 900 °C reactor outlet temperature
- 500 °C reactor inlet temperature – desired
- 55 kPa pressure drop across core
- 55 kPa pressure drop across intermediate heat exchanger (IHX) – when used
- 5 MPa reactor inlet pressure – desired
- 1% heat loss
- 98% Generator Efficiency
- 1% BOP loads

Based on the cycle's heat balance results, components were sized components and costs developed. Also, the technological maturity of each cycle was assessed based on a technological readiness level scale similar to that used in the aerospace industry. Additionally, issues such as integration, materials, safety, flexibility, etc; were considered.

20.4.2 PCS Cycle Results Comparison

Cycle thermodynamic data, component sizing, cost data and other parameters for each of the evaluated power conversion system cycles are presented for comparison in Table 20-3 after which commentary is provided.

Table 20-3: Comparison of Power Conversion System Cycles.

Parameter	Cycle						
	Direct Brayton	Indirect Brayton	Supercritical CO2	Cascaded Supercritical CO2	CCGT	Subcritical Steam	Supercritical Steam
Hot Temperature (°C)	900	850	642	850	850	565.6	601.7
Cold Temperature (°C)	25	25	32	32	32	32	32
Net Cycle Power (MWe)	268.9	251.6	267.8	281.6	270.9	241.7	264.8
Net Cycle Efficiency (%)	47.6%	44.5%	47.4%	49.8%	47.6%	42.8%	46.9%
Sizes							
Required Heat Transfer Area (m^2)	54,921	63,988	60,976	68,073	49,336	17,957	18,704
Heat Exchangers (m^3)	80	220	144	184	796	285	297
Turbomachinery (m^3)	564	580	5	7	916	676	710
Total Volume (m^3)	645	801	149	191	1,712	961	1,007
Heat Exchangers Floor Space (m^2)	143	160	137	143	108	63	67
Turbomachinery Floor Space (m^2)	131	134	4	9	170	107	111
Total Required Floor Space (m^2)	273	294	141	152	277	170	177
Costs							
Heat Exchanger Costs (Rel.)	21.4	27.1	30.7	29.6	45.9	28.7	29.8
Turbomachinery Costs (Rel.)	57.3	58.9	0.5	0.7	51.7	25.3	26.8
Building Costs (Rel.)	2.3	2.5	1.2	1.3	2.4	1.4	1.5
Total Costs (Rel.)	81.0	88.5	32.4	31.7	100.0	55.4	58.1
O&M	-	+	+	+	0	0	0
TRL (Development Costs)	4	4	3	3	6	9	8

Note: Costs comparisons are relative to CCGT cost assignment (100).

Power/Efficiency

cascaded supercritical CO_2 cycle provided the most power and the greatest thermodynamic efficiency followed closely by the CCGT, the direct Brayton cycle, the supercritical CO_2 cycle, and the supercritical steam-Rankine cycle, the indirect Brayton cycle and the subcritical steam cycle provide the lowest power output and thermodynamic efficiency.

Heat Exchangers

The trend in heat exchanger sizes is that the Brayton and supercritical CO_2 cycles require the greatest amount of heat transfer surface area. This primarily due to the high degree of recouperation required for these cycles to operate efficiently. The direct Brayton cycle area required is lower because an IHX is not required. The CCGT heat transfer surface area requirement is driven mostly by the IHX and the condenser in the steam bottoming cycle.

Both the subcritical and the supercritical steam-Rankine cycles have the lowest heat transfer surface area requirements. This is due to the large driving temperature differential between the primary at 900 °C and the steam at 565 and 602 °C, respectively. Most of the required heat transfer area for these two cycles occurs in the condenser and the feedwater heaters.

Turbomachinery

The trend in turbomachinery sizes is that the supercritical CO_2 cycle and the cascaded supercritical CO_2 cycle turbomachinery is very compact. This due to the nature of the supercritical CO_2 cycle operating near the critical point where the fluid is very dense compared to an ideal gas. The Brayton cycle turbomachinery size is moderate, and the steam turbines in the CCGT bottoming cycle and the two steam-Rankine cycles are the largest due to the large expansion ratios. Further, the CCGT also has the gas-turbine size included.

Cost

Capital cost trends are quite interesting. The supercritical CO_2 cycles tend to be the lowest due to their compact turbomachinery; costs for these cycles are driven by heat transfer equipment. The costs of the Brayton cycles is large and is driven by both turbomachinery and heat transfer equipment. The costs of the steam-Rankine cycles are less than the Brayton cycles and are driven by the costs of the steam turbine and the condenser. The costs of the CCGT are the greatest due to the costs of a large IHX, a gas-turbine, plus a steam turbine and condenser.

Operations and Maintenance

Operation and maintenance has not been addressed until now. A simple rating system of +, 0, - was used; + being the highest and - the lowest. Because the direct Brayton cycle is coupled directly to the core, where the potential for radioactive contamination to spread through the PCS, and get into inaccessible areas such as bearings, seals, narrow cooling passages, is high, which may require special procedures and equipment to perform maintenance, this system was rated as -. The CCGT and both steam-Rankine cycles were rated as 0 for their corrosion and water chemistry control issues. The indirect Brayton and both the supercritical CO_2 cycles were rated as + because they do not have the radioactive contamination issue and corrosion is judged to be less significant.

Technology Readiness Level

For cycle maturity, the steam-Rankine cycles are the most mature; the subcritical steam-Rankine cycle was rated at level 9, and the supercritical steam-Rankine was rated at level 8. The CCGT was rated at a moderate maturity level of 6, while the Brayton cycles, rated at level 4, are somewhat less mature. The supercritical CO_2 cycles are the least mature and were rated at level 3.

20.4.3 Power Conversion System Evaluation

A simple grading system was used to compare the raw values of key parameters presented in Table 20-4 in a concise but clear way. A **+**, **0**, **-**, **-** grading system was used with **+** being very good or favorable, **0** being acceptable, **-** being marginal or a moderate concern, and **-** meaning that this is a major concern or disadvantage.

The parameters used for the evaluation were judged to be the most salient for evaluating a PCS option. The evaluation results are presented graphically below in Table 20-4 which is followed by commentary.

Table 20-4: Power Conversion System Evaluation.

Parameter	Cycle						
	Direct Brayton	Indirect Brayton	Supercritical CO2	Cascaded Supercritical CO2	CCGT	Subcritical Steam	Supercritical Steam
Cycle Power / Efficiency	0	–	0	+	0	–	0
Total Costs (Rel.)	0	0	+	+	–	0	0
O&M	–	+	+	+	0	0	0
TRL (Development Costs)	–	–	–	–	0	+	+
Schedule	–	–	–	–	0	+	+

Cycle Power/Efficiency

The cascaded supercritical CO_2 cycle delivered the most power and was given the highest ranking. The CCGT, supercritical CO_2 cycle, direct Brayton cycle, and supercritical steam-Rankine cycle gave comparable, but lower than the cascaded supercritical CO_2 cycle, powers and were given a good rating. The other two cycles, the indirect Brayton and the subcritical steam-Rankine, produced the least amount of power and were given a moderate rating.

Total Costs

The two supercritical CO_2 cycles had the lowest costs and the highest rating. The two Brayton cycles and the two steam-Rankine cycles were rated as good, and the CCGT was rated as moderate.

Operations and Maintenance

The indirect Brayton and the two supercritical CO_2 cycles were given the highest rating. The CCGT and the two steam-Rankine cycles were rated lower because of corrosion and water chemistry control issues, and the direct Brayton cycle was given a moderate rating because of the radioactive contamination issue.

Technological Readiness Level

For system maturity, the two steam-Rankine cycles were rated the highest. The CCGT was rated as acceptable due to its moderate maturity, and the other cycles were rated as marginal due to their low maturity.

Schedule

In addition to the above categories, each PCS cycle was assessed whether it would be sufficiently mature to meet a start-up date of 2018. The two steam-Rankine cycles were rated as favorable due to their maturity. The two Brayton cycles and the two supercritical CO_2 cycles were rated as marginal due to their lack of sufficient maturity to provide confidence that they would be available in time. The CCGT was rated as acceptable even though there are risk issues which need to be addressed such as closed loop operation, high compressor inlet temperature, and materials issues

20.4.4 Commercial Plant Applicability Evaluation

In addition to the PCS ratings discussed above, the various PCS cycles were assessed for their impact upon commercial plant considerations. Specifically, three categories were examined: reactor impact which examines how reactor materials, schedule, fuel cycle length, and power level are affected by selection of a given PCS cycle; process heat flexibility which examines the adaptability of the PCS to provide process heat to a broad, diverse, and most likely near term markets for medium and high temperature steam (i.e., < 550 °C); and very high-temperature process heat (VHTPH) which examines the ability to service the thermo-chemical hydrogen production processes being considered. The results of this evaluation are shown in Table 20-5.

Table 20-5: Commercial Plant Applicability Evaluation.

Parameter	Cycle: Direct Brayton	Indirect Brayton	Supercritical CO2	Cascaded Supercritical CO2	CCGT	Subcritical Steam	Supercritical Steam
Reactor Impact	0	–	+	0	0	+	+
Process Heat (PH) Flexability	–	–	?	?	+	+	+
VHTPH	–	+	?	?	+	–	–

Reactor Impact

The supercritical CO_2 cycle and the two steam-Rankine cycles had the most positive impact on the reactor. This is because these cycles operate at lower turbine inlet temperatures implying that the reactor temperatures can be lowered, thus permitting higher reactor power, increased fuel cycle length, and a positive impact on reactor material selection and schedule.

The CCGT, direct Brayton cycle, and the cascaded supercritical cycle had a smaller positive impact on the reactor because these cycles could be operated at 850 °C reactor outlet temperature.

The indirect Brayton cycle did not have a favorable impact on the reactor because reactor outlet temperature could not feasibly be brought lower than 900 °C without having a major impact on performance, and the high reactor inlet temperature of 550 °C significantly affects vessel cooling.

Process Heat Flexibility

For process heat flexibility, the two steam-Rankine cycles and the CCGT were given the highest rating, because the ability to extract steam at the desired conditions from a steam cycle is well established. The two Brayton cycles were given negative ratings because the ability to provide steam for process heat is impractical. The two supercritical CO_2 cycles have not been rated because the cycle has not been extensively studied to determine the practicality of providing process heat.

Very High-Temperature Process Heat (VHTPH)

For applicability of providing VHTPH, the indirect Brayton and the CCGT were given the highest ratings because of the high reactor outlet temperature coupled with the use of an IHX which can incorporate a VHT loop for thermo-chemical processes. The two steam-Rankine cycles were given poor ratings because electric supplemental heating would most likely be needed to achieve the desired high temperatures. The direct Brayton cycle was given a negative rating due to the difficulty of incorporating a parallel VHT loop. The two supercritical CO_2 cycles were not rated for the same reasons given for not rating them for process heat flexibility.

20.4.5 Conclusions

The best PCS cycle for the NGNP is dependent upon the worth of cycle efficiency and the importance of cycle maturity, especially as it relates to achieving NGNP startup in 2018:

1. The supercritical CO_2 cycles are very promising for longer term applications. The need for development is a disadvantage for near term applications. The ability to arrange them in a cascaded configuration for large ΔT applications is a plus.
2. Steam-Rankine cycles are the most mature, but the cost of steam turbines and supporting equipment reduces their attractiveness. The supercritical steam cycle with two reheats is the best steam cycle option.

3. The Brayton cycles are marginal in cost and performance. Operation and maintenance difficulties from radioactive contamination of the PCS are a negative for the direct Brayton cycle. On the other hand, the loss of efficiency as a result of the temperature drop across the IHX reduces the attractiveness of the indirect Brayton cycle. Further, because of the relative unattractiveness of the Brayton cycles when compared to the supercritical CO_2 cycles brings further pursuit of Brayton cycle development into question.
4. The CCGT performance is good but the costs, added complexity and lower maturity when compared with steam-Rankine cycles reduces its attractiveness. The potential for long term economic advantage from small efficiency differences when compared to the supercritical steam-Rankine cycle or the indirect Brayton cycle may swing the advantage to the CCGT.

20.4.6 Recommendation for NGNP

Steam-Rankine cycle (possibly supercritical) is clearly the best fit for a near term applications such as the NGNP. It provides high efficiency electricity production and can readily service near term process heat markets. Moreover, it is a familiar technology that is directly coupled to reactor system.

The promising benefits of the supercritical CO_2 cycle warrant continuing development for long term electricity production applications. Further, a more detailed evaluation of equipment costs and size would be beneficial for confirmation of these recommendations.

More detailed evaluation of equipment cost & size would be beneficial for confirmation of recommendations.

21.0 NGNP FUTURE STUDIES

In the course of the NGNP Preconceptual Design Engineering Studies, the AREVA NGNP team has identified several additional studies that would be very beneficial. These studies include both new work completely outside the current NGNP preconceptual design effort as well as the extension of important existing work. These future studies should be given high priority early in the next phase of the NGNP program, whether that is at the initiation of Conceptual Design or an extension of the current Preconceptual Design Studies phase.

The future studies identified in this section vary significantly in both magnitude and importance. Some are fundamental to the basic concept definition of the NGNP. It is important that those studies be resolved at the very beginning of conceptual design phase. Others are precursors or complementary to the normal engineering design activities of Conceptual Design and subsequent phases. Those studies should be addressed during the Conceptual Design phase. The remaining future studies would likely not directly impact NGNP, but they could have significant potential benefit for future plants beyond the NGNP.

The critical studies which have the potential to modify the basic configuration of the NGNP and which should therefore be resolved prior to the start of major conceptual design work include the following:

- Steam Cycle Concept Evaluation
- Water Ingress White Paper
- Demonstration Plant Size Confirmation
- H2 Process Selection
- In-Depth Analysis of MTE H2 Process

Special attention is directed to the recommended studies within the proposed Steam Cycle Concept Evaluation. This evaluation is fundamental, since it has the potential to completely redefine the NGNP concept. AREVA believes very strongly that the steam cycle concept should be considered as a probable replacement for the current very high temperature NGNP approach. The steam cycle concept is judged to be the most likely concept capable of leading to large scale commercial deployment in the foreseeable future. It requires less R&D and it has less technical and schedule risk compared to other concepts. Finally, it has much greater market flexibility.

AREVA strongly recommends that a thorough evaluation of the steam cycle concept’s capability to fulfill the NGNP mission objectives be performed prior to the start of full conceptual design.

Each of the recommended future studies is discussed briefly in the next section.

21.1 Discussion of Future Studies

The suggested future studies are listed in Table 21-1. Individual studies are discussed in the subsections below.

21.1.1 Steam Cycle Concept Evaluation

A steam cycle HTR concept provides an attractive alternative to other concepts such as the direct or indirect Brayton cycle or the combined cycle gas turbine (CCGT). In the steam cycle concept, the reactor outlet helium supplies heat directly to a steam generator which produces high temperature, high pressure steam. Modern steam systems can achieve electricity generating efficiencies comparable to those of Brayton cycles (40-48%) but with less demanding reactor operating conditions.

Perhaps more importantly, high temperature steam is directly applicable to many process heat markets, and it is relatively easy to deliver. High temperature steam (~550°C) can be transported over longer distances compared to very high temperature gas. This temperature range encompasses the majority of potential process heat markets. For very high temperature applications, a variety of means exist to augment the steam heat with supplemental electrical heating. While this imposes the inefficiency of the electricity generating process on part of the delivered energy, it minimizes the pumping and heat losses associated with very high temperature heat delivery. It also provides greater flexibility in the design of the chemical plant or other process heat facility. For example, scoping analyses of high temperature hydrogen production processes supplied with steam and electricity suggest that the actual impact on overall performance is small compared to a facility supplied directly with very high temperature heat.

A major advantage of steam cycle systems is that they can be ready to meet these energy needs in a shorter time frame with less R&D and less technical risk than higher temperature systems. Steam cycle systems require reactor outlet temperature on the order of 750°C compared to a reactor outlet temperature in the range of 850-950°C for high performance Brayton systems or very high temperature heat delivery systems. They also avoid the need for very high temperature heat exchangers or for advanced helium turbomachinery. The resulting reduction in development reduces the project schedule risk to a reasonable level.

In summary, the major potential advantages of the steam cycle concept include:

- Greater reliance on existing technology
- Reduced risk (technical, schedule, and cost)
- Shorter development and deployment schedule
- Less demanding performance for reactor and associated systems
- Comparable electricity generating efficiency to more advanced concepts
- Broad applicability to a variety of process heat markets
- Greater commercialization potential (market flexibility and earlier deployment)

AREVA strongly recommends that the steam cycle concept be strongly considered as an alternative to the higher risk reference NGNP approach. The following four activities are proposed in order to better define the characteristics of a steam cycle NGNP configuration and to characterize its relative advantage compared to the reference concept.

- Develop alternate NGNP Preconceptual Design employing steam cycle instead of high temperature IHX or direct Brayton.
- Evaluate adaptability of a separate high temperature test loop to the basic steam cycle configuration (in parallel with the steam generator in order to allow testing of high temperature components).
- Define required R&D for the steam cycle concept relative to the R&D required for the reference (~900°C) NGNP concept.
- Define project schedule for steam cycle concept relative to the reference NGNP concept.

21.1.2 Water Ingress White Paper

Any decision to adopt a steam cycle HTR configuration increases the significance of water ingress events due to the potential for steam generator leaks. This issue was successfully managed in previous operating HTRs. However, the possibility for water ingress continues to be perceived as a significant issue within the broader nuclear community. There are various reasons for this including misunderstanding of the source of water ingress in the Fort St. Vrain reactor, failure to appreciate the differences in steam generator technology between HTRs and LWRs, and unfamiliarity with the consequences and mitigation of water ingress in HTRs.

Steam line breaks within the reactor building also must be considered for steam cycle concepts. Steam line breaks must be evaluated for building pressurization issues and for any impact on building venting and filter systems, if a vented confinement concept is used for the NGNP.

A white paper should be developed addressing water ingress and steam line break events and their likely impact on NHS design. The intent is not necessarily to provide detailed analyses of such events. Rather the focus should be on describing the issues and concerns associated with each type of event, the potential significance of these events on operation, safety, and licensing, mitigation of these events including likely design features which might be utilized, and likely R&D that might be necessary to resolve any open issues.

The intent of the recommended white paper would be to provide an objective perspective for the assessment of this issue as part of the overall consideration of a steam cycle concept.

21.1.3 Air Ingress Assessment

Air ingress events are a potential issue for all graphite moderated HTRs, due to the concerns associated with graphite oxidation. This issue is similar to the water ingress scenarios in that, while there is a credible technical issue which must be addressed in the course of the reactor design and safety analysis, there is also a large perception issue that is somewhat independent of the technical issues.

An objective characterization of air ingress events is recommended in order to put these events in the proper context. The recommended assessment of air ingress events should include scenario definition, controlling phenomenon, potential consequences, and mitigation strategies. The objective is to provide a reasonable framework for the discussion and quantitative evaluation of these events.

21.1.4 Demonstration Plant Size Confirmation

The NGNP Power Level Special Study concluded that the NGNP should be built at the full commercial size in order to maximize the benefit of the project in support of subsequent HTR commercialization. Building the NGNP at full size minimizes the technical risk, design cost, and licensing risk and effort for the future commercial plant.

However, while building the NGNP at full size minimizes these risk elements, it does not completely eliminate them. The actual level of minimization of risk and residual FOAK design cost for the first commercial plant that the NGNP project will provide is dependent on the similarity between the commercial plant and the NGNP prototype.

Moreover, placing the entire risk burden on the NGNP FOAK plant with the resulting impact on design and construction costs may actually maximize the risk of funding discontinuity for the NGNP, if the total project cost is a significant determinant of this risk. This consideration was not part of the existing plant power level study.

Since the completion of the Prototype Power Level Study, it has been recognized that previous assumptions regarding the relationships between prototype capital cost, design cost, plant size, and NGNP and commercial plant design differences should be refined. This is particularly true in light of the recommendation to refocus future efforts on the steam cycle concept.

Therefore, AREVA suggests a further effort with the two objectives

1. Confirm the conclusion of the initial power level study and

2. Explore approaches to minimize the residual FOAK design costs and risks that might affect the first subsequent commercial plant.

This follow-on study should take into account FOAK vs NOAK development costs, fabrication costs, and risks, and the partitioning of these between the initial demonstration plant and subsequent first commercial plant. Reevaluate major breakpoints based on component transportability, capital cost (overall, number of loops, etc.), technology demonstration, etc.

It is anticipated that the conclusions of the power level study will not change substantially. Nonetheless, it would be prudent to reconfirm the study results, and it is important to identify clear strategies to minimize the residual risk and FOAK development cost in the first commercial plant.

21.1.5 Evaluate Elimination of SCS

In simple terms, the mission of the Shutdown Cooling System is to provide redundancy for decay heat removal and to provide the capability for timely recovery from conduction cooldown events (which necessitates the ability to accept high temperature primary coolant). However, given that the reference NGNP concept has multiple parallel cooling loops each employing a high temperature heat exchanger, the SCS may not be essential.

Therefore, a detailed evaluation is recommended to determine whether the Shutdown Cooling System is really necessary for an NGNP concept with multiple independent loops and high temperature heat exchangers.

21.1.6 Maximum Power Level with SA508 Reactor Pressure Vessel

PWR material (e.g., SA508) may provide an alternative reactor vessel material instead of modified 9Cr-1Mo. However, the limitations of SA508 would likely impact the design of the reference NGNP configuration. The lower temperature capability of SA508 poses design challenges for both normal operation and accident conditions.

The normal operation challenge stems mainly from the high reactor inlet temperature. Design approaches to deal with the higher inlet temperature in combination with the lower temperature material would significantly complicate the design of the reactor internals and core flow design, but they are thought to be achievable.

The accident condition challenge is associated with the sustained high temperatures during conduction cooldown when decay heat must be radiated from the reactor vessel to the RCCS. Mitigation of this concern is more problematic, since it generally requires a reduction in reactor power level to control the decay heat load with an associated impact on plant economics.

Therefore, a detailed study is proposed to determine the maximum power level achievable with an SA508 reactor vessel. Since the achievable power level is influenced by the initial temperature distribution in the reactor at the beginning of the accident, two cases should be considered. One will focus on the reference (~900°C) NGNP concept. The other will determine the maximum power level achievable with SA508 for the alternative steam cycle NGNP concept.

21.1.7 Confirm Selection of 9Cr-1Mo RPV Material

Modified 9Cr-1Mo steel provides significant performance advantages for the reactor pressure vessel material including high temperature capability and improved irradiation resistance compared to SA508. However, 9Cr-1Mo is not an established reactor vessel material, and its use will require development in terms of procurement, fabrication, qualification, and code acceptance.

Therefore, a more detailed study should be planned and implemented to amplify, refine, and elaborate the factors in the assessment and selection of 9Cr-1Mo steel for the primary pressure vessels (e.g., forging, fabrication, procurement, codification). This study must distinguish perception from reality regarding the fabrication difficulties associated with 9Cr-1Mo. Attention must be given to the relative schedule risks associated with 9Cr-1Mo compared to SA508 for HTR applications against the relative associated performance advantages.

21.1.8 High Temperature Testing Mode Evaluation

The potential capability has been identified to perform temporary testing in the dedicated loop with reactor outlet high temperatures significantly higher than normal with corresponding reduction in reactor power. A detailed evaluation of this capability should be performed to assess its viability and to determine the potential range of test conditions that might be achieved. Considerations of such a study should include

- Consideration of low power stability for reactor and heat removal systems.
- Identify options for integrating high temperature tests in dedicated side loop and identify candidate equipment and systems to be tested (e.g., high temperature IHX, Brayton cycle, supercritical CO_2, etc.).
- Design provisions in the nuclear island, particularly for the dedicated test loop, required to support this testing should be identified as part of this study.

21.1.9 Requirements for Irradiated Component Testing and Examination Facilities

One mission of the NGNP is the testing of advanced materials, components, and systems under operational conditions. During the preconceptual design phase, significant emphasis was not placed on defining the requirements for facilities necessary to support future testing activities including post-irradiation examination facilities, etc.

A dedicated effort is recommended to define the specific governing requirements for these facilities.

21.1.10 Ceramic IHX Technology Evaluation

Ceramic heat exchanger technology is only at a very immature state of development, particularly for large scale process fluid applications. Nonetheless, ceramic heat exchangers are reasonably viewed as the ultimate solution to the very high temperature heat delivery challenge, if the formidable development challenges can be met.

Therefore, a detailed assessment of ceramic IHX technologies should be performed. Specific points to be addressed include:

- Current alternative IHX concepts
- Current capability of ceramic heat exchangers
- Required R&D to develop large scale ceramic IHX module
- Anticipated development timeline to achieve industrial scale ceramic IHX module

21.1.11 High Temperature Heat Transport Fluid Comparison

The NGNP preconceptual design special study on the high temperature heat transport loop was not included in AREVA's work scope. Without access to the study being completed by the assigned NGNP team, it is somewhat speculative to discuss the selection of heat transport fluid to be used in this loop. Nonetheless it is an important area both for the design of the NGNP and for the development of the technology needed for subsequent commercial plants. Therefore, AREVA tentatively recommends that a follow-on study be performed to evaluate the different heat transport fluids that might be used in such loops.

This study should consider the full range of candidate fluids including not only the currently favored options of high pressure helium or molten salts but also examining other fluids including high temperature steam, commercial high temperature heat transport fluids on the market, liquid metal, and other options that may be identified. The comparison needs to consider heat transport efficiency, heat exchanger performance, material compatibility, toxicity and safety, impurity control, resulting system design difficulty, etc.

21.1.12 IHX Technology Recommendation for Test Loop

For the reference AREVA NGNP design developed for this project, a compact IHX is recommended for the side test loop which will supply up to 60 MWth to the hydrogen production plant. This recommendation is based on the fact that for the small IHX, the environment and heat transfer conditions are more favorable with helium as the secondary coolant, as well as the premise that a more conservative (and costly) design can be employed given the small size. Several compact heat exchanger technologies have been considered for HTR IHX applications (e.g., plate-fin, PMHE, etc.). A study is needed to recommend the best compact heat exchanger technology for the IHX serving the hydrogen production plant.

21.1.13 Hydrogen Process Selection

The NGNP preconceptual design studies did not include an explicit study to select the preferred hydrogen production process for the NGNP. While the relative immaturity of most hydrogen processes make it impossible to make a final selection of a single hydrogen process, it is important in designing the NHS to focus on the interface requirements of the most likely process. Therefore, a detailed study should be performed to obtain the latest recommendation for this key decision. Understanding that numerous similar studies have been performed in the past, it is still recommended that an updated study be performed taking into account the latest information on the candidate hydrogen processes as well as new information on the potential limitations on the NHS side. At a minimum, the evaluation should consider the top five hydrogen production plants (HTE, S-I, Medium temperature Electrolysis, nuclear assisted SMR, alternate thermochemical). Factors to consider include:

- Technology readiness of hydrogen process (system and individual components)
- Impact on NHS technology requirements
- Anticipated performance
- Economics (capital cost, development cost, NOAK cost, product cost)
- Deployment strategy and schedule
- CO_2 avoidance (near-term impact and long-term impact)

21.1.14 In-Depth Analysis of Medium Temperature Electrolysis Hydrogen Process

A detailed process analysis and economics evaluation should be performed for moderate temperature (~400-650°C) steam electrolysis encompassing innovations in materials of construction, fabrication, and manufacturing of high temperature membrane systems.

21.1.15 Refueling Strategy Selection

A detailed comparison and selection of the NGNP refueling strategy and equipment should be performed. This study would use time and motion studies to evaluate different refueling equipment designs, including the "industry default" cask system and AREVA's fuel server system, as well as various reflector element management scenarios.

21.1.16 Graphite Waste Disposition Strategy Evaluation

A comparison and selection of graphite waste disposition strategies should be performed, including assessment of graphite radionuclide inventory and prioritization of work on future disposition approaches.

21.1.17 Graphite Qualification and Acquisition Strategy

A graphite qualification and acquisition strategy similar to the current fuel program is needed and should be developed.

21.1.18 Review Existing Regulation Applicability for HTRs

In preparation for the anticipated NGNP formal licensing activity, a detailed review of existing licensing regulations should be initiated including the identification of those that apply, those that do not apply, and additional regulations that may be needed.

21.1.19 Develop NGNP Licensing Strategy Topical Report

A detailed NGNP licensing strategy topical report should be prepared for submittal and review by the NRC. This report should provide a roadmap for both licensing of the NGNP prototype plant and design certification of one or more subsequent commercial plant designs, including the strong interrelationships between these two activities. The need for supporting technology must be addressed including the required timing of this data. Direct vendor involvement in this activity is essential if the resulting plan is to support the projects commercialization objective.

21.1.20 Evaluate Role of Design Stage PRA

An assessment and evaluation should be performed of design stage PRA and its value in risk-informed design and as a licensing tool.

21.1.21 Strategy to Use Modular Construction Techniques

Assessment of modular construction techniques and their application to NGNP and the commercial plant should be performed early in the design process. Appropriate design guidelines should be developed for relevant systems.

21.1.22 Evaluation of C-C Composites

During the development of the detailed design of various core components, and other very high temperature equipment, it is anticipated that the use of ceramic composite materials may be required. To support their use, a performance evaluation of Carbon-Carbon composite materials in HTR core environment should be conducted, including assessment of need for SiC-SiC composite development.

21.1.23 Assessment of HTR Computer Code Suite

In parallel with the development of the NGNP plant design, development of a suite of computer codes to support plant licensing and performance evaluation must be accomplished. As a first step, an assessment and comparison of existing thermal-hydraulic, neutronic, fuel performance, and radionuclide transport code suites should be completed.

21.1.24 Evaluation of Confinement/Containment Strategy

Due to the small size and inherent safety characteristics of modular HTGRs, it has been proposed that such reactors, including the NGNP and subsequent commercial VHTRs, be built without traditional pressure retaining containment structures such as those required for LWRs. Moreover, in most of the modern HTGR development programs, it has been assumed that elimination of an LWR-style containment system is desirable for economic deployment.

The containment decision obviously has significant effects on licensing, plant siting flexibility, safety strategy, required fuel performance, external hazard mitigation, and plant economics. Therefore, a detailed study is proposed that examines each of these issues, considering their potential impact on commercial HTR deployment as well as the immediate impact on NGNP design and licensing. This study should include the various options ranging from unfiltered confinement structures and filtered confinements all the way to unvented structures with either short-term or long-term pressure retaining capability. The objective of the study would be to recommend the best design option based on the considerations identified.

21.1.25 Reactor Embedment Evaluation

The current NGNP plant preconceptual design utilizes a fully embedded reactor building to address concerns ranging from reactor protection from external threats to management of direct offsite doses. The cost of such a strategy is significant. A comparative evaluation of the costs and benefits of this design should be conducted.

21.1.26 INL Site Heavy Component Transportation Issues

The proposed location of the NGNP on the INL site will require over-land transportation of all heavy components, either as complete units or in pieces to be assembled on-site. Transportation issues associated with the INL site, and the NGNP reactor, should be evaluated, including the range of vessel and heavy component sizes required for construction, the basic feasibility of various transportation options, their relative costs, and schedule considerations (time to prepare route, get permits, actual transport duration, etc.).

21.1.27 Main Component Fabrication Strategy

In parallel with the INL site heavy component transportation issues study, a fabrication strategy for main components should be developed. This study should include identification of potential suppliers, assessments of on-site versus off-site fabrication issues, and comparison of relative costs.

21.1.28 ISI Strategy and Design Impact

A preliminary In-Service Inspection (ISI) strategy should be developed for the NGNP, including consideration of the Reactor Vessel, other primary vessels, the reactor internals, and other key components. The impact of the developed ISI requirements on Nuclear Heat Source design should be identified.

21.1.29 Protection of Ni-Based Alloys in HTR Environment

A protection strategy for high temperature Ni base alloy components in HTGR environment should be developed. A detailed R&D program to address unresolved issues should be compiled to support the strategy.

21.1.30 Advanced Fuel Cycle Strategies

The preconceptual design of the NGNP considers the use of a once-through, uranium-based fuel cycle. One of the main advantages of the HTGR technology is its adaptability to other fuel cycles. A study of various other fuel strategies, including use of alternate fuel types, deep burn strategies, and potential synergies with other reactor types should be conducted to demonstrate the HTGR's abilities to improve resource utilization and minimize waste quantities and longevity.

21.1.31 NGNP Preconceptual Design Lessons Learned

A review of the AREVA NGNP preconceptual design activities should be performed to identify any lessons learned which might make future preconceptual or conceptual design activities more efficient, would help avoid any difficulties encountered in the current preconceptual design work, or might enhance the quality of results achieved.

Table 21-1: Suggested NGNP Future Studies

Study	Applicability		Purpose		
	Steam	Indirect	Concept Definition	Design Dev.	Beyond NGNP
Steam Cycle Concept Evaluation			X		
Water Ingress White Paper	X		X	X	
Air Ingress Assessment	X	X		X	
Demonstration Plant Size Confirmation	X	X	X	X	
Evaluate Elimination of SCS		X		X	
Maximum Power with SA508 RPV	X	X	X		
9Cr-1Mo RPV Material Confirmation		X		X	
High Temperature Testing Mode Evaluation	X	X		X	
Requirements for Irradiated Component Testing and Examination Facilities	X	X		X	
Ceramic IHX Technology Evaluation		X			X
H.T. Heat Transport Fluid Comparison		X			X
IHX Technology Recommendation for Test Loop		X		X	
H2 Process Selection	X	X	X	X	X
In-Depth Analysis of MTE H2 Process	X		X	X	X
Refueling Strategy Selection	X	X		X	
Graphite Waste Disposition Strategy Evaluation	X	X		X	
Graphite Qualification and Acquisition Strategy	X	X		X	
Review Regulation Applicability for HTRs	X	X		X	
NGNP Licensing Strategy Topical	X	X		X	
Evaluate Role of Design Stage PRA	X	X		X	
Strategy to Use Modular Construction Techniques	X	X		X	
Evaluation of C-C Composites		X		X	
Assessment of HTR Computer Code Suites	X	X		X	
Evaluation of Confinement/Containment Strategy	X	X		X	
Reactor Embedment Evaluation	X	X		X	
INL Site Heavy Component Transportation Issues	X	X		X	
Main Component Fabrication Strategy	X	X		X	
ISI Strategy and Design Impact	X	X		X	
Protection of Ni-based Alloy in HTR Environment		X		X	
Advanced Fuel Cycle Strategies (e.g., improved utilization and waste destruction)	X	X			X
NGNP Preconceptual Design Lessons Learned	X	X		X	

22.0 SUMMARY OF CONCLUSIONS

This section captures the main conclusions of AREVA's NGNP preconceptual design studies.

22.1 Special Study Conclusions

The AREVA NGNP team was asked to perform four of the special studies identified in the original INL/BEA Scope of Work.

Reactor Type Comparison Study – The AREVA study concluded that the prismatic block HTR was the preferred option in order to maximize the unit power level for best economic performance.

Prototype Power Level Study – The AREVA study tentatively concluded that the NGNP prototype should be full size (565 MWth) in order to maximize the benefit to future commercial plants. This study also concluded that 60 MWth should be provided to the test loop supporting the hydrogen process demonstration facility.

Power Conversion System Study – The AREVA study concluded that a steam cycle system was preferred for near-term deployment. The study also concluded that supercritical CO_2 systems may have significant potential advantages and should continue to be pursued for long-term deployment.

Primary and Secondary Cycle Concept Study – The AREVA study concluded that the reference NGNP should have reactor inlet/outlet temperatures of 500°C /900°C as the best compromise between hydrogen production performance and NHS feasibility. The study also concluded that a multiple loop primary circuit configuration using robust tubular IHXs for the main energy transfer to the PCS should be used to maximize feasibility and minimize risk.

22.2 Design Adaptation Conclusions

The AREVA NGNP preconceptual design work scope required that AREVA develop an NGNP design concept adapted directly from AREVA's existing ANTARES HTR concept employing an indirect cycle combined cycle gas turbine (CCGT) energy utilization system. AREVA's scope did not include either the hydrogen production facility or the high temperature heat transport loop.

The resulting reference NGNP design is a 565 MWth prismatic HTR with a modified 9Cr-1Mo reactor vessel. Heat is supplied to three parallel loops each with a helical coil tubular IHX and a dedicated primary circulator. Tubular IHXs are used to maximize design feasibility and component lifetime for very high temperature service.

The secondary coolant from the three IHXs is combined to drive a single closed loop gas turbine. The secondary coolant is a mixture of nitrogen and helium selected to allow the use of air-breathing gas turbine technology. Residual heat from the turbine outlet drives the Rankine bottoming cycle through a heat recovery steam generator.

A fourth primary loop is included to provide heat for demonstration of high temperature hydrogen processes. Given the smaller size of this loop and the use of helium as the secondary fluid in the heat transfer loop, a compact heat exchanger is specified for this loop in order to demonstrate that new technology.

The resulting configuration is the best configuration that can be achieved in the near-term for direct high temperature heat supply. It minimizes technical risk to the maximum extent possible.

However, AREVA does not actually recommend this configuration at this temperature unless direct high temperature heat supply is the sole objective regardless of the technical challenges. As concluded in AREVA's PCS study and further recommended in the future studies section of this report, a simple steam cycle concept is the preferred configuration based on increased market flexibility, minimized technical risk, and most rapid deployment schedule.

22.3 Hydrogen Production Conclusions

While the design of the hydrogen production plant was not part of the AREVA team's assigned scope, the AREVA team did agree to perform an evaluation of high temperature steam electrolysis. This evaluation focused on overall system performance assuming heat was supplied using extraction steam from an adjacent steam cycle HTR and electric energy. Over a range of electrolyzer operating temperature of 600°C to 800°C, the best performance was predicted at 600°C. This result is based on the overall system performance, not just the electrolyzer.

22.4 Risk and R&D Conclusions

The key risks identified for the NGNP project are

- Fuel performance (the probability of this risk can be reduced, but the potential consequence can not be minimized)
- Heavy component procurement and fabrication (industrial capacity is limited both in forging size and supply schedule in the current market)
- Nitriding of secondary circuit components (assuming a nitrogen-helium mixture is used for the PCS secondary coolant)
- Licensing (lack of design-specific licensing rules presents a schedule uncertainty and applying existing LWR rules presents cost uncertainty, e.g., regulatory demand for pressure retaining reactor building)
- Funding continuity (consistent effort is required for R&D, design, and procurement/construction in order to achieve the NGNP mission)

The key R&D needs identified for the NGNP project are

- Fuel manufacturing and qualification
- Graphite qualification
- Modified 9Cr-1Mo qualification and codification
- High temperature materials
- Methods qualification

The overall technical, schedule, and cost risks are judged to be manageable. However, the challenges are significant. Prompt aggressive action is required in key areas, if risks are to be adequately mitigated. Control of overall NGNP project risk requires prompt project execution, the development of a technology roadmap with off-ramps for key technology risks, a strategy to avoid funding and resource constraints, and alignment with commercial market needs.

22.5 Cost and Economic Conclusions

Costs were estimated for the First-of-a-Kind reference NGNP prototype, including R&D, design, capital, operation, and decommissioning.

An economic analysis was also performed to estimate the output product cost for the commercial Nth-of-a-Kind VHTR plant.

While the AREVA team did not assess the cost or economic performance of a comparable steam cycle NGNP alternative, it is clear that both the development cost and the capital cost of the steam cycle would be significantly lower than for the reference concept (probably 20-30%), while the reduction in electricity or hydrogen production efficiency would be relatively minor.

22.6 Future Studies Conclusions

A number of future studies have been identified as being important in determining the future direction of the NGNP project, as being necessary to support anticipated conceptual design activities, or as being beneficial to the long-term deployment of future HTRs.

Two key studies are important in determining the future direction of the program.

The first is a thorough evaluation of the steam cycle as an alternative path to fulfill the NGNP mission on the desired near-term schedule with greater probability of success and greater potential for near-term commercialization. The components of the recommended study would evaluate the relative technical risk and technology development requirements of the steam cycle concept compared to the alternative direct high temperature heat approach. The capability of supporting direct high temperature hydrogen production with a steam cycle system would also be examined. A key part of the study would be to develop an optimized preconceptual design of a steam cycle HTR plant.

The second study is related to confirmation of the recommended size of the NGNP prototype plant. A study is suggested to evaluate the potential differences between the NGNP prototype plant and possible future commercial HTR/VHTR plants and the impact that these differences might have on the partitioning of design cost, required R&D, and risk between the prototype and the first commercial plant. The likely impact of plant size on the risk of prototype project funding discontinuities would also be assessed. The objective of this study would be to first confirm the initial size recommendation and then to define NGNP design guidelines that will minimize the residual risk and development cost associated with the first subsequent commercial plant.

22.7 Overall Conclusion

AREVA believes that the HTR has the potential to make a major impact on the broader energy market, if the necessary technology hurdles can be surmounted.

The AREVA reference NGNP concept using multiple tubular IHXs is the best approach, <u>if the direct delivery of nuclear high temperature heat is a fundamental requirement</u>. However, the remaining technical challenges are significant. The overall project risk is manageable, but aggressive action will be required in several key areas in order to achieve project objectives. Alternate paths might better reduce overall risk to an acceptable level in light of a broader interpretation of NGNP project objectives and priorities.

In fact, AREVA believes that the steam cycle concept is the best path forward for near-term HTR deployment. High temperature steam best meets the near-term process heat market for liquid fuels production and is suitable for steam electrolysis hydrogen production. It also provides a more solid foundation for the long-range deployment of advanced concepts as more advanced technology becomes available.

23.0 REFERENCES

1. Ref: “Work Plan – Pre-Conceptual Engineering Services for the NGNP with Hydrogen Production,” BEA Contract No. 00060209, December 13, 2006.

2. Next Generation Nuclear Plant – High Level Functions and Requirements, INEEL/EXT 03-01163, September 2003.

3. Design Features and Technology Uncertainties for the NGNP; ITRG group review, INL EXT-04-01816, June 2004.

4. 4.3 Next Generation Nuclear Plant – Preliminary Project Management Plan, INL/EXT-05-00952 Rev. 1.

5. Reference is for doc id 51-9041508-000).

6. NP-MHTGR, Project Closeout Report, GEGA Project No. 89001, Appendices, Section A.2, Site Investigation, 1993.

7. Environmental and Other Evaluations of Alternatives for Siting, Constructing, and Operating New Production Reactor Capacity, Volume 2, Appendix D, Geologic Resourses, Section D.3, Idaho National Engineering Laboratory. US Department of Energy, September 1992.

8. AREVA NP Document 51-9039141-000, “NGNP with Hydrogen Production System Requirements Manual.”

9. “Guidelines for Facility Siting and Layout”, Center for Chemical Process Safety (CCPS), AIChE, 2003.

10 U.S. NRC, “Title 10, Chapter 1, Code of Federal Regulations”, Published by Office of the Federal Register National Archives and Records Administration, January 1, 2004.

11 U.S. NRC, “Policy for the Regulation of Advanced Nuclear Power Plants,” U.S.NRC, 51FR24643, July 1986.

12 U.S. NRC, “Safety Goals for the Operation of Nuclear Power Plants,” Policy Statement, Federal Register, Vol. 51, No. 149, pp.28044-28049, August 4, 1986.

13 U.S. NRC, “Policy Issues Related to Licensing Non-Light Water Reactor Designs, Policy Issue SECY 03-0047, March 2003

14 U.S. NRC, “Staff Requirements Memorandum for SECY 03-0047 – Policy Issues Related to Licensing Non-Light Water Reactor Designs”, June 2003.

15 Department of Energy, “Preliminary Safety Information Document for the Standard MHTGR,”, DOE-HTGR-86-024, September 1988 transmitted by AREVA Document 38-5055530-00, December 2004.

16 U.S. NRC, “Draft Preapplication Safety Evaluation Report for the Modular High Temperature Gas-Cooled Reactor”, NUREG-1338, March 1989.

17 Exelon Generation Company, “Proposed Licensing Approach for the Pebble Bed Modular Reactor in the United States”, January 31, 2002 transmitted by AREVA Document 38-5055532-00, December 2004.

18 U.S. NRC, “NRC Staff’s Preliminary Findings Regarding Exelon Generation’s (Exelon’s) Proposed Licensing Approach For The Pebble Bed Modular Reactor (PBMR)”, March 26, 2002.

19 U.S. NRC, “10 CFR 52--Early Site Permits; Standard Design Certifications; And Combined Licenses for Nuclear Power Plants” Published by Office of the Federal Register National Archives and Records Administration, January 1, 2004.

20 NUREG/CR-2300, “PRA Procedures Guide,” January 1983.

21. “NGNP with Hydrogen Production – Primary to Secondary Cycle Concept Study”, March 2007, Document no. 12-9045707-000.

22. “Codes and Standards and other Guidance Cited in Regulatory Documents”, Rev. 3 NUREG/CR-5973.

23. NOAK = Nth of a Kind, Mature Reactor Performance After Several Hundred Reactor Years Experience.

24. Next Generation Nuclear Plant Project Preliminary Project Management Plan, March 2006.

25. 51-9039141-000 System Requirements Manual.

26. 51-9041508-001 NGNP Prototype Design Baseline.

27. J. C. Mankins, *Technology Readiness Levels; A White Paper*, NASA Office of Space Access and Technology, NASA (1995).

28. J. W. Bilbro and R. L. Sackheim, *Managing a Technology Development Program*, NASA Marshal Space Flight Center, NASA (2002)

29. C. Robert Kenley of Kenley Consulting & Terry R. Creque of LM Idaho Technologies Company, *"Predicting Technology Operational Availability Using Technical Maturity Assessment,"* (1999)

30. Plot from: Driscoll, M.J., Report No: MIT-GFR-019, “Interim Topical Report, Supercritical CO2 Plant Cost Assessment”, September 2004, Center for Advanced Nuclear Energy Systems, MIT Nuclear Engineering Department.

Doc. No. 12-9051191-001

NGNP with Hydrogen Production

Preconceptual Design Studies Report

APPENDICES

A	Systems Requirements Manual
B	Special Studies
C	Research & Development
D	System Analysis of High Temperature Steam Electrolysis

BEA Contract No. 000 60209

AREVA NP Inc.

PRECONCEPTUAL DESIGN STUDIES REPORT

APPENDIX A

(Issued Previously as 12-9039141-000)

NGNP with Hydrogen Production System Requirements Manual

March 2007

BEA Contract No. 000 60209

AREVA NP Inc.

Record of Revisions (i.e, 51-9039141-000)

Revision	Date	Pages/Sections Changed	Brief Description

Table of Contents

1 Introduction and Structure of the SRM

The System Requirements Manual (SRM) defines the requirements for the NGNP Plant with Hydrogen Production. A similar systems requirements manual will be required for the fuel fabrication plant.

This document is being prepared at the pre-conceptual design phase of the project and will detail the design requirements at this level. This format can be expanded as required for use in the conceptual design and later phases of the project. The scope of the initial effort did not include preparing requirements for all sections. The outline contains the sections needed for a complete SRM but only limited requirements are included in certain sections as necessary to derive the AREVA NGNP design adaptation work.

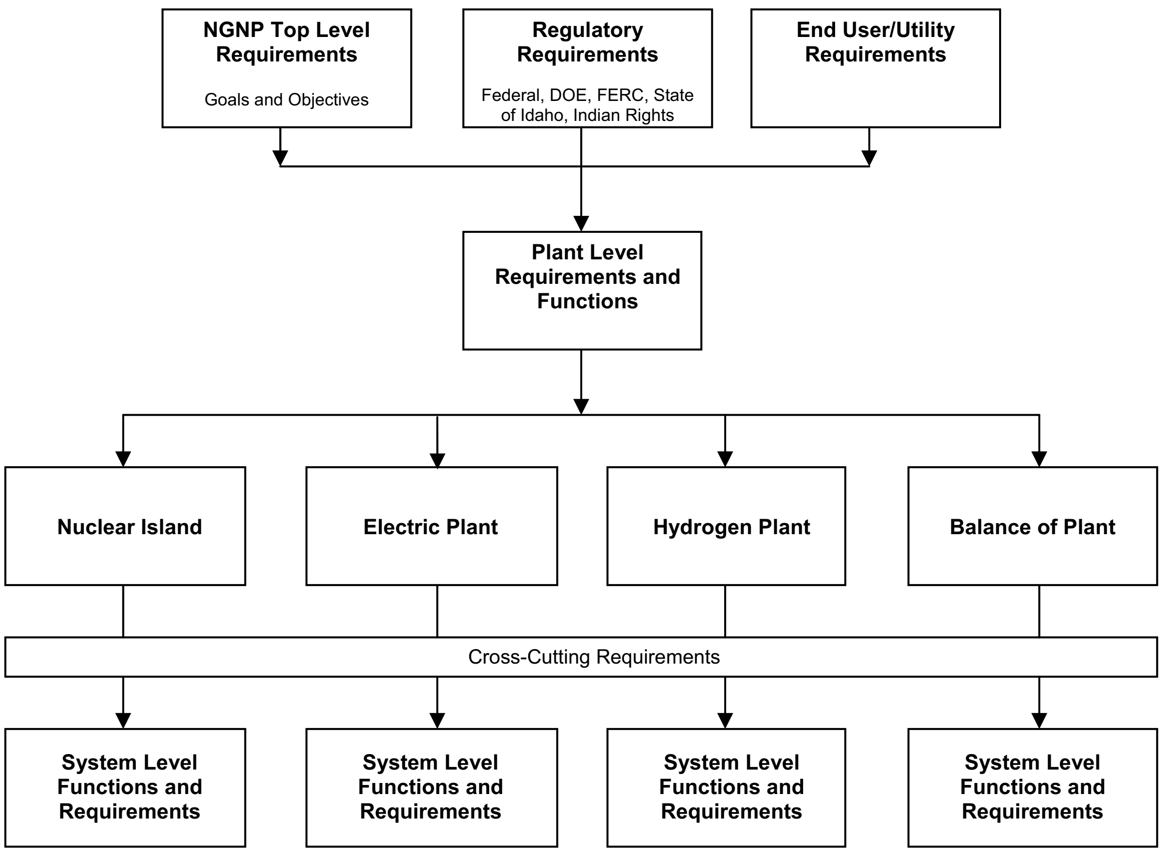
The NGNP process heat load demand drives the nuclear heat source design options and decisions. For the purposes of this project the hydrogen production plant design is not in AREVA scope of work. Therefore, the requirements for the process heat load are based on a hydrogen production plant using the thermo-chemical S-I process or a thermally assisted electrolysis method [1]. Note that the heat load requirements for other hydrogen production schemes can be different from what is assumed for this work, and therefore the design of the nuclear heat source proposed by this set of requirements is optimized for the SI or the high temperature electrolysis methods of hydrogen production.

It is a general practice that systems requirements must be uniquely numbered in order to be referenced and traceable in down-stream design documentation. At the pre-conceptual design stage, it is premature to define an exact requirement numbering system that would allow sufficient flexibility for design, function, and requirement changes that may occur during conceptual design. Therefore, requirements in this document are listed with a rather elementary numbering system and it is recommended that requirements listed in this document be referred to by section number followed by a dash and the requirement number. For instance, requirement #3 from Section 7.3.1.2 would be referred to as 7.3.1.2 - 3.

2 Requirements Hierarchy

2.1 Overall Requirements Hierarchy

The following block diagram shows the proposed hierarchy for the requirements of the NGNP. At the highest levels are the project goals and objectives as well as the regulatory requirements for the plant, which are documented in Section 3.0. From these requirements, the plant level functions and requirements are defined and documented in Section 4.0 and Section 5.0, respectively. The functions and requirements of each of the individual systems and sub-systems are then defined in Section 7.0; each of these system level functions and requirements is based upon one or more of the top-level requirements defined above.



2.2 Mature Requirements Development Process

From a systems engineering and functional analysis point of view, a formal process for developing requirements at each level must be applied. At the top level of this project, requirements are drawn from NGNP program goals and objectives as well as various regulatory requirements. These top level requirements are all externally defined. At the plant level, requirements are developed by determining what will be required of the plant to satisfy each of the top level requirements, this may involve several levels of requirements at the plant level. From this point, systems requirements, divided between each of the areas of the plant (Nuclear Island, Electric Plant, Hydrogen Plant, and Balance of Plant), and possibly involving several levels of requirements, are developed by determining what will be required of each system, based upon each of the above requirements. This process is repeated at the sub-system and eventually the component level.

2.3 Pre-Conceptual Design Requirements Process

In the pre-conceptual design phase, the plant design is not at a mature enough level to strictly follow the process as described in Section 2.2. Therefore, this phase uses a simpler approach. It lists top level requirements taken from NGNP project goals and objectives as well as various regulatory requirements. At the system and sub-system level, however, requirements were developed based upon requirements from other more mature designs, such as the AREVA ANTARES concept as well as the General Atomics GT-MHR and MHTGR designs.

In the conceptual design phase of this project a more structured and rigorous approach to system engineering would be necessary. This will be a top-down approach to system design development and AREVA will use a standard computer software tool such as CORE™ from Vitech Corporation. System engineering processes and procedures will be uniquely configured and implemented at the start of conceptual design.

2.4 Pre-Conceptual Design SRM

This Systems Requirements Manual document will contain general requirements for each level of the plant design, from the top level requirements to the plant level and eventually the system and sub-system levels. The set of requirements developed here are considered sufficient for the NGNP pre-conceptual design adaptation work and set the starting point for the subsequent conceptual design phase.

3 NGNP Top Level Requirements

3.1 Prototype Mission Objectives

The NGNP project mission is the development of a prototype design for a full-scale commercial plant which provides (a) high efficiency electricity generation and (b) CO_2 free hydrogen production based on high temperature modular gas-cooled reactor technology as the driving heat source.

3.2 NGNP Goals and Objectives

3.2.1 Overall Objectives

The NGNP project has the following objectives, taken from Reference 3:

1. The NGNP shall develop and implement the technologies important to achieving the functional performance and design requirements determined through close collaboration with commercial industry. [1]

2. The NGNP shall demonstrate the basis for commercialization of the nuclear system, the hydrogen production facility, and the power conversion concept. An essential part of the prototype operations will be demonstrating that the requisite reliability and capacity factor can be achieved over an extended period of operation[1]

3. The NGNP shall establish the basis for licensing the commercial version of NGNP by the Nuclear Regulatory Commission (NRC). This will be achieved in pajor part through licensing the prototype by NRC and initiating the process for certification of the nuclear system design.[1]

4. The NGNP shall foster the rebuilding of the U.S. nuclear industrial infrastructure and contributing to making the U.S. industry self-sufficient for our nuclear energy production needs.[1]

3.2.2 NGNP Project Requirements

The following requirements, which do not have direct references, are adapted from Reference 2 as modified by the Independent Technology Review Group Recommendations as defined in Appendix B of Reference 2.

1. NGNP shall be designed, constructed, licensed, and operating by 2020 with initial operations in 2018.

2. NGNP design configuration shall consider cost and risk profiles to ensure that NGNP establishes a sound foundation for future commercial deployment.

3. NGNP nuclear heat source shall be based on the modular high temperature gas-cooled reactor concept and utilize passive safety features to cool the core from full power to safe shutdown conditions.

4. NGNP shall produce high efficiency electricity and generate hydrogen on a scale that sets a foundation for future commercial deployment.

5. NGNP shall be licensed by the NRC as a commercial cogeneration facility producing electricity and hydrogen.

6. NGNP shall include provisions for future testing.

7. NGNP shall enable demonstration of energy product and processes utilizing its nuclear heat source.

8. The project shall include identification of necessary and sufficient R&D technical scope and priorities.

9. NGNP plant licensing shall support potential future US-NRC technology neutral rulemaking activities (i.e. Risk-Informed, Performance-Based Alternative to 10CFR Part 50).[3]

10. For the purposes of the AREVA pre-conceptual design activity, existing AREVA Nuclear Island and Power Conversion System Shall be utilized as the bases for the NGNP pre-conceptual design studies.

3.3 Regulatory Requirements

The following regulatory requirements, from various regulatory bodies are applicable to the NGNP.

3.3.1 NRC/EPA Regulatory Requirements

Federal regulatory requirements are defined by the Code of Federal Regulations, which is controlled by several regulatory bodies, such as the NRC and the EPA.

1. 51 FR28044 - Policy Statement on safety Goals for the Operation of Nuclear Power Plants
2. 10 CFR 20 – Standards for protection against radiation - Permissible dose levels and activity concentrations in restricted and unrestricted areas.
3. 10 CFR 50 – Domestic Licensing of Production and Utilization Facilities (applicable portions as needed)
4. 10 CFR 51 – Environmental protection regulation for domestic licensing and related regulatory functions
5. 10 CFR 52 – Early site permit; standard design certification; and combined license for nuclear power plants
6. 10 CFR 50 Appendix I – Numerical dose guidelines for meeting the criterion “ALARA” for power reactor effluents
7. 10 CFR 73 – Physical Protection of Plants and Materials
8. 10 CFR 74 – Material Control and Accounting of Special Nuclear Material
9. 10 CFR 75 – Safeguards on Nuclear Material – Implementation of US/IAEA Agreement
10. 10 CFR 95 – Security Facility Approval and Safeguarding of National Security Information and Restricted Data
11. 10 CFR 100 – Reactor site criteria - Numerical dose guidelines for determining the exclusion area boundary, low population zone, and population center distances.
12. 29 CFR 1910 – Occupational Safety and Health Standards, Subpart H – Hazardous Materials
13. 40 CFR 50-99 – Clean Air Act
14. 40 CFR 100-149 – Clean Water Act
15. 40 CFR 190 – Environmental radiation protection Standards for Nuclear Power Operations
16. 40 CFR 1502 – Environmental Impact Statement
17. EPA – 520/1-75-001 – protective Action Guide Doses for Protective Actions for Nuclear incidents
18. 47 FR47073 – Accident radioactive Contamination of Human Food and Animal Feed; Recommendations for State and Local Agencies
19. NGNP plant licensing shall comply with the US-NRC new technology neutral regulatory framework as described in NUREG 1860, July 2006.

3.3.2 DOE Requirements

If the NGNP becomes a DOE procurement, the below requirements will become effective.

1. DOE O 413.3A – Program and Project Management for the Acquisition of Capital Assets
2. DOE O 420.1B – Facility Safety
3. DOE O 435.1 – Radioactive Waste Management

4. DOE Policy 450.4 – Safety Management System Policy

3.3.3 Federal Energy Regulatory Commission (FERC) Requirements

FERC sets requirements for all electricity being fed into the national power grid. Since the NGNP is expected to produce electricity for commercial use, it must follow applicable requirements of FERC.

3.3.4 State of Idaho Regulatory Requirements

The NGNP will be located at INL in the State of Idaho and, therefore, must meet applicable state requirements related to this technology.

3.3.5 Indian Reservation Rights and Requirements

The Shoshone-Bannock Tribes are the region's primary Native American residents. Because they believe the land is sacred, the entire Idaho National Engineering Laboratory reserve is potentially culturally important to them. Cultural resources, to the Shoshone-Bannock peoples, include all forms of traditional lifeways and usage of all natural resources. This includes not only prehistoric archaeological sites, which are important in religious or cultural heritage context, but also features of the natural landscape, air, plant, water, or animal resources that might have special significance. DOE has committed to additional interaction and exchange of information with the Shoshone-Bannock Tribes at the Fort Hall Reservation.

4 Functions of the plant (Nuclear Island, Electric Plant, Hydrogen Plant, Balance of Plant)

The following is a list of the primary functions of the plant. Each of the requirements of the systems defined below in Section 7.0 supports one or more of these functions.

1. Produce Safe and Economical Electrical Power
2. Demonstrate scaled production of Safe and Economical Hydrogen Fuel
3. Maintain Control of Plant Operation
4. Maintain Plant Personnel Safety and Public Protection
5. Maintain Social & Environmental Acceptability
6. Maintain Emergency Planning and Preparedness
7. Maintain Non-Proliferation

5 Top Level Plant Requirements

The top level plant requirements include restraints based on commercial considerations, customer specific requirements, any design constraints placed on the project, safety considerations and regulatory considerations. The top level requirements are listed below:

1. The electric plant shall demonstrate electricity production with an efficiency as high as reasonably achievable.
2. The electric plant shall demonstrate an efficiency of at least [44%].
3. The electric plant shall feed an outside commercial electrical grid system
4. The electric plant shall provide electricity at 60 Hz to the electrical grid system
5. The hydrogen production plant shall demonstrate commercial scale production and economics
6. The NGNP hydrogen product shall be sent to (TBD) for commercial usage.
7. No provisions shall be made for onsite produced hydrogen storage.
8. The plant shall meet the NRC commercial power plant licensing requirements and be licensed for commercial operation supporting creation of regulatory requirements and acceptance criteria for future plant licensing similar to NGNP design.
9. The NGNP shall be designed in accordance with defense-in-depth principles and philosophy with the intent to eliminate the need for evacuation due to any nuclear plant design basis accident beyond the plant site boundary of 450 meters.
10. The nuclear heat source shall use TRISO-coated particle fuel in prismatic fuel blocks
11. The nuclear heat source shall use once through uranium fuel cycle with uranium enrichment less than 20% ^{235}U.
12. Fuel burn-up shall be such that it maximizes uranium utilization, minimizes proliferation risks, minimizes waste streams for open fuel cycle, and optimizes fuel economics.
13. The primary nuclear heat source shall utilize an indirect cycle heat transport system (although a study will be performed addressing alternatives).
14. Hydrogen production plant will require high temperature ([TBD]) process heat from the nuclear plant.
15. The plant shall be designed to the USA industrial Codes and Standards as necessary to meet the USA industrial and regulatory requirements.
16. The nuclear heat source shall be passively cooled in case of loss of all off-site and on-site motive power.
17. The plant design shall result in ease of required maintenance activities with appropriate considerations for layout space, access, and maintenance equipment.
18. The nuclear plant fuel loading shall allow an [18 month] fuel cycle.
19. The NGNP shall demonstrate commercial plant viability.
20. The commercial nuclear plant shall be designed for 60 year life and with fifteen year onsite used fuel storage capacity within the plant boundary fence.
21. The nuclear plant shall be designed for a high availability [≥ 90%]
22. Public safety due to any accident at the nuclear plant shall not depend on public or personnel beyond site boundary.
23. The nuclear plant design basis accidents shall not lead to nuclear plant write-off.
24. The plant shall be located at the NPR site at the Idaho National Laboratory, Idaho USA, approximately 3.5 miles northeast from the INTEC facilities.
25. The plant shall be designed such that the Nth-Of-A-Kind plant costs will be economically competitive with a comparable energy source.

26. The NGNP technology specific licensing requirements shall be derived as necessary using "risk-informed and performance-based" framework and proposed as license "exception and/or condition" under 10 CFR Part 50 licensing regulations.

6 Plant Technical Overview

Provided in this section is an overview of the AREVA NGNP reference configuration for pre-conceptual design. The description presented in this section are not design requirements.

A reference design for the NGNP concept was established early in the pre-conceptual design program. This design was selected based on experience from participation in prior HTR programs.

The main features of the reference NGNP design are held largely fixed during the course of the pre-conceptual design effort in order to provide a stable environment for the development of detailed design information and analyses. A number of special studies have been performed or are ongoing to support major design decisions, and separate peripheral studies of alternative configurations are in progress where appropriate. However, changes to the reference design have only been made in a disciplined controlled process (i.e., as a result of a trade study). The objective of this approach has been to develop a detailed understanding of the reference design including the difficulties which might be encountered in its completion. This knowledge, combined with the broader perspective obtained from the more limited evaluations of alternatives will provide a strong foundation for the program in its next phase (i.e., Conceptual Design).

6.1 Overall Description of AREVA NGNP Concept

The reference NGNP concept is based on a modular HTR coupled to a combined cycle gas turbine (CCGT) as its electricity generating system. An indirect cycle configuration was selected in which heat from the reactor is transferred to a closed loop Brayton cycle through an Intermediate Heat Exchanger (IHX). A nitrogen based fluid is used in the secondary circuit in order to allow air-breathing gas turbine technology to be used.

The reference NGNP concept has remained relatively static throughout the pre-conceptual design phase. The reactor outlet temperature is 900 ºC and four heat exchangers are utilized, three for the electricity generating loops and one for the hydrogen production loop. Section 2.1 of the NGNP with Hydrogen Production Design Baseline defines the plant parameters.

The indirect cycle offers several advantages to reduce the overall development risk in contrast to a direct cycle concept. As already noted, selection of the indirect cycle allows the use of air-breathing gas turbine technology, avoiding the development of helium turbo-machinery. The indirect cycle also makes maintenance and potential modification and adjustment of the system more practical, since the equipment is distributed rather than being in a tightly integrated configuration. Also, contamination of the power generating equipment is avoided, since contamination is confined to the primary circuit. In addition, operation and analysis of plant performance is simplified, because the dynamics of the reactor and primary circuit are partially decoupled from the power generating system.

Figure 6-1 shows the overall plant layout. Only one IHX is shown in this figure for simplicity.

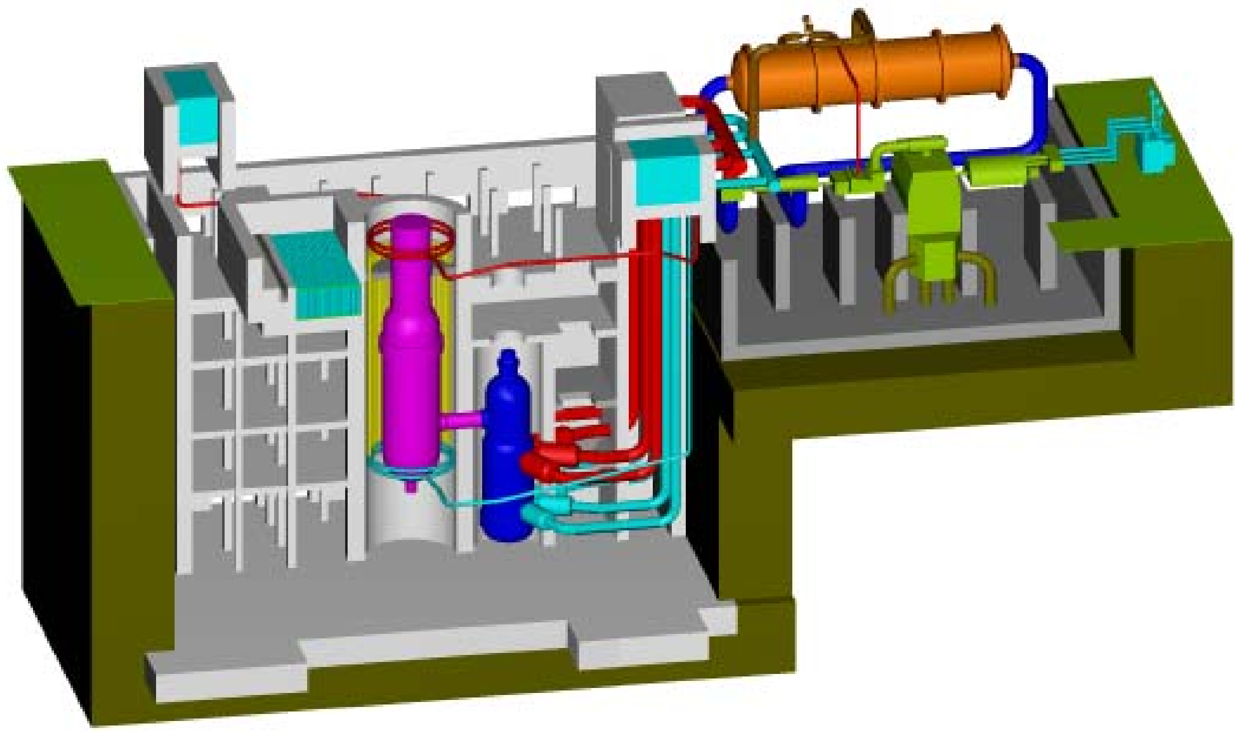


Figure 6-1: Overall AREVA NGNP Plant Layout

6.1.1 Nuclear Heat Source

The Nuclear Heat Source (NHS) is a graphite-moderated, helium-cooled nuclear reactor. This type of reactor has the capability to supply high temperature heat for a variety of applications, and its excellent safety characteristics provide advantages in locating plants, in investment protection, and in minimizing the number of active safety systems required. The NGNP uses a completely ceramic prismatic block reactor core. The absence of metal alloys in the core allows very high reactor outlet temperatures to be achieved during normal operation.

The fuel consists of approximately 10 billion ceramic particles, each being about 1 mm in diameter. Each "TRISO" particle has a uranium oxide or oxy-carbide fuel kernel at the center surrounded by three successive layers of low and high density carbon and silicon carbide. Figure illustrates the coating layers, including a final outer layer of high density carbon which protects the SiC during manufacturing. These layers retain fission products within the fuel particles during normal operation and accident conditions.

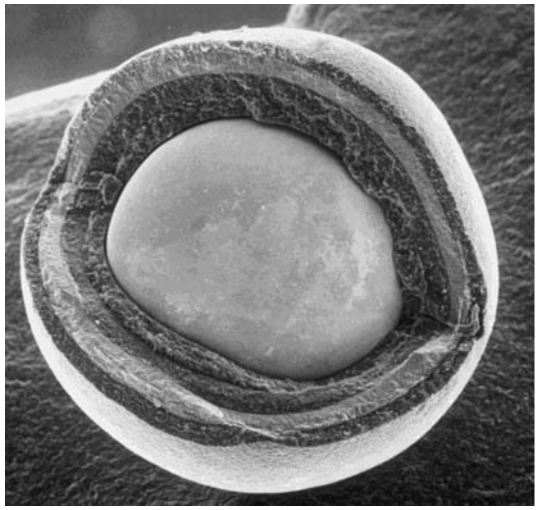


Figure 6-2: Internal Structure of TRISO Coated Fuel Particle

The fuel particles are molded into cylindrical rods called compacts, and loaded into the prismatic fuel blocks. All modern HTRs use TRISO fuel, but in some concepts the particles are loaded in prismatic blocks while in others they are placed in spherical “pebbles”. Both concepts have advantages and disadvantages. The assembly concept has been selected for AREVA NGNP, because it provides precise control of fuel placement (advantageous for many fuel cycle schemes), and because it allows a higher core power level while maintaining the desired passive safety characteristics. The fuel blocks have axial holes for coolant flow and closed holes to contain the fuel compacts (Figure).

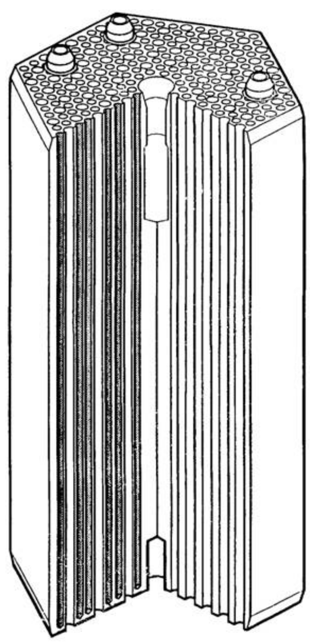


Figure 6-3: HTR Prismatic Fuel Element

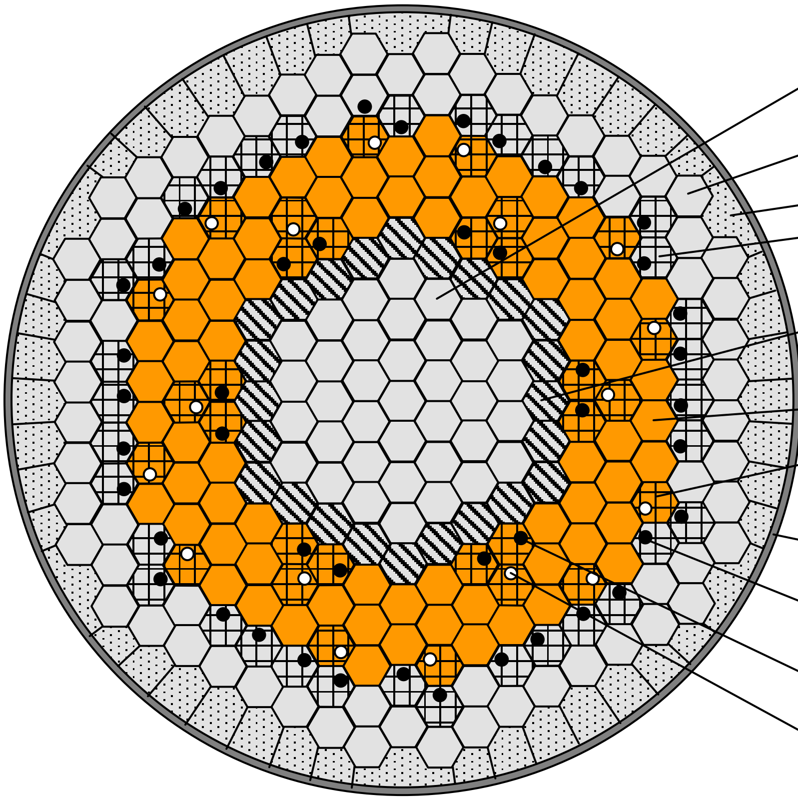


Figure 6-4: Reference ANTARES Core Layout

The AREVA NGNP concept uses an annular core as shown in Figure . The active core consists of 102 columns of 10 blocks each, for a total 102 fuel elements. This configuration is a direct adaptation of the AREVA ANTARES configuration.

Heat produced in the reactor is transferred to four Intermediate Heat Exchangers (IHXs) via the Primary Heat Transfer System (PHTS). The IHX is the functional interface between the NHS and the energy user. The AREVA NGNP concept uses four IHXs, one for a hydrogen production loop, and three for electricity generation. In addition to the IHX, the PHTS includes the main helium circulators, and the required ducting to channel the coolant from the reactor to the IHXs, from the IHXs to the circulators, and from the circulators back to the reactor inlet.

The NHS systems are contained in a steel vessel system. The reactor pressure vessel is fabricated from Modified 9 Cr – 1 Mo. This alloy provides increased high temperature capability which is compatible with the reactor operating temperatures and provides sufficient margin for off-normal events.

Nine vessels form the primary coolant envelope. The Reactor Vessel contains the reactor core and internals. The four IHX Vessels contain the modules of the IHXs, the main circulators, and the necessary interconnecting primary and secondary coolant ducts. The Cross Vessels connect the Reactor Vessel and the IHX Vessels, and it contains the concentric flow path between the reactor and IHXs. During normal operation the IHXs separate the primary and secondary coolant, however isolation valves are provided to complete the vessel boundary in case of IHX or secondary duct leakage.

6.1.2 Power Generating System

As already stated, the reference NGNP concept uses a CCGT for electricity generation. This Power Generating System is very similar to that of a modern natural gas-fired combined cycle plant. The top part of the CCGT system is a closed loop Brayton cycle. A closed loop gas turbine system is used for the HTR, since combustion air is no longer required. The waste heat not used by the gas turbine is transferred to the Rankine steam cycle through the Heat Recovery Steam Generator. The remaining usable energy is extracted by the conventional steam system.

A nitrogen based mixture is used as the working fluid in order to simulate the properties of air. As was already stated, this allows the use of conventional air-breathing gas-turbine technology. Additionally, the use of air-based gas-turbine technology means that conventional design approaches and analysis techniques can be used, and extensive developmental testing is not required.

For given conditions, the combined cycle provides inherently higher efficiency than a simple Brayton cycle. This overcomes the loss in efficiency that would otherwise result from the selection of the indirect cycle configuration.

6.1.3 Hydrogen Production System

6.1.4 High Temperature Heat Transport System

6.1.5 Heat Removal Systems

Heat removal is normally provided by the Primary Heat Transfer System (PHTS) which transfers heat from the NHS to the secondary coolant and the Power Generating System through the IHXs.

A Shutdown Cooling System (SCS) provides decay heat removal when maintenance is to be performed on the PHTS or the secondary system. The SCS circulates primary coolant from the reactor core to a shutdown cooling heat exchanger where the heat is transferred to a secondary water loop. The SCS is not classified as a safety system. However, the system is robust. It is designed not only for the normal decay heat loads, such as during refueling and maintenance, but also for the more extreme thermal conditions encountered when removing decay heat following a sustained loss of forced convection (normally supplied by the main circulator).

If all active decay heat removal systems are unavailable, the Reactor Cavity Cooling System (RCCS) is relied upon. The RCCS is a natural circulation water system which removes heat from the reactor cavity (Figure). This passive system is the only heat removal system classified as a safety system. The reactor has two redundant RCCS cooling loops. Heat is carried from the reactor core to the reactor vessel by conduction and thermal radiation and by radiation and convection to the RCCS panels lining the concrete walls of the reactor cavity. The RCCS provides cooling for the vessel and concrete during both normal operation and accident conditions. Normally, heat removed by the RCCS is dissipated by a non-safety secondary cooling system. During accident conditions, the RCCS heat is dissipated by long-term boiloff of the large volume of water in the RCCS water storage tanks.

The power removal capability of the SCS and RCCS will be developed further in the conceptual design phase.

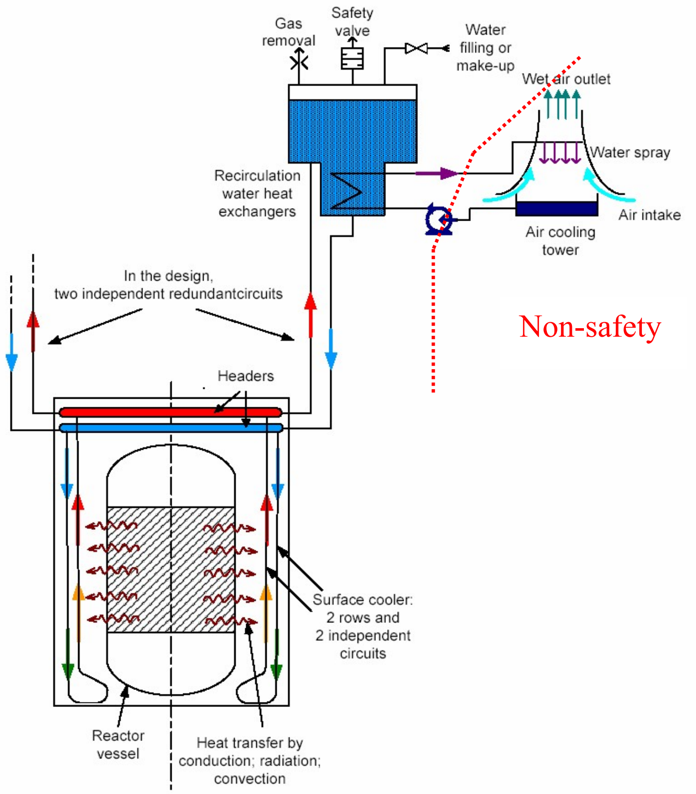


Figure 6-5: Reactor Cavity Cooling System

6.1.6 Supporting Systems

As is the case for any power generating facility, there are a number of necessary supporting systems. Two such systems which are unique for the ANTARES concept are the Fuel Handling System and the Helium Purification System.

6.1.6.1 Fuel Handling System

HTR refueling is performed using robotic systems with the primary coolant boundary intact. Following shutdown, the primary system temperature is reduced to approximately 100 ºC to ensure fuel block surface temperatures are less than 400 ºC, precluding graphite oxidation. At the same time, the primary system helium inventory is reduced and then maintained to keep the primary system pressure slightly sub-atmospheric. Upon satisfying these conditions, refueling access is then gained through the control rod drive penetrations at the top of the reactor vessel.

At each refueling, the reactor is refueled in 1/6 core segments. All fuel elements from the segment are removed, and then the segment is rebuilt using the appropriate set of new and partially irradiated elements. Adjacent reflector elements are also replaced at this time, if they have reached the end of their lifetime. Processing then moves to the next segment.

The robotic refueling equipment is computer controlled using predetermined block movement sequences. The process is also monitored by system operators. Multiple features are built into the system to ensure that all blocks are properly positioned. All blocks are uniquely identified, and the location and position of each block is verified and recorded throughout the refueling process. Each block also has interlocking dowels and sockets which prevent misalignment.

The AREVA NGNP refueling process is identical in principle to that used successfully in previous HTRs that used block type fuel. The systems are updated to take advantage of current robotics and computer technology in order to maximize system efficiency and reliability. Based on current studies, refueling is expected to be accomplished in the 21-day target refueling window; however, further refinement and confirmation of fuel handling times will be performed during the conceptual design phase.

6.1.6.2 Helium Purification System

The primary coolant purity is maintained by the Helium Purification System. The two-fold purpose of this system is to maintain the primary coolant composition in order to maximize the lifetime of reactor components and to remove fission products released by the fuel in order to control circulating activity.

A small fraction of the primary coolant is continuously removed from the primary circuit for purification. The purification system passes this side stream of gas through a series of filters, cold traps, and charcoal beds to remove the undesired impurities. The purified helium can then be returned to the primary circuit or sent to storage as appropriate.

Helium purification as described is based on existing technology which relies on the use of cryogenics to sufficiently cool the helium. Promising advances in the use of filter membranes (being developed at the University of Montpelier), may result in system simplification by eliminating the need for cooling down the helium, eliminating the need for cryogenic equipment.

6.1.7 AREVA NGNP Safety Philosophy

The safety design for any nuclear power plant addresses the selection of the inherent features of the reactor, i.e. the choice of a fuel, moderator, and coolant and the basic materials and design characteristics of these, the design of the barriers to contain the inventory of radioactive material, and the safety functions that are provided to maintain and protect these barriers. In addition to the inherent reactor features, the safety design includes the design of systems, structures, and components (SSCs) that provide the safety functions and the approaches that are taken to support the capability and reliability of these SSCs. These approaches include the use of passive and active SSCs, and appropriate application of redundancy and diversity.

The AREVA NGNP concept benefits from the experience of the prototype, testing, and demonstration HTGRs designed, built, and operated to date with their ceramic-coated fuel particles, graphite moderator, and inert helium coolant, which comprise the inherent characteristics of all modular HTGRs. A safety objective of the AREVA NGNP design is to achieve the public safety without needing to implement off-site emergency management measures. To accomplish this with high assurance, the design is guided by a safety design with emphasis on radionuclide retention at the source within the fuel particles with minimal reliance on active design features or operator actions.

This design philosophy has a profound impact on the design of the AREVA NGNP in two important ways. First, the philosophy requires control of radionuclide releases primarily by retention within the coated fuel particles. This decreases reliance on secondary barriers (such as the helium pressure boundary or the reactor building) and therefore their costs. This leads to important design selections: the type of fuel, the specification of its as-manufactured quality, and its required in-service performance. Proof of radionuclide retention is dramatically simplified if arguments can focus on issues associated with fuel particle integrity alone.

Second, the philosophy requires that control of radionuclides be accomplished with judicious reliance on passive systems and minimal reliance on active systems and prompt operator actions. This leads to fundamental design selections such as the core size and geometry, the power density and vessel type. By minimizing the need to rely on active systems or operator actions, the safety strategy centers on the behavior of well understood laws of physics and on the integrity of passive design features. Arguments need not center on an assessment of the reliability of active pumps, valves, and their associated services or on an operator rapidly taking various actions, given the associated uncertainties involved in such assessments. In fact, an explicit objective of the AREVA NGNP safety design approach is to provide long response times for operator actions, on the order of several hours or days instead of a few minutes.

An important element of the safety design of the AREVA NGNP is to demonstrate that the principles of defense-in-depth are effectively applied. These include implementation of barrier defense-in-depth, process defense-in-depth, and scenario defense-in-depth principles. The AREVA NGNP uses inherent and passive capabilities to prevent the release of radioactive material from the fuel during both design basis and beyond design basis accident conditions. Compared to other nuclear plants, safety design emphases the radionuclide retention by the coated particle fuel, with much more reduced retention requirements on the other barriers. This leads in particular to exclude any severe fuel failure. This is probably the main safety issue which has to be accepted by licensing authorities.

The overall intent is to provide a simple safety case that provides high confidence that the safety requirements are met.

7 System Functions and Requirements

7.1 Requirements applicable to multiple Systems, Buildings and Structures

7.1.1 System Configuration and Essential Features Requirements

1. The NGNP nuclear heat source shall use modular high temperature gas-cooled reactor concept
2. The NGNP nuclear heat source shall demonstrate commercial viability of modular high temperature gas-cooled reactor
3. The NGNP nuclear heat source shall be connected to a power conversion system for demonstration of high efficiency [>44%] commercial scale electricity generation
4. The NGNP nuclear heat source shall be connected to a hydrogen production demonstration plant through an intermediate loop and intermediate heat exchanger and deliver ~60 MWth of process heat.
5. The overall layout shall not favor natural convection of primary coolant.

7.1.2 Operational Requirements

1. The NGNP nuclear heat source shall have an operational lifetime of 60 years
2. The NGNP required operational lifetime shall be met by using components designed for a 60 year lifetime or by using components that are replaceable
3. The NGNP nuclear heat source and the power conversion system shall be designed and licensed as a commercial nuclear utilization facility for generation of electricity and process heat
4. The NGNP nuclear heat source shall be designed for load following of the electricity generation plant
5. The NGNP nuclear heat source shall be designed for load following in the hydrogen production plant
6. The NGNP nuclear heat source shall be designed to use prismatic fuel blocks
7. The NGNP shall use low enriched uranium (LEU) TRISO coated particle fuel
8. The NGNP shall demonstrate an [18 month] refueling interval capability
9. The NGNP shall be designed to operate during loss of hydrogen production and stabilize in the electricity generation phase.

7.1.3 Structural Requirements

1. [NGNP plant nuclear reactor and major nuclear systems and components including the IHX shall be located below grade]
2. NGNP plant systems and structures and components shall be designed and constructed using and demonstrating modular plant construction
3. NGNP plant external structures, important to nuclear safety, shall be designed and constructed to withstand the impact of a single large commercial airliner without exceeding the 1.0 rem protective action guide (PAG) radioactive exposure limit at the site boundary

7.1.4 Environmental Requirements

1. The NGNP and hydrogen production facilities shall comply with applicable requirements of the Clean Air Act/Air Programs.
2. The NGNP and hydrogen production facilities shall comply with applicable requirements of the Clean Water Act/Water Programs.
3. The NGNP project shall minimize the generation of all wastes, including radioactive, non-radioactive, and mixed wastes and it shall comply with applicable DOE Orders, NRC Regulations, and EPA Regulation in the treatment of these wastes.

7.1.5 Instrumentation and Control Requirements

1. The NGNP plant shall be controlled from a single control room.
2. The main control room shall include controls for the power conversion system (PCS) and high temperature heat transport loop.

7.1.6 Surveillance and ISI requirements

7.1.7 Availability Requirements

7.1.8 Maintenance Requirements

7.1.9 Safety Requirements

1. The nuclear system shall not depend on active cooling systems during design basis accident conditions

7.1.10 Codes and Standards Requirements

1. U.S. Commercial Codes and Standards shall be selected and followed for all structures and systems as appropriate.

7.1.11 Quality Assurance Requirements

1. The NGNP project shall use the U.S. national consensus standard ASME NQA-1-1997 "Quality Assurance Program Requirements for Nuclear Facilities Applications" and Subpart 4.2 of ASME NQA-1-2000 "Guidance on Graded Application of Quality Assurance (QA) for Nuclear-Related Research and Development" for project specific development R&D activities.

7.1.12 Construction Requirements

7.1.13 Decommissioning Requirements

7.2 Fuel Requirements

The fuel shall be designed with the following requirements:

- As-manufactured quality requirements:

1. Allowable [TBD] failure of fuel particle coatings at the time of manufacture
2. Free uranium contamination in fabricated fuel

- In-service fuel performances requirements:
  3. Fuel performance retention capabilities during normal operation (accounting for the failure of fuel particle coatings and, if significant, for the radiocontaminants diffusion in the fuel particles).
  4. Fuel performances retention capabilities during off-normal events (accounting for any incremental failure of fuel particle coatings and, for any increased diffusion of radiocontaminants in the fuel particles)

7.3 Nuclear Heat Source Requirements

7.3.1 Reactor System

The following are the required functions of the reactor system:

1. Generate heat and transfer it to the primary coolant
2. Maintain reactor shutdown

The reactor system shall be designed with the following requirements:

3. The reactor system shall be designed to provide the possibility of passive residual heat removal.
4. The reactor system shall be designed for an operational lifetime of 60 years.
5. The reactor system shall be designed to provide dual hydrogen and electricity generation.
6. The core shall use forced circulation helium as the heat transport fluid.
7. Non replaceable structural materials in contact with helium shall resist corrosion and erosion during plant cycle life.

7.3.1.1 Reactor Core

The following are the required functions of the reactor core:

1. Generate heat
2. Transfer heat to coolant and/or reactor internals

The following requirements shall be placed on the reactor core:

1. The decay heat removal shall be possible by passive heat transfer means(conduction and radiation) from the fuel to the reactor internals without reaching unacceptable fuel temperatures during all design basis accident conditions.
2. The core shall utilize thermal spectrum neutrons for fission reaction
3. The core shall be moderated with graphite.
4. The active core height shall ensure the axial stability of the neutron flux and preclude the risk of xenon oscillations.
5. Reference fuel shall be low enriched uranium based (UCO or UO_2) with an enrichment limited to <20.0% (in mass) and with a peak burn-up limited to 20% FIMA.
6. The core bypass flow shall be maintained within an acceptable range [TBD] which ensures a good compromise for the fuel temperature in normal and accidental conditions (existence of a minimum amount of bypass in lateral reflector).
7. The fuel handling sequences for core reloading shall be completed within a time interval consistent with the prescribed outage period.
8. The reactivity temperature coefficient shall be sufficiently negative to shutdown the nuclear chain

reaction before an unacceptable fuel temperature is reached, and maintain the core in a safe state for a time offering the certainty to reliably introduce absorber elements.

7.3.1.2 Reactor Internals

The following are the required functions of the reactor internals:
1. Maintain reactor core geometry
2. Provide heat transfer during conduction cooldown
3. Conserve neutrons in the reactor core and provide shielding

The following are requirements on the reactor internals:
1. The reactor internals shall be designed to properly control bypass flows.
2. The reactor internals shall be designed to transport residual decay heat from the reactor core to the reactor vessel.
3. The reactor internals shall be designed to channel primary coolant to and from the reactor core for transfer of heat to the Primary Heat Transport System (PHTS).
4. The reactor internals shall be designed to provide radiological shielding to limit neutron fluence to the reactor vessel.
5. The reactor internals shall be designed to limit gamma radiation exposure to the plant personnel and equipment.
6. The reactor internals shall be designed to limit damage to plant components during conduction cooldown events

7.3.1.3 Neutron Control Elements

The following are the required functions of the neutron control elements:
1. Control the nuclear chain reaction in the reactor core by absorbing neutrons in any operational mode

The following are requirements placed on the neutron control elements:

1. The neutron control elements shall be designed to provide sufficient negative reactivity to shutdown the reactor and maintain it in sub-critical condition for any state by compensating the worst positive reactivity insertion.

7.3.2 Vessel System

The following are the required functions of the vessel system:
1. Contain and support the components of the reactor core, reactor internal supports and structures, and the nuclear heat transport components

The Vessel System shall be designed with the following requirements:
1. Design commonalities between dual electrical and process heat plants shall be maximized.
2. The duration of maintenance, in-service inspection (ISI), and repair/replacement operations of the Vessel System shall be minimized.
3. All parts of the Vessel System shall be designed for operation duration of 60 years.
4. Lifetime of isolation valves of the Vessel System shall be optimized according to the investment cost and replacement duration.
5. The vessel system shall be designed for design basis duty cycle events

7.3.2.1 Reactor Vessel

The following are the required functions of the reactor vessel:

1. Provide core support and maintain its relative position to the control rods.
2. Provide decay heat and residual heat removal by radial conduction during conduction cooldown

The following are requirements placed on the reactor vessel:

1. During normal operation, the reactor vessels shall maintain its operating temperature through a thermal balance between the core heat flux, core inlet helium flow and the reactor cavity cooling system.
2. The reactor vessel shall maintain the primary pressure boundary integrity
3. The operating conditions shall be considered according to the following statements
    - or normal operation, the creep effects on the reactor vessel shall be avoided (negligible creep).
    - For Anticipated Operating Occurrences (AOO) and design basis accidents, the reactor vessel shall not prevent re-starting of the plant within a time period compatible with the availability requirements. In particular, no leakage resulting from AOO.

7.3.2.2 Cross Vessels

The following are the required functions of the cross vessels:

1. Provide a primary heat transport path to/from reactor vessel and IHX vessels

The following are requirements placed on the Cross Vessels:

1. Maintain primary pressure boundary integrity
2. Primary heat transport path to/from reactor vessel and IHX vessels

7.3.2.3 IHX Vessels

The following are the required functions of the IHX Vessels:

1. Support the reactor internals & IHX modules

The following are the required functions of the IHX Vessels:

1. The IHX vessels shall maintain the primary pressure boundary integrity

7.3.2.4 Vessel Supports

The main requirements of the Vessels Supports are:

1. The vessel supports shall support the vertical load
2. The vessel supports shall include keying for lateral support
3. The vessel supports shall accommodate thermal expansion
4. The vessel supports shall accommodate duty cycle events

7.3.2.5 Pressure Relief System

The following are the required functions of the pressure relief system:

1. Provide the primary coolant loop's overpressure protection as required by ASME pressure relief code

The following requirements are placed on the pressure relief system:

1. The pressure relief system shall be designed to depressurize the primary system in the following conditions:
   - In case of primary overpressure, the safety valves shall open to eliminate the overpressure and re-closes once the overpressure condition terminates.
   - The redundancy of the primary pressure relief system may be required for investment protection reasons.

7.3.3 Primary Heat Transport System

The following are the required functions of the Primary Heat Transport System (PHTS):

1. Transfer heat from the reactor core to the secondary circuit

The PHTS shall be designed with the following requirements:

1. Pure helium shall be used in the secondary circuit for the H_2 plant heat transfer and a He/N_2 mixture shall be used in the secondary circuit for the electric plant gas turbine.
2. All parts of the PHTS shall be replaceable.

7.3.3.1 Main Helium Circulator

The following are the required functions of the Main Helium Circulator (MHC):

1. Control the flow of helium to match the heat generation of the reactor core with the heat removal of the PHTS

The MHC shall be designed with the following requirements:

1. The MHC shall be driven by electrical motors capable of rated and variable speeds
2. Active magnetic bearings shall be used to avoid any lubricating product ingress in the primary circuit.
3. Thermal insulation shall be required to protect the internal components by reducing heat migration due to primary system temperatures.
4. The MHC shall be designed with a minimum lifetime of 10 years.
5. The MHC shall be designed with hydraulic characteristics as stable as possible over the required speed range without distinctive reversal points and without pronounced peak.

7.3.3.2 Hot Duct Assembly

The following are the required functions of the Hot Duct Assembly (HDA):

1. Channel high temperature helium from the reactor core outlet plenum to the IHX inlet

The following shall be requirements placed on the HDA:

1. Radial keys shall provide a radial support during operating & seismic conditions.
2. Helium leak tightness at each end (with Core support structure & IHX).

7.3.3.3 Intermediate Heat Exchanger (IHX)

The following are the required functions of the IHXs:

1. Transfer heat from the primary loop to the secondary loop during all normal conditions and between various power levels, and certain accident conditions
2. Separate the primary loop from the secondary loop during all normal and abnormal conditions and during accident conditions for a specified time

The IHX shall be designed with the following requirements:

<u>Type 1 IHX</u>

1-1. The Type 1 IHX shall be designed for a lifetime of [5 years]

1-2. The Type 1 IHX shall be designed with an effectiveness of [≥94%]

Type 2 IHX

2-1. The Type 2 IHXs shall be designed for a lifetime of [20 years]
2-2. The Type 2 IHXs shall be designed with an effectiveness of [≥89 %]

7.3.3.4 Secondary Gas Isolation Valves

The following are the required functions of the Secondary Gas Isolation Valvaes:

1. Provide isolation between the primary and secondary circuits during maintenance or abnormal conditions.

The following requirements are places on the secondary isolation valves:

1. The secondary gas isolation valves shall accommodate a pressure differential of [5 MPa]

7.3.4 Reactor Support Systems

7.3.4.1 Shutdown Cooling System

The following are the required functions of the Shutdown Cooling System (SCS)

1. Transport core residual and decay heat from the Reactor System to the environment when the Reactor System is shutdown and the Primary Heat Transport System (PHTS) is not operational. The helium primary coolant may be pressurized or depressurized.
2. Transport core residual and decay heat from the Reactor System to the environment when the helium primary coolant is depressurized during reactor core refueling operations, during scheduled maintenance of core, vessel and internal components and during certain potential unscheduled maintenance or repair activities.
3. Support cooling of the IHX as needed, and potentially for other components, when the PHTS is not operating.
4. Limit core bypass flow through its components during PHTS operation.
5. Retain helium and radionuclides within the parts of the SCS comprising the primary Helium Pressure Boundary (HPB).
6. Limit the ingress of potential contaminants into the primary helium circuit from components of the SCS external to the primary helium pressure boundary.

The following requirements are placed on the SCS:

1. The SCS shall retain helium and radionuclides within the parts of the SCS comprising the primary Helium Pressure Boundary (HPB).
2. The SCS shall limit the ingress of potential contaminants into the primary helium circuit from components of the SCS external to the primary helium pressure boundary.

7.3.4.2 Reactor Cavity Cooling System

The following are required functions of the Reactor Cavity Cooling System (RCCS):

1. To protect the reactor cavity concrete structure including the support structures of the reactor pressure vessel from overheating during all modes of operation.
2. To provide an alternate means of reactor core heat removal from the Reactor System to the environment when neither the PHTS nor the SCS is available.

The following requirements are placed on the RCCS:

1. The RCCS shall operate continuously and maintain reactor cavity concrete temperatures less than

[90ºC] during normal operations and less than [150ºC]for off-normal events (short term).
2. The RCCS shall be designed to operate through the utility/user duty cycle events for the number of cycles specified [TBD] plus those events and even combinations determined to be required by plant transient analysis.
3. Inaccessible parts of the RCCS shall be designed for an operating life of 60 years.
4. The need for access to individual components during normal plant operation and under accident conditions shall be considered in developing building and component arrangements.
5. The RCCS shall be designed to meet availability/investment protection requirements.
6. The RCCS shall be designed to accommodate continuous operation at any power level up to 100% of rated power.
7. Where cost effective, the design of the RCCS shall incorporate features required to implement on-line surveillance and performance monitoring.
8. The design of the RCCS shall incorporate those features required to accomplish in-service inspection activities within the time and scheduling constraints imposed by the allotted design planned outage time.
9. The RCCS is required to operate continuously in all plant states, including shutdown following loss of forced reactor cooling by the PHTS and SCS with simultaneous loss of pumped circulation of RCCS cooling water and a safe shutdown earthquake.
10. All components and piping of the RCCS shall be designed against seismic loads.
11. All components and piping inside the reactor building including the connections for emergency water supply (fire brigade) are designed against external events, e.g., aircraft crash or pressure waves.

7.3.4.3 Fuel Handling Systems

The following are the required functions of the Fuel Handling System(FHS):

1. Remove and replace fuel from the reactor core
2. Prepare new fuel for use in the reactor core
3. Store spent fuel

The following are the requirements placed on the FHS:

1. During reactor shutdown, the FHS shall receive new and irradiated fuel, reflector blocks, and other core elements from the spent fuel storage system SFSS and place them in the reactor vessel, physically replacing and re-stacking the core.
2. The FHS shall provide shielding to protect workers from radiation during certain fuel handling operations.
3. The FHS shall limit the ingress of potential contaminants into the primary helium circuit from components of the FHS external to the primary helium pressure boundary.
4. The FHS shall be designed to accomplish plant refueling within a time interval specified in planned outage allocations.

7.3.4.4 Spent Fuel Cooling System

7.3.4.5 Nuclear Island Cooling System

7.3.4.6 Helium Service System

The following are the required functions of the Helium Service System:

1. Removal of chemical and particulate contaminants from the primary coolant to maintain specified values

2. Supplying purified helium to systems filled with helium
3. Removal of helium from the primary system and the helium filled supporting systems and storage in a gas store for purified helium
4. Accepting helium from helium filled auxiliary and supporting systems during depressurization activities and, possibly, storing of radioactively contaminated helium
5. Evacuation of primary systems and helium supporting systems

7.3.4.7 Radioactive Waste and Decontamination System

7.3.4.8 Electrical Systems

7.3.4.9 Component Handling System

7.3.5 NHS Protection System

The following are the required functions of the Protection System:

1. Maintain plant parameters within acceptable limits established for design basis accidents

The following are requirements placed on the Protection System:

1. The protection system shall implement the relevant monitoring, analysis, and actuation functions which are necessary to reach the controlled state in case of abnormal events

7.3.6 NHS Control System

7.3.7 NHS Control Room and Operator Interface System

7.3.8 NHS Monitoring System

7.3.9 Startup and Decay Heat Removal System

7.3.10 Other NHS Systems

7.4 Hydrogen Production Plant Requirements

The Hydrogen Production Plant requirements fall outside of the scope of work for this contract; however, the plant breakdown structure is presented as Section Headings in this Section.

7.4.1 H_2 Plant Systems

7.4.1.1 S-I Plant Systems

7.4.1.2 High Temperature electrolysis Plant System

7.4.2 H_2 Plant Protection System

7.4.3 H_2 Plant Control and Instrumentation System

7.4.4 H_2 Plant Fire Protection and suppression System

7.4.5 H_2 Plant Cooling Water System

7.4.6 Product Conditioning, Transport, and Storage System

7.5 High Temperature Heat Transport Loop Requirements

7.5.1 Heat Transport gas Circulating System

7.5.2 Heat Transport Gas Support Services

7.5.3 Pressure Relief System

7.6 Power Conversion System

The following are the required functions of the Power Conversion System (PCS):

1. Convert heat from the PHTS into electricity for distribution on the commercial gird.

The following are the requirements placed on the PCS:

1. The NGNP PCS shall be connected to a local public transmission line for external distribution and sale of [276] MWe.
2. The NGNP PCS shall produce electricity at 60 Hz.
3. The NGNP plant electrical output shall be delivered to the operating utility at the low-voltage bushings of the main power transformer.

7.6.1 Steam Turbine & Generator

The following are the required functions of the steam turbine and generator:

1. Produce electricity using steam generated in the heat recovery steam generator

The following are the requirements placed on the steam turbine and generator:

1. The steam turbine and generator shall be designed for superheated steam at [TBD-pressure] and [TBD-temperature] at the turbine throttle.
2. The steam turbine and generator shall be designed with a high pressure, intermediate pressure, and low pressure turbine on a single shaft.
3. The turbine shall be designed for main steam temperature variations of up to [TBD]
4. The steam turbine shall produce [210 MW] of electricity

7.6.2 Gas Turbine, Compressor & Generator

The following are the required functions of the gas turbine, compressor, and generator:

1. Produce electricity using the secondary gaseous fluid (nitrogen based mixture)

The following are the requirements placed on the gas turbine, compressor, and generator:

1. The turbine shall be designed to operate with the appropriate secondary gas mixture.
2. The turbine shall be designed to operate continuously at a nominal turbine inlet temperature of [850°C]
3. The generator rotors shall be supported by magnetic bearings.

7.6.3 Heat Recovery Steam Generator

The following are the required functions of the heat recovery steam generator (HRSG):

1. Generate steam using the transfer of heat from the secondary gaseous fluid to the main feedwater

The following are the requirements placed on the HRSG:

1. The HRSG shall be a closed loop system.
2. The HRSG shall be designed to a rated thermal output of [500 MW].

7.6.4 Generator Cooling System

7.6.5 Stator/Rotor Carbon Dioxide (CO_2) System

7.6.6 Main Feedwater System

The following are the functions of the main feedwater system:

1. Deliver feedwater to the steam generators at the specified temperature, pressure, flow rate, and water chemistry.
2. Provide storage to accommodate process fluid surge and volume fluctuations.
3. Provide isolation of the feedwater to prevent water inflow to a failed steam generator.

The following are the requirements placed on the main feedwater system:

7.6.7 Main Steam System

The following are the required functions of the main steam system:

1. Convey steam from the steam generator outlet nozzles to the inlet nozzles of the high pressure turbines.

7.6.8 Main Condensate System

7.6.9 PCS Control and Instrumentation System

7.7 Plant Auxiliary Systems

7.7.1 Systems

7.7.2 Cooling Water Systems

7.7.3 Liquid and Gas Supplies

7.7.4 Piping Systems

7.7.5 Electrical Systems

The following are the required functions of the Electrical Systems:

1. Deliver power generated by the plant to the offsite transmission network.
2. Take power from the offsite transmission network for various plant operations, including startup.
3. Provide backup power to select auxiliaries when the plant power units and offsite power are not available.

7.7.5.1 High Voltage Power System

7.7.5.2 Medium Voltage Power System

7.7.5.3 Low Voltage Power System

7.7.5.4 DC/UPS System

The following are the required functions of the DC/UPS System:

1. Provide a stored energy source for the all plant DC loads.

7.7.5.5 Grounding System

The following are the required functions of the Grounding System:

1. Protect personnel and equipment from system faults and lightning strikes.
2. Minimize electrical noise in signal cables

7.7.5.6 Communication & Lighting

The following are the required functions of Communication and Lighting:

1. Provide intra-plant communications.
2. Provide internal and external lighting.

7.7.6 Plant Control Room System

The following are the required functions of the Plant Control Room System:

1. Provide an interface between plant operators and each of the necessary systems within the plant.

7.7.7 Plant Mechanical Services System

7.7.8 Fire Detection and Suppression System

The following are the required functions of the Fire Detection and Suppression System:

1. Rapidly detect and annunciate the presence and location of combustion by-products or the presence of fire within the plant
2. Control and extinguish fires that do occur
3. Provide protection for structures, systems, and components such that the performance of safety functions are not prevented

7.7.9 Communications System

The following are the required functions of the Communications System:

1. Provide plant to offsite communications.

7.7.10 Safeguards and Security System

The following are the functions of the Safeguards and Security System:

1. Provide physical protection of the plant

7.8 Plant Instrumentation & Control (I&C) & Protection

7.8.1 NGNP Plant Supervisory and Control System

7.9 Site and Civil Works

8 References

1. Next Generation Nuclear Plant – Preliminary Project Management Plan, INL/EXT-05-00952 Rev. 1
2. Next Generation Nuclear Plant – High Level Functions and Requirements, INEEL/EXT 03-01163, September 2003

APPENDIX B: SPECIAL STUDIES

The following four special studies that were previously issued by the AREVA NGNP Team to BEA/INL are included herein for completeness.

The original AREVA document identification number is indicated in the parentheses.

Appendix B1 Reactor Type Comparison Study (12-9045308-000) Pages B1-1 through B1-58

Appendix B2 Prototype Power Level Study (12-9045442-001) Pages B2-1 through B2-45

Appendix B3 Power Conversion System Study (38-9049582-000) Pages B3-1 through B3-36

Appendix B4 Primary Secondary Cycle Concept Study (12-9045707-001) Pages B4-1 through B4-66

In conformance with AREVA document procedures, this document's identification and page numbering have been to the header/footer sections of the appended reports. However, the original page numbering has been preserved such that a one-to-one correspondence remains.

PRECONCEPTUAL DESIGN STUDIES REPORT

APPENDIX B1

(Issued Previously as 12-9045308-000)

NGNP with Hydrogen Production

Reactor Type Comparison Study

March 2007

BEA Contract No. 000 60209

AREVA NP Inc.

AREVA

Record of Revisions

Revision	Date	Pages/Sections Changed	Brief Description
000	3/14/2007	None.	Original Issue

Table of Contents

Table of Contents
(continued)

Table of Contents
(continued)

List of Tables

List of Figures

Page

1.0 INTRODUCTION

The Reactor Comparison Type Study is one of the four special studies which the AREVA NGNP Team is performing for INL. This study will compare the pebble bed reactor concept to the prismatic reactor concept as specified in Section 6.3.1 of the Statement of Work [1]. The report identifies the most important discriminating criteria between the two concepts and provides an assessment of the important technical, operational and maintenance differences and the important developmental risks for each.

2.0 STUDY OBJECTIVES

This study provides an answer to the main question posed by the study, namely:

What type of reactor should NGNP be?

The answer depends on which concept best fulfills the NGNP mission [2]; namely, to develop and demonstrate:

- A commercial-scale prototype Generation IV HTR
- Commercial-scale high-efficiency electricity production
- Hydrogen production
- Process heat delivery for industrial applications
- The licensing process with the USNRC and the commercial licensing protocol for future HTR commercialization
- The inherent safety characteristics via special testing of the HTR.
- The economics of the HTR
- New technologies, i.e., high temperature capability, advanced fuel design, advanced energy conversion systems.

The main question can only be answered after appropriate comparisons have been made for each option with respect to the relevant NGNP functions and requirements [2] and a detailed assessment of the key discriminating criteria.

Furthermore, consideration must be given to the future commercialization aspects of the chosen type of reactor. Commercialization of HTR technology is the real NGNP success criterion that can only be measured by the extent of HTR deployment in the decade following NGNP startup and operation.

Industry Acceptance of HTR Technology

Industry acceptance of the chosen reactor concept and the subsequent deployment of a fleet of NGNP-based HTRs is the true measure of success for the NGNP program. The NGNP program plays a key part in the successful commercialization of an advanced reactor concept especially with respect to the developing those essential elements that are critical to success.

The essential elements [3] for the successful commercialization of a new reactor concept are as follows:

Certifiable design

Regulatory certainty

Competitive plant financials

Capital, O&M, cost of product

Risk mitigation & financing (i.e., similar legislation to EPAC of 2005)

Successful demonstration plant (NGNP)

Public acceptance

Predictability

It is beyond the scope of this report to provide a detailed explanation of how each these tenets promote commercialization, nevertheless, it can be safely said that if an advanced reactor concept can successfully satisfy each of the above tenets, it will have a high probability of acceptance in the market place.

These are important considerations to keep in mind as the comparison of key discriminators are presented.

3.0 ASSUMPTIONS

For the purposes of this study, it is assumed that the NGNP plant is a full-sized demonstration plant. This is consistent with the finding of the Power Level Trade Study Report.

Furthermore, this study does not assume a fixed power level; rather, it assumes that each reactor pebble bed or prismatic – has been optimized for its mission at its maximum achievable power level.

Also, the comparison between pebble bed and prismatic options in this study does not assume a given design for each technology. Granted, the pebble bed offering of PBMR and AREVA's prismatic offering (ANTARES) offer good starting points and information, nevertheless, they should be viewed as examples of available HTR technology. Hence, the comparison should be more accurately considered as more a "generic" comparison of pebble and prismatic technology. Furthermore, this assessment is limited to reactor type and confined to the envelope defined by the reactor vessel. Hence, the reactor as viewed herein is considered a "universal" heat source that can be connected to the application of choice.

4.0 ASSESSMENT APPROACH

Figure 1 below shows the general process followed in performing the study. The selection of reactor type will be based on appropriate comparisons made for each option with respect to the relevant NGNP functions and requirements [2] and a detailed assessment of the key discriminators presented in Section 6.0.

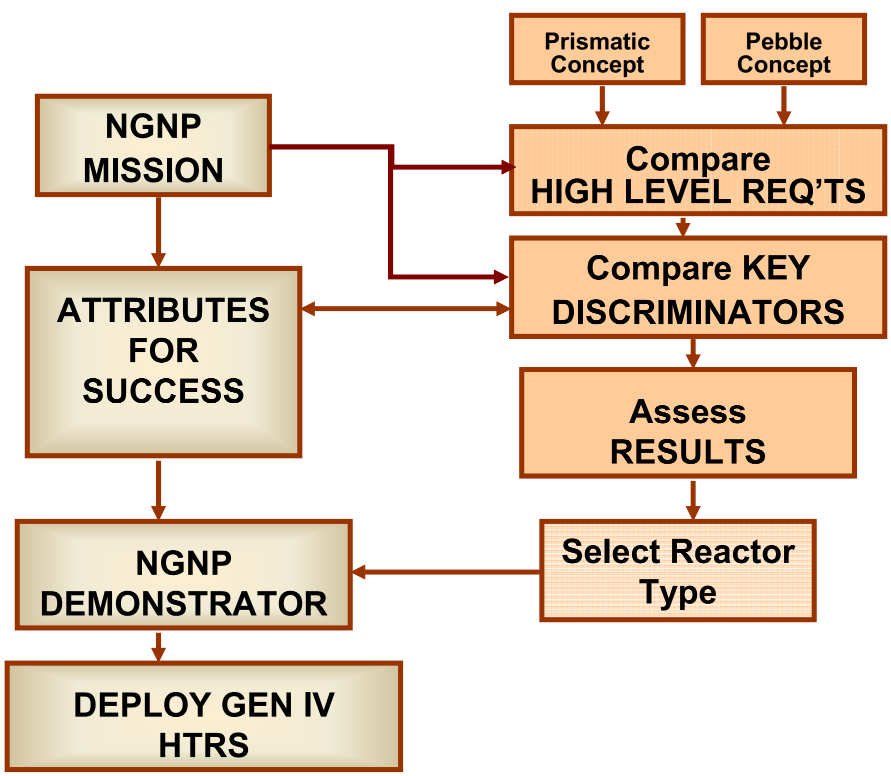
First, the NGNP functions and requirements are reviewed to assess the ability of each option to fulfill the function or requirement. The purpose the review is simply to ensure that there are no "showstopper" functions or requirements at this level for a given option. Also, to determine if there is any advantage or disadvantage that is sufficiently remarkable.

Second, key discriminators are compared for each option. This was done in several steps. First, a list of key, top-level discriminators was compiled via an expert panel. Then, the list was prioritized by a ranking scheme which combined relative discriminator importance and degree of prismatic-pebble difference. This ranking serves only to present the results in order of impact of reactor type choice. Finally, the top-level discriminator list was expanded by adding subordinating items, as appropriate for assessment. The assessments provide the bases for the selection of reactor type.

The following main discriminators (ordered by degree of importance and option difference) serves as the basis for ordering the detailed assessments provided in Section 6.0:

1. Performance Capabilities
2. Fuel Design, Performance, & Development Issues
3. Fuel Handling & Refueling Issues
4. Economics Factors
5. Research & Development
6. Core Design Issues
7. Maintenance issues
8. Operational Considerations
9. Safety & Licensability
10. Level Of Difficulty Of Key Mechanical Hardware Design And Fabrication
11. Schedule Risk
12. Non-Proliferation, Safeguards, SNM Accountability
13. Behavior of reactor systems and fuel during and after key accident conditions
14. Plant Layout and Construction
15. Plant Security

Figure 4-1: Reactor Type Study Process



5.0 NGNP REQUIREMENTS REVIEW

5.1 Top Level Requirements

The following top level requirements are taken from the NGNP High-Level Functions and Requirements Document [2]. The NGNP Functions and Requirements document is technology neutral to the extent specified in Section 3.1.3, "Reactor Type – Prismatic or Pebble Bed" which states:

The reactor shall be either prismatic or pebble bed.

The reason for limiting the choice is that in the Generation IV technology roadmap selection process both the prismatic and pebble bed reactor concepts were highly rated as potential VHTR systems. Both concepts received high scores in economics because of their high efficiency and also in safety and reliability due to the inherent safety features of the fuel and reactor.

In the following matrix, each top level requirement is reviewed with respect to the impact that choice of reactor type may have upon the ability to meet the requirement. If there is a discernable difference arising from the choice of reactor, commentary is provided. Conversely, the requirement is 'Assumed Equivalent' if both options would appear able to meet the requirement, discounting the degree of difficulty (unknown at this time) that may be encountered.

Table 5-1: Top Level NGNP Requirements Review

Top Level NGNP Requirement	Impact of Reactor Type Rating/Comments
1. NGNP prototype shall be designed, constructed, licensed, and operating by 2020 with initial operations in 2018.	Meeting the 2018 schedule date is mission critical. Pebble technology may have a schedule advantage with the potentially earlier (i.e., before 2018) startup of the PBMR prototype in South Africa. 2018 should be achievable with prismatic technology. In either case, the design of critical components (fuel, IHX) will govern.
2. NGNP prototype design configuration shall consider cost and risk profiles to ensure that NGNP establishes a sound foundation for future commercial deployment.[1]	Gas reactor technology represents a paradigm shift for prospective plant owners. Prismatic reactor technology may present less of a paradigm shift than do pebble bed reactors and can achieve a higher power level. There is less perceived risk with concepts that have a high element of familiarity/similarity to current day practice. Public and investor confidence needs to be demonstrated.
3. NGNP prototype shall produce high efficiency electricity and generate hydrogen on a scale that sets a foundation for future commercial deployment.[1]	**Assumed Equivalent** – Both prismatic and pebble bed reactors should be able to meet this requirement.
4. NGNP prototype shall be licensed by the NRC as a commercial cogeneration facility producing electricity and hydrogen.[1]	**Assumed Equivalent** –Both prismatic and pebble bed reactors should be able meet this requirement.

5. NGNP prototype shall include provisions for future testing. [1]	**Assumed Equivalent** –Both prismatic and pebble bed reactors should be able meet this requirement.
6. NGNP prototype shall enable demonstration of energy product and processes utilizing its nuclear heat source. [1]	**Assumed Equivalent** –Both prismatic and pebble bed reactors should be able meet this requirement.
7. The project shall include identification of necessary and sufficient R&D technical scope and priorities. [1]	**Assumed Equivalent** –Both prismatic and pebble bed reactors should be able meet this requirement.

5.2 Plant Level Requirements

The top level plant requirements include restraints based on commercial considerations, customer specific requirements, any design constraints placed on the project, safety considerations and regulatory considerations. The top level requirements are listed below:

Table 5-2: Plant Level NGNP Requirements Review

Top Level Plant Requirement	Impact of Reactor Type Rating/Comments
The electric plant shall demonstrate high efficiency electricity production [>44%]	Prismatic reactors have a significant efficiency advantage over pebble bed reactors because its substantially lower core pressure drop translates into less parasitic power loss.
The plant shall meet the NRC commercial power plant licensing requirements and be licensed for commercial operation supporting creation of regulatory requirements and acceptance criteria for future plant licensing similar to NGNP design.	Both prismatic and pebble bed reactors should be able to meet this requirement; however, the licensing basis for HTR technology has not yet been established. Integrated test and acceptance criteria (ITACC) may lead to differentiation. Prismatic reactors have past licensing history (Ft. St. Vrain, GT-MHR) with the NRC that may be helpful.
Fuel burn-up shall be such that it maximizes uranium utilization, minimizes proliferation risks, minimizes waste streams for open fuel cycle, and optimizes fuel economics.	Prismatic reactors have a 3-4% advantage with respect to fuel utilization (higher burn-up and plant efficiency). Diversion of pebbles may be easier than fuel assembly compacts. Non proliferation fuel surveillance is less difficult with the prismatic core.
The plant shall be designed such that the Nth of a kind plant costs will result in an economically competitive plant.	Nth of a kind cost differences between prismatic and pebble bed reactors will be a key determinant. The prismatic reactor's 50% higher power level translates into a significant economic advantage.
Fuel Qualification	Pebble bed reactors without advanced fuel may have a schedule advantage. PBMR, with its reliance on German particle fuel experience, may be able to qualify its fuel in a shorter time frame as compared to the time needed to qualify an advance fuel type. However, this downside of this strategy is significant should PBMR not be able to

	demonstrate the ability to make fuel commensurate with the quality of the German fuel.
Fuel Performance	A pebble bed reactor with is less limiting fuel service conditions may be easier to demonstrate compliance with requirements. Prismatic reactors with advanced fuel and more limiting service conditions may have more difficulty with compliance.
The electric plant shall feed an outside commercial electrical grid system	**Assumed Equivalent** – Both prismatic and pebble bed reactors should be able meet this requirement.
The hydrogen production plant shall demonstrate commercial scale production and economics	**Assumed Equivalent** – Both prismatic and pebble bed reactors should be able meet this requirement.
The NGNP hydrogen product shall be sent to (TBD) for commercial usage.	**Assumed Equivalent** – Both prismatic and pebble bed reactors should be able meet this requirement.
No provisions shall be made for onsite produced hydrogen storage.	**Assumed Equivalent** – Both prismatic and pebble bed reactors should be able meet this requirement.
The NGNP shall be designed in accordance with defense-in-depth principles and philosophy with the intent to eliminate the need for evacuation beyond the plant site boundary of 450 meters.	**Assumed Equivalent** – Both prismatic and pebble bed reactors should be able meet this requirement.
The nuclear heat source shall use TRISO-coated particle fuel	**Assumed Equivalent** – Both prismatic and pebble bed reactors should be able meet this requirement.
The reactor shall be graphite moderated.	**Assumed Equivalent** – Both prismatic and pebble bed reactors should be able meet this requirement.
The nuclear heat source shall use once through uranium fuel cycle with uranium enrichment less than 20% 235U.	**Assumed Equivalent** –Both prismatic and pebble bed reactors should be able meet this requirement.
The primary nuclear heat source shall utilize an indirect cycle heat transport system (although a study will be performed addressing alternatives).	**Assumed Equivalent** –Both prismatic and pebble bed reactors should be able meet this requirement.
Hydrogen production plant will requires high temperature (~ 900°C range) process heat from the nuclear plant.	**Assumed Equivalent** –Both prismatic and pebble bed reactors should be able meet this requirement.
The plant shall be designed to the USA industrial Codes and Standards as necessary to meet the USA industrial and regulatory requirements.	**Assumed Equivalent** –Both prismatic and pebble bed reactors should be able meet this requirement.
The nuclear heat source shall be inherently safe and passively cooled in case of loss of all off-site and on-site motive power.	**Assumed Equivalent** –Both prismatic and pebble bed reactors should be able meet this requirement.
The plant design shall result in ease of	**Assumed Equivalent** – Both prismatic and pebble bed

required maintenance activities with appropriate considerations for layout space, access, and maintenance equipment.	reactors should be able meet this requirement.
The nuclear plant refueling outage shall allow 18 months operation between refuelings.	**Assumed Equivalent** – The prismatic reactor should be able meet this requirement. The requirement is not applicable to the pebble bed reactor with on-line refueling.
The nuclear plant shall be designed for 60 year life and with fifteen year onsite used fuel storage capacity within the plant boundary fence.	**Assumed Equivalent** – Both prismatic and pebble bed reactors should be able meet this requirement.
The nuclear plant shall be designed for an availability of greater than 90%.	**Assumed Equivalent** – Both prismatic and pebble bed reactors should be able meet this requirement.
Public safety due to any accident at the nuclear plant shall not depend on public or personnel beyond site boundary.	**Assumed Equivalent** –Both prismatic and pebble bed reactors should be able meet this requirement.
The nuclear plant design bases accidents shall not lead to nuclear plant write-off.	**Assumed Equivalent** –Both prismatic and pebble bed reactors should be able meet this requirement.
The plant shall be located at the NPR site at the Idaho National Laboratory, Idaho USA, approximately 3.5 miles northeast from the INTEC facilities.	**Assumed Equivalent** – Both prismatic and pebble bed reactors should be able meet this requirement.
The reactor shall be built to commercial scale with a power level consistent with passive safety features.	**Assumed Equivalent** – Both prismatic and pebble bed reactors should be able meet this requirement.
The NGNP shall have adequate passive safety systems to cool the core down from full power to safe shutdown mode and limit the fuel temperatures under accident conditions to levels consistent with fuel performance requirements.	**Assumed Equivalent** – Both prismatic and pebble bed reactors should be able meet this requirement.

5.3 Top /Plant Level Requirements Review Summary

Based upon the above discussion, most high level requirements are neutral (i.e., assumed equivalent) with respect to “reactor type.” For those requirements that are remarkable, the review reveals no real “show stoppers,” i.e., a requirement that would be impossible to meet. These are simply requirements for which one type of reactor has a greater level of difficulty meeting the requirement versus the other. Nevertheless, to summarize, the pebble reactor may have an edge in schedule but the prismatic reactor will always have an economic advantage due to its higher power capability and the added power is needed and can be used.

6.0 ASSESSMENTS

This section presents assessment of the key discriminator topics listed in Section 4. Each discriminator topic area has been further subdivided into the key parameters comprising the discriminator topic area. These parameters are then evaluated with respect to reactor type and the results summarized in a table for each discriminator topic area. A simple rating scheme is applied to the comparison results. For each parameter compared; a simple score is assigned depending whether the reactor type displays a weal advantage (+), a moderate advantage (++) or a strong advantage (+++) or not advantage over the other (o). Some parameters are listed for information. The bases for the evaluations are discussed following each table.

While each performance parameter is viewed independently, the reader must be aware of the strong interdependence between certain parameters. Furthermore, it assumed for comparison purposes, that each option is designed to produce the maximum power level that can be supported by the design and still achieve inherent safety goals.

As mentioned in Section 4, the discriminators topic areas have been prioritized and are presented in that order in the following sub-sections.

6.1 Performance Capabilities

The performance capabilities of the pebble bed reactor and prismatic reactor are compared in this section as shown in Table 6-1.

Table 6-1: Performance Capability Comparison

PERFORMANCE PARAMETER	Prismatic Reactor	Prismatic Rating	Pebble Reactor	Pebble Rating
Power Level ($MWT_{thermal}$)	600	+ + +	400	-
Electric Output (MWe)	284	+ + +	165	-
Modules per 1200 MWe site	4.2	+ + +	7.3	-
Capacity Factor, %	92	-	95	+ +
Efficiency	>44	+ +	42	-
Core Outlet Temperature, ºC	900	-	950	+ +
Core Pressure Drop	low	+ + +	high	-
Fuel Enrichment, %	14	-	9.6	+
Fuel Burnup, GWD/MTU	120	+ +	91	-
Uranium Requirements, Tonnes/Gwe-Year	192	o	198	o
Plant Lifetime/License, Years	60/60	o	60/40	o
Overall Performance Capability Rating		**+++**		

6.1.1 Power Level

The achievable thermal power in the prismatic core is about 50% greater than the thermal power achievable in the pebble bed core. The main reason for this is geometry. Because pebble cores have a lower void fraction (19% versus 40%), the prismatic core can achieve a higher power rating for a given gross core volume while still retaining the passive safety attributes. Hence, 81% of the gross core volume has fuel in the prismatic case while only 60% of the pebble bed gross core volume contains fuel.

The key to the prismatic reactor's power level advantage lies in its performance during the limiting design basis event (i.e., depressurized conduction cool down). The additional mass in the prismatic core behaves as a heat sink for decay heat energy that serves to buffer the internal core and fuel temperature rise. Hence, for a fixed maximum post-accident fuel temperature (i.e., 1600 ºC) limit, the prismatic core is able to meet this requirement at a higher power level.

Hence, as the designs have evolved over the past several years, the maximum achievable core power is approximately 600 MWth for the prismatic core versus 400 MWth for the pebble bed core. Clearly, the higher power capability of the prismatic core is a strong advantage that the option possesses over the pebble bed option and is rated accordingly.

6.1.2 Electrical Output and Modularity

Both reactor types can be matched with similar energy conversion systems, whether it be for electricity generation or process heat. The higher power capability for the prismatic reactor simply translates into more useable power. Furthermore, when considering a multi-unit site, the prismatic reactor has almost a 3:2 advantage in terms of the number of unit required to supply the total plant output.

6.1.3 Capacity Factor

Capacity factor is the ratio of the actual energy produced by the facility to the maximum amount of energy that the facility could have produced in a given period.

With on-line refueling, the pebble bed reactor can potentially achieve a nominally higher capacity factor than the prismatic option which must shut down periodically for refueling. PBMR claims a 95% capacity factor is achievable whereas this prismatic vendor (i.e., AREVA) supports a 92% capacity factor which includes a 22-day refueling every 18 months.

The theoretical advantages of online refueling have not been realized over the long term in other commercial designs. AREVA believes that this will be the pebble bed reactor experience as well. Additionally, the PBMR pebble bed design supports an outage every 5-years. Conventional wisdom questions the ability to continuously operate mechanical equipment for such a lengthy period without maintenance and maintenance requires downtime (preferably scheduled downtime). It also could promote a "run-to-failure" plant philosophy that is detrimental to overall plant safety. This, in turn, could result in a higher forced-outage rate for the pebble bed option, negating any benefit from continuous operation with online refueling

The potentially higher capacity factor of the pebble bed reactor with on line refueling is not a strong advantage of this reactor type over the capacity factor of the prismatic reactor.

6.1.4 Plant Efficiency

Plant efficiency is simply the ratio of useful energy produced to the amount of energy used by a power generation facility. For the case of electricity production only, the plant efficiency the net electrical output produced divided by the plants thermal power rating. Furthermore, plant efficiency in the case of electrical production is strongly dependent on the choice of power production cycle (Brayton, Rankine, combinations etc.).

Regardless of the cycle match to the plant, the prismatic core offers a significant advantage in achievable plant efficiency (>44% versus 42% for pebble bed using comparable Brayton cycles). The main reason for this is again core geometry. In this case, the pressure drop across the prismatic core is a factor of 2 to 3 less than the pressure drop across the pebble bed core. This translates into significantly less power being consumed by the prime mover for primary coolant flow (i.e., helium circulators in an indirect cycle or compressors in a direct cycle). Hence, more net power is available in the prismatic case.

The higher achievable plant efficiency of the prismatic core reactor is a strong advantage over the pebble bed core. Note also that the current plant efficiency for PBMR (42%) is slightly less than the plant efficiency requirement specified in the NGNP functions and requirements document [2].

6.1.5 Maximum Core Outlet Temperature

The maximum core outlet temperature that can be achieved by either option is a function of the temperature difference between the fuel and the coolant and the maximum allowable post-accident fuel temperature (approximately 1600 ºC). In the pebble bed reactor, the average fuel particle-to-coolant temperature drop is 50-70 ºC as opposed to 150-200 ºC in the prismatic design. Several phenomena are responsible for this, primarily, better core flow mixing. This translates into an advantage for the pebble bed design because the core outlet coolant temperature can be higher for the same maximum fuel temperature. Conversely, for the same coolant outlet temperature, the pebble bed design has more margins.

The pebble bed reactor type does have an advantage of having a slightly higher temperature capability than the prismatic reactor.

6.1.6 Core Pressure Drop

The core pressure drop can significantly affect the power required to either compress or circulate primary helium. The higher the core pressure drop the more energy is required to circulate a given amount of helium flow. The pebble bed reactor is significantly disadvantaged in this regard because the torturous flow path helium must follow through the pebbled bed. This results in a relatively high core pressure differential as compared to the prismatic core. For, example, the pressure drop in the PBMR core demands approximately 40-60 MWe (estimated) to circulate primary helium as opposed to only 15 MWe for the ANTARES prismatic reactor. Note that while these are electric power requirements, a similar ratio of compressor power requirements is expected in the case of a direct power conversion system.

Hence, the prismatic reactor is judged to have a significant advantage due to its relatively low core pressure drop and the attendant power savings that this realizes.

6.1.7 Fuel Enrichment

Because of continuous refueling, the pebble bed core requires only a nominal amount of excess reactivity above that necessary to maintain criticality at power. As a result, the fuel enrichment requirement for pebble fuel is approximately 8-9%. Conversely, the prismatic core requires additional fuel material to remain critical through out its 18 month cycle. This translates into an enrichment requirement of approximately 14-16%. The excess reactivity that the additional fuel represents is readily offset by the use of burnable poisons.

The pebble bed fuel, because of the lower enrichment requirement, will therefore bear a cost advantage over the prismatic reactor for this element of fuel costs. However, on an overall basis, this is not a strong advantage considering that HTR fuel costs are estimated to be about 26% [4] of the overall plant production costs and that a significant portion of the cost of particle fuel is in its fabrication.

6.1.8 Fuel Burnup

The burnup capability of the TRISO particle fuel is independent of the reactor type but is wholly dependent on the fuel qualification program and the conditions at which it is qualified.

For PBMR, the fuel has a target burnup of approximately 90 GWD/MTU, this translates into about 9 overall core passes. For the prismatic core, the fuel must achieve an average burnup of 120 -140 GWD/Mtu in order to achieve an 18-month cycle length (36-month residence time for each element). Should prismatic fuel be successfully qualified to this burnup level, it will clearly be an advantage over pebble fuel performance because of the higher fuel utilization. Attaining similar burnup performance with pebble fuel may be possible but the additional fissile material (i.e., more enrichment) required to take advantage of the higher burnup capability will have to be accommodated in the core design. Conversely, should prismatic fuel not achieve target burnups, the result is a reduced cycle length and should pebble fuel suffer the same, pebbles will simply be passed through the core fewer times; however, this will result in using more fuel.

The prismatic core is seen to have an advantage over the pebble bed core due to its higher burnup capability. This advantage results in better fuel utilization.

6.1.9 Uranium Requirements

Plant uranium requirements are function of initial fuel loading of U-235 (i.e., enrichment), power level, and burnup capability. How efficiently the uranium is used is also a function of the plant efficiency. Despite the significant difference in initial enrichment between the pebble reactor and the prismatic reactor (8% versus 14%), natural uranium requirements on a unit energy basis for each option are approximately the same. Uranium consumption is estimated at 192 T/GWe-Year the prismatic reactor versus 198 T/GWe-Year for the pebble bed reactor assuming published data for the each option.

As approximations, the above uranium requirements are judged to be roughly equal; hence, neither the prismatic reactor nor the pebble reactor has a clear advantage over the other with respect to natural uranium requirements.

6.1.10 Plant Lifetime

Based on the Atomic Energy Act, the Nuclear Regulatory Commission (NRC) issues licenses for commercial power reactors to operate for up to 40 years and allows these licenses to be renewed for up to another 20 years. A 40-year license term was selected based on economic and antitrust considerations, not technical limitations (source - NRC website).

Currently, PBMR is designing for 40-year plant lifetime whereas ANTARES is being designed for a 60-year plant life for prismatic. These choices are arbitrary. There is no reason you could not design a pebble reactor for 60 years. PBMR simply made a design decision to go with 40 years. Clearly, the 60-year lifetime and license of the prismatic reactor is advantageous; however, it is unlikely that the NRC will license a reactor for 60 years. Nevertheless, by designing for 60 years, the life extension process is simplified.

6.2 Fuel Design and Fuel Performance Issues

The safety case for the HTR relies heavily on TRISO-coated particle fuel technology with its high temperature capability and high fission product retention capability. Both the pebble bed and prismatic reactors rely on this technology. The similarity in required performance and reliance on fuel to hold together during accidents is effectively identical; however, the fuel development strategy taken by PBMR to demonstrate required performance is very different from the strategy taken for ANTARES. The extent to which fuel strategy differences matter will depend on one's perspective with respect to the NGNP mission relative to fuel development

6.2.1 Fuel Service Conditions

In this section, the fuel service conditions of the pebble bed reactor as represented by PBMR and the prismatic reactor as represented by the potential baseline design for NGNP (i.e., AFCI program) are reviewed. A comparison of fuel service conditions is presented in Table 6-2 below. These data were compiled from various publicly available sources (e.g., material from reference[5]).

Table 6-2: Fuel Service Condition Comparison

Fuel Service Conditions	Prismatic Reactor	Prismatic Rating	Pebble Reactor	+ or -
Service Conditions - Normal Operations:				
Avg. Fuel-Coolant Temp. Difference, ºC	100-200	-	50-70	**++**
Avg. Fuel Temperature, ºC	1250	**+**	1100	-
Fluence, 10^{25} n/m^2	4.7	**++**	3.5	-
Burnup, % FIMA	15	**++**	10	-
Power Density, W/cc	6.6	**+++**	4.7	-
Packing Fraction	30	+	10	-
Operational Fuel Performance Target, failure rate	1.00E-05	o	1.00E-05	o
Service Conditions - Post Accident	<1600 ºC	o	<1600 ºC	o
Overall Fuel Service Conditions Rating		**++**		

6.2.1.1 Fuel Service Conditions – Normal Operation

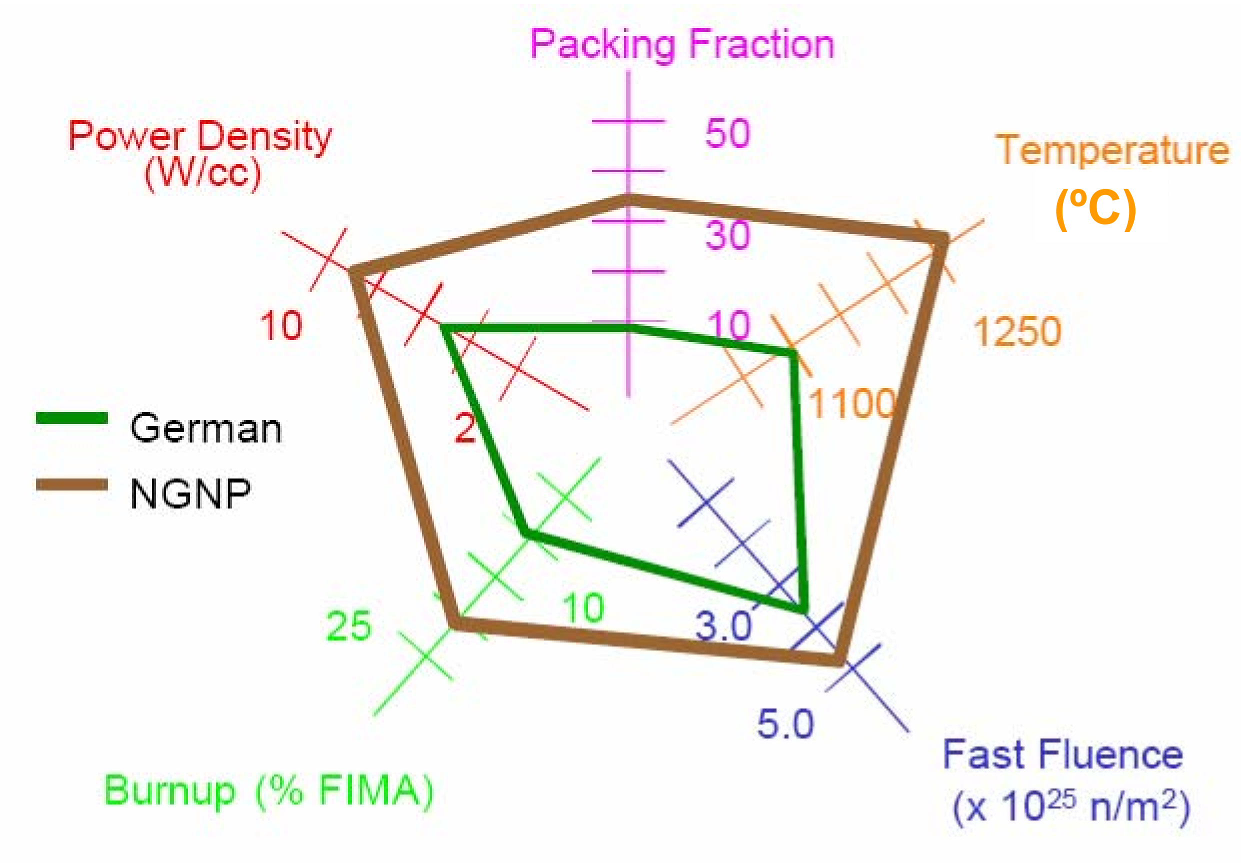
As shown in the preceding table, the target fuel service conditions found in the pebble bed reactor design are less challenging than those found in the prismatic design. Note, however, both designs must meet similar fuel performance targets.

First, the pebble reactor does benefit from a lower average fuel temperature during operation because conditions support a much lower fuel-to-coolant temperature difference. This is attributed to lower core power density, more heat transfer surface area and, perhaps more importantly, a significantly higher degree of coolant mixing in the pebble core. This is obviously an advantage. The lower average fuel temperature is also below the silver "cliff" region with respect to silver release. The other parameters (fluence, burnup, power density, packing fraction) are, to some extent, variables that can be controlled through design as dictated by fuel qualification results.

6.2.1.2 Fuel Development Strategy

However, the fuel development strategy being followed by PBMR relies heavily on its fuel operating near or within the German fuel operating envelope (inner green circle) as shown below in Figure 6-1 (from reference [5]). Because PBMR is adapting the German fuel design, PBMR may be able to credit pas German fuel qualification work in licensing and qualification, thereby reducing the scope of fuel qualification. It is not all clear, however, that this strategy will ultimately be successful. Reliance on this strategy presents a significant schedule risk given the uncertainty in regulatory response to the issue. In contrast, the fuel operating envelope proposed for ANTARES and, potentially NGNP, is similar to that of the AGR program (outer, brown circle) which is very aggressive. Hence, the qualification and licensing of ANTARES fuel must meet the demands of a much more challenging design envelope.

Figure 6-1: Normal Operation Fuel Service Conditions: AGR versus German Fuel Development Programs



The service conditions for pebble fuel - lower average operating temperature, burnup, power density etc. - may translate into less licensing risk and research and development risk. The reverse is true for the advanced fuel design. However, the PBMR fuel strategy is wholly dependent on their being able to demonstrate the applicability of the German data set to their current or to be developed fuel process. Should this demonstration not succeed, PBMR will be faced with a fuel development challenge similar to that of the advanced fuel design but at a much later stage in their overall plant development after a significant amount of investment.

Regarding the advanced fuel strategy, there is significant benefit to be gained by developing an advanced particle fuel capable of meeting the more stringent requirements represented by the outer envelope above. The higher burnup results in greater fuel efficiency and longer fuel cycles while the higher temperature capability permits extending the range of applications to which the plant can be marketed. Hence, the advanced fuel design endorsed by ANTARES is clearly an advantage and, moreover, is fully compatible with the objective of the NGNP program to develop advanced technology.

Note that in the foregoing discussion, the focus is fuel development strategy which is independent of reactor type. There is no reason to believe that fuel for a prismatic reactor could not be designed to stay within the German fuel envelope. Obviously, that would have operational implications because of the lower burnup. Conversely, there is nothing that would prevent PBMR from expanding its operating envelope to that of the AGR program as well; however, this would not be in line with PBMR's current low risk fuel qualification strategy. The point is that either fuel stratagem has its advantages and disadvantages but, albeit with some difficulty, fuel strategy decisions are reversible. New fuel strategies can be introduced as newer designs evolve or, more importantly, as chosen designs lose favor due to unfavorable operational experience.

6.2.1.3 Fuel Operational Performance

Actual fuel performance during operation must be consistent with established fuel performance limits which include:

1. As-manufactured quality requirements:
    a. Allowable failure of fuel particle coatings at the time of manufacture,
    b. Free uranium contamination in fabricated fuel.
2. In-service fuel performances requirements:
    a. Fission product retention capabilities during normal operation (accounting for the failure of fuel particle coatings and, if significant, for the radionuclide diffusion out of the fuel particles).
    b. Fission product retention capabilities during off-normal events (accounting for any incremental failure of fuel particle coatings and, for any increased diffusion of radionuclides out of the fuel particles).

Currently, the operational fuel performance target for both reactor types is the same: 1.00E-05 or 1 failure for each 100,000 particles.

The actual limits are established to meet acceptable dose consequences for normal operation and as a result of a design basis accident. Though the acceptable dose consequence is the same for both reactor types, the resulting fuel performance limits may be different for a number of reasons:

- Accident severity/duration
- Core radionuclide inventory (source term)
- Fission product retention capability (fuel, filtering)
- FP retention within primary system (plateout)
- Degree of Graphite Oxidation
- Release of non-fuel radio-contaminants (e.g., dust)
- Distance to site boundary

Hence, given the above, it is premature to acknowledge one reactor type as having a significant advantage over the other with respect to operational fuel performance. There a too many variables that can be adjusted to achieve acceptable results. However, the fact that the pebble bed reactor is more prone to creating dust that could potentially be released is remarkable. Furthermore, should the pebble bed reactor be coupled to a direct cycle, dust transport into sensitive turbo-machinery and heat exchange equipment could be problematic.

Also remarkable is the fact that the prismatic core is more resistant to graphite oxidation than the pebble bed core. Due to the presence of the fuel kernels and to assure the integrity of the SiC coating, neither the pebble fuel spheres nor the fuel compacts can be fully graphitized. However, in the prismatic core, the fuel compacts are completely sealed in a fully graphitized matrix. Hence, the result is significantly less graphite oxidation in post-accident heat-up scenarios.

6.2.1.4 Fuel Service Conditions – Post-Accident

It is not clear if either reactor type option has an advantage with respect to post accident fuel performance. In the pebble bed case, the more favorable normal operational environment would challenge the fuel less because fuel performance is heavily dependent on burn-up and temperature resulting in the fuel being in "better" condition prior to an operational event. However, the prismatic fuel, even running at a higher temperature and burn-up, may perform better post accident (i.e., less release) due to its quality being higher as a result of having to meet a more stringent qualification process due to its more aggressive operational envelope. In either case, the maximum post accident fuel temperature will need to be maintained at or below the maximum allowed post accident temperature of 1600ºC, dependent on the fraction of fuel experiencing this temperature. Under the assumption that this fraction is very low (<1%), even a potentially higher accident temperature may be justified.

6.2.2 Fuel Qualification and Fabrication

Fuel qualification and fabrication issues are summarized in Table 6-3 below and discussed.

Table 6-3: Fuel Qualification and Fabrication

Fuel Qualification & Fabrication	Prismatic Reactor	Prismatic Rating	Pebble Reactor	+ or -
Fuel Qualification Program	Harder	-	Easier	++
Fuel Fabrication				
Fuel Quality Requirements	Higher	-	Lower	++
Unitized Material Burden				
Natural Uranium (Tonnes/GWe-Year)	192	o	198	o
*Graphite (Tonnes/GWe-Year)	156	++	218	-
Process Complexity	Similar	o	Similar	o
TRISO Particles/GWe-Year	$1.2x10^{10}$	++	$1.6x10^{10}$	-
Fuel Compacts/GWe-Year	$3.9x10^{6}$	o	N/A	
Fuel Elements/Gwe-Year (prisms or pebbles)	1396	o	1,084,717	o
Fabrication Cost	TBD	o	TBD	O
Overall Fuel Qualification & Fabrication Rating		**o**		**O**

*Includes fuel block and compact graphite for prismatic fuel and graphite portion of pebble fuel.

6.2.2.1 Fuel Qualification Program

Due to adhering to the German fuel service conditions as shown above, PBMR fuel qualification appears to be less challenging than that faced by ANTARES. By effectively replicating the German fuel manufacturing process, PBMR may effectively reduce the scope of fuel qualification, relying heavily on the applicability of existing data. Furthermore, PBMR fuel program relies on a step-wise burn-up escalation program whereby burn-up levels are increased only after satisfactory fuel performance has been demonstrated in-reactor. This permits PBMR to introduce fuel improvements and higher burn-up gradually in a fashion similar to that seen in commercial LWRs.

Conversely, ANTARES fuel qualification is more challenging than that faced by PBMR because its fuel service conditions significantly exceed the German fuel service conditions for the major parameters. Hence, the scope of fuel qualification is greater because of the research and development that is involved (i.e., significant fuel irradiation program is required to obtain the data necessary to demonstrate the desired level of fuel performance.

Based on the above, it currently appears that pebble bed fuel development presents less risk than prismatic fuel development. This less risk, however, comes at a price – limited fuel performance capability. By adopting 25-year old, UO_2 based German particle fuel technology; PBMR is sacrificing the prospective benefits of an advanced fuel design for less licensing risk and less fuel R&D.

It remains to be seen if the PBMR fuel development strategy will work. The excellent performance of German particle fuel is well known; however, the reasons why it performed so well are not fully understood. Nevertheless, the PBMR strategy hinges on replicating the German fuel fabrication to maximum extent possible and then demonstrating similar fuel performance. This allows PBMR to present a reasonable licensing case based on past German fuel development and at the same time significantly reduce R&D requirements. PBMR is confident that this approach will be successful; and, is willing to accept the risk that failing to demonstrate fuel performance similar to German fuel represents.

In comparison, ANTARES, by virtue of its more severe fuel service conditions, must pursue an advanced fuel design to accomplish its mission. ANTARES is not limited to UO2 based fuel and is highly likely to adopt UCO or another advanced fuel type in order to achieve satisfactory performance levels. This aspect of ANTARES and prismatic fuel development is more fully in line with NGNP's mission to develop and demonstrate new technologies.

6.2.2.2 Fuel Fabrication

Given that both pebble bed and prismatic reactor fuel is based on TRISO-particle technology, the main question to be answered with respect to fuel fabrication is this: "Is the manufacturing burden and cost greater for one option than the other?"

First, fuel quality requirements for the prismatic fuel will be more restrictive than for pebble reactor fuel because the prismatic fuel must meet similar operational and post-accident failure targets but while being qualified to the more severe service conditions. This will add to fuel costs.

With respect to natural uranium usage, ANTARES requires 192 Tonnes/GWe-year versus PBMR requiring 198 Tonnes/GWe-year. Or, on an annual basis, ANTARES requires 54 Tonnes/year versus 33 Tonnes/year for PBMR. ANTARES requirement is higher due to its higher power capability.

Graphite requirements for fuel manufacturing must also be considered. PBMR requires approximately 218 Tonnes/GWe-year versus 156 Tonnes/GWe-year that ANTARES requires. However, on a yearly basis, again due to its higher power output capability, ANTARES requires 44 Tonnes/year versus PBMR's 36 Tonnes per year. Clearly, the lower graphite requirement for the prismatic reactor is an advantage.

The process to make particle fuel for either the pebble or prismatic reactor is of comparable complexity. The basic process is based on particle formation by the sol-gel process followed by the application of the successive particle layers via the chemical vapor deposition process. Obviously, the procedures and technologies involved are fundamental elements, including 'art-of-the-trade' considerations. Once particles are fabricated, they then must be fabricated into pebbles or compacts and prisms.

Based on advertised conditions [9.6% enrichment and 90.7 GWD/MTU], PBMR requires an estimated 1,100,000 pebbles per gigawatt-year of electrical energy output or 179,000 pebbles per year (490 pebbles/day). On a particle basis, PBMR requires about 16 Billion particles per gigawatt-year of electric energy output or about 3 billion particles per year. Conversely, ANTARES [14% enrichment, 120 GWD/MTU] requires 1397 prismatic fuel elements per gigawatt-year of electrical output or about 400 prismatic fuel elements per year. On a particle basis, this translates into 12 billion particles/GWe-year or 3 billion particles per year. Hence, the prismatic reactor produces approximately 50% more energy per particle than the pebble reactor.

From the perspective of manufacturing burden, ANTARES and PBMR have remarkably similar annual material flows – each must process relatively the same amount of fuel particles and process similar amounts of graphite. Hence, any real differences must arise in the fabrication process. Because of its more demanding service condition envelope, ANTARES fuel will be required to meet more stringent quality requirements and, therefore, will be subject tighter fabrication controls. However, that is not to say PBMR fuel will be subject to lesser fabrication controls. It does mean that the ANTARES fuel qualification program requires fuel that performs in its service envelope to the required quality level.

In summary, it is premature at this point to make a judgment on fuel fabrication cost. The fuel for the prismatic reactor has an advantage due to requiring less material; however, it is not clear, despite requiring 33% less particles that the costs associated with particles, compacts and prisms will be less than the cost of particles and pebbles. More detailed data is required to assess fuel cost; however, the availability of such data is limited due to its proprietary nature.

6.2.2.3 Fuel Design Options

In the foregoing discussion, we have focused on PBMR and ANTARES fuel programs as they are currently known; hence, it must be viewed in "snapshot" fashion. Furthermore, one needs to acknowledge the many degrees of freedom that are possible in fuel and reactor design. Thus, caution is advised when extrapolating comparison results of ANTARES and PBMR on the fuel utilization question.

Assuming that a pebble bed reactor uses UO_2 and only has a burnup of 60% of the prismatic (possibly using UCO or other advanced fuel), then clearly the pebble bed reactor will require 70% more particles for the same energy output. However, it is not a requirement that pebble bed reactors be limited to UO_2. Pebble bed reactors are flexible in that they can operate with lower particle burnups, but they are not required to. With a successful UCO fuel qualification program, there is no reason they could not go to higher enrichments and run their pebbles through 12-15 times instead of 8-10. This would take them to burnup levels similar to that of the prismatic reactor fuel.

With online refueling, the pebble bed reactors have less need to pursue higher burnup, but there is no reason that they could not do it. PBMR has opted to build upon the German experience with respect to fuel qualification, but they could go farther if the value of fuel utilization and spent fuel charges became significant enough to justify UCO qualification.

6.2.3 Waste Disposal and Reprocessing

Table 6-4 below compares the back end material flow for ANTARES and PBMR, respectively. These data have been assembled based on published information.

Table 6-4: Spent Fuel Disposal and Reprocessing

SPENT FUEL DISPOSAL- REPROCESSING	Prismatic Reactor	Prismatic Rating	Pebble Reactor	Pebble Rating
Spent Fuel Disposal				
Fuel Units/Gwe-Year compacts (prisms) or pebbles)	4.265×10^6(1396)	o	1.085×10^6	o
Fuel Unit Volume, M^3/GWe-Year	124	o	123	o
Fuel Unit Stored Volume, M^3/GWe-Year	124	++	204	-
Heavy Metal Waste, Tonnes/Gwe-Year	6.4	+	9.8	-
Non-Heavy Metal Waste, Tonnes/Gwe-Year	156	+	218	-
High Level Waste, M^3/Gwe-Year	26	++	204	-
High Level Waste, Tonnes/Gwe-Year	53	++	228	-
Non-Compact Graphite, Tonnes/Gwe-Year	109	o	Na	o
Residual Uranium-235 content, %	~5		<1	+
Reprocessing Considerations	Easier	+	Harder	-
Overall Rating for Spent Fuel Dipsosal & Reprocessing		++		

6.2.3.1 Spent Fuel Disposal

The prismatic reactor has a significant advantage over the pebble bed reactor with respect to spent fuel:

1. First, as seen in the preceding table, the discharged volume of spent fuel is, remarkable, almost identical. However, the similarity ends when considering stored volume – pebble fuel, due to packing efficiency, requires nearly twice its volume for storage.

2. Second, the actual amount of heavy metal waste and fission products is less in the prismatic case than for the pebble case due better fuel utilization.

3. Third, the non-heavy metal portion (i.e., fuel block graphite plus compact graphite) of the waste is also less (156 vs 218). Furthermore, about 1/3 (47 tonnes) of the non-heavy metal waste by weight for the prismatic fuel is compact graphite. Conversely, 100% (i.e., 218) of the value for the pebble reactor represents the graphite portion of the pebble sphere.

4. Assuming the compacts are separated from the bulk fuel block, and amount combined with the heavy metal waste yields 53 tonnes/GWe-Year of high level waste. This is a significant advantage when compared to the pebble waste of 228 tonnes/GWe-year (which is simply the spent fuel pebble mass)

Whether or not compacts are separated from the prismatic blocks, the above data show the prismatic advantage with respect to spent fuel management. It is very important to know that for prismatic designs spent fuel waste can be segregated and dispositioned via to most economical path.

6.2.3.2 Graphite Waste

Finally, a word about fuel-related graphite waste is in order. The balance of the fuel-related graphite waste or approximately 110 Tonnes/GWe-Year is attributed to the prismatic block. Because the fuel compacts can be removed, this graphite could most likely be disposed of as low level waste; however, its radioactivity content may challenge Class C limits. Because it is unlikely to be performed on-site, the benefit of compact removal in terms of high level waste volume reduction will not be realized until a suitable infrastructure exists that will support the required processing. This infrastructure, of course, will require a sufficient population of reactors to support its viability.

Graphite lifetime is an essential consideration and there is a significant amount of graphite in the core structures, and reflectors. In the prismatic design, reflectors are routinely replaced as part of the refueling process. Current thinking is that a replacement frequency on the order of 6-years will be appropriate for reflectors directly adjacent to fuel.

In a pebble bed design, there is not a convenient opportunity to replace the massive central fixed reflector. It is highly likely that the central reflector will have to be replaced at least once. A dedicated or extended scheduled outage will be required to replace it.

There is considerable amount of uncertainty in core structure and reflector graphite lifetime material requirements. AREVA estimates that ANTARES will require approximately 124 Tonnes/GWe-year of graphite over its lifetime. PBMR appears to have a slight advantage here in that it requires approximately 103 Tonnes/GWe-year of graphite over its life. (Note that these are rough estimates and that actual graphite lifetimes need to be determined through appropriate material qualification programs.)

6.2.3.3 Waste Storage / Reprocessing

Spent fuel (i.e., fuel + moderator graphite) and reflector graphite comprise the majority waste flow for either the pebble or the prismatic option. ANTARES is estimated to produce 162 Tonnes of spent fuel elements per GWe-Year versus an estimated 228 Tonnes of spent pebbles/GWe-year for PBMR. This is clearly an advantage for ANTARES. On an annual basis, the values are 46 Tonnes and 38 tonnes, respectively for ANTARES and PBMR, acknowledging ANTARE/s higher power level as the reason for the higher annual material flow.

From an order of magnitude perspective, these numbers are similar and, even though one may be higher or lower, these differences would not, by themselves, be significantly influential with respect to reactor type. However, some key basic differences between pebbles and prisms now must be considered.

First, consider storage implications. ANTARES produces 124 m3 of fuel elements per GWe-year versus 123 m3 of pebbles per GWe/year. Assuming a 60% packing fraction, the required storage volume for pebbles is nearly doubled at 205 m3 per GWe-year. Furthermore, because of the homogeneous nature of the pebble fuel, volume reduction is not practical. For prismatic fuel, the case is markedly different. Fuel compacts comprise 20% of the fuel element waste volume. Separating the fuel compacts from the balance of the fuel element is possible and can reduce the high level waste volume by 80%. The balance of the fuel element graphite can be disposed of as low level waste.

Second, there are fuel handling implications to consider as well. Pebble fuel is easier to move than prismatic fuel blocks because, as in the PBMR design, they pneumatically transferred into storage tanks that serve as both short term and long term storage. Prismatic blocks need equivalent storage as well; but, at least initially, they must be handled individually. In the ANTARES design, on-site storage will be provided that is sufficient to hold the spent fuel output of 10-years of a 4-module plant's operation.

Third, reprocessing is the essential element in the closure of any nuclear fuel cycle. The following points demonstrate a strong case for reprocessing:

1. It allows the recovery of valuable residual fissile (U-235, Pu-239) and fertile fuel (U-238), which may be recycled for further energy production.
2. It permits more efficient management of the remaining waste, allowing for waste reduction and waste conditioning (in which waste volume and radiotoxicity are significantly reduced)
3. These achievements (1 and 2, above) are consistent with Generation IV goals of sustainability and waste reduction.
4. With the exception of head-end processing, it is wholly compatible with the existing reprocessing technology
5. It allows for the potential customization of the waste for final disposal (i.e., a waste form that is specially designed and qualified to optimize characteristics for long term disposal).

Because of its robustness, reprocessing TRISO-fuel is a challenge that is borne by both pebble and prismatic choices. However, the first phase of reprocessing consists of separating fuel particles from graphite moderator. This phase is easier for prismatic fuel than for pebble fuel. In prismatic fuel, the fuel particles are concentrated in the fuel compacts which can be readily separated from the bulk graphite of the fuel element. Conversely, in the case of pebble bed fuel, the fuel particles are homogeneously mixed throughout the pebble. Hence, all of the graphite moderator must be separated from the particles in pebble reactor fuel as opposed to only a much smaller amount of graphite contained in the compact. While the prismatic fuel has an advantage in this regard, it is not out of the question to crush pebble fuel and separate out the fuel particles; however, the issue of failed particles and their presence in the bulk graphite would have to be addressed.

In countries where reprocessing is available (e.g., France, Russia, United Kingdom, Japan), spent fuel is shipped to the reprocessor after an acceptable cooling period. It is the ideal situation because the availability of reprocessing eliminates the need for large and costly on-site fuel storage facilities. This is certainly not the case currently in the US. Nevertheless, with the advent of the GNEP program and the initiative to re-establish reprocessing in the US, the prismatic reactor is wholly compatible with this mission.

6.3 Fuel Handling and Refueling Issues

Geometry drives the choice of refueling method in the pebble-bed reactor. The basic fact that fuel in a pebble bed reactor is in the form a billiard ball-sized sphere and can roll makes this option a natural candidate for some form of an on-line refueling system. Geometry and reactor physics also pose difficulties for the alternate form - a batch-type pebble bed reactor (i.e., if one could manage the required excess reactivity required for a batch core, the pebbles would still need to be recirculated within the core to assure even burnup). The geometry of the prismatic reactor naturally leads to the choice of periodic refueling.

Given preceding primer, one can readily see that, realistically, the selection of the refueling option is a defacto decision inherent to selection of reactor type. Hence, the pebble bed reactor choice implies on-line refueling while the prismatic reactor choice implies periodic refueling. A comparison of the attendant fuel handling and refueling issues is provided in Table 6-5 and discussion provided thereafter.

Table 6-5: Fuel Handling and Refueling Issues

Fuel Handling and Refueling Issues	Prismatic Reactor	Prismatic Rating	Pebble Reactor	Pebble Rating
Fuel Elements in Core	1020	o	460,000	o
Refueling Method	Batch, 50%	o	Continuous	o
Refueling Interval, Months	18	o	na	o
Refueling Duration, Days	22	o	Continuous	o
Fuel Moves/Day During Refueling or Pebbles handled per day, continuously.	100-200	++	3000-5000	-
Reflector Replacement (i.e., during refueling)	Yes	+++	No	-
Special Equipment for Reflector Removal	No	+	Yes	-
Complexity of Refuel Equipment	High	+	Very High	-
Module Sharing of Refuel Equipment	Yes	++	No	-
Other Maintenance Opportunity	High	++	Limited	-
Planned Major Outage	Not Req'd	++	Req'd	-
Major Outage Frequency, Years	Not Req'd	o	5	o
Impact of Refuel Equipment Breakdown	Extended Refueling Outage	++	Unplanned Outage	-
Ability to Maintain/Repair Refuel Equipment During Normal Operation	Yes	+	Limited	-
Overall Rating for Fuel Handling and Refueling		++		

6.3.1 Fuel Handling Benefits, Risks and Tradeoffs

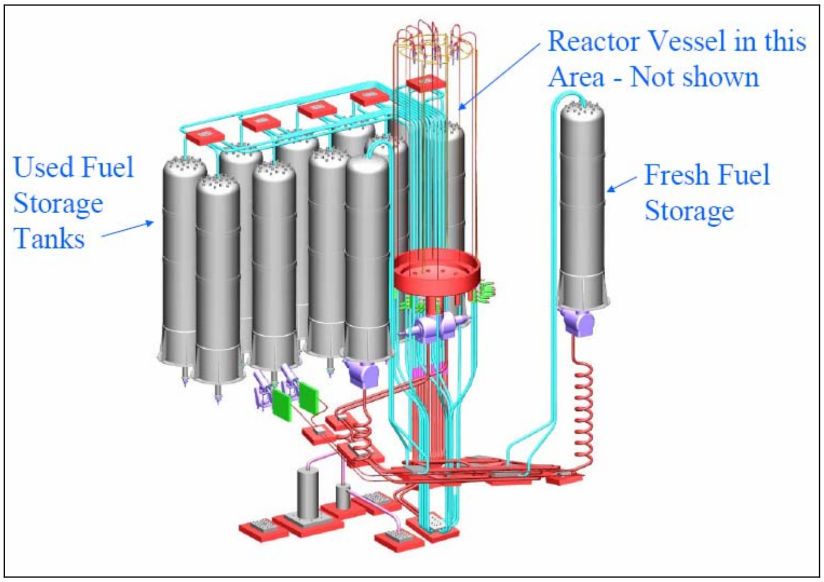
The benefits, risks and tradeoffs associated with online vs. offline refueling is therefore an important consideration in choosing the reactor type.

6.3.1.1 Pebble-bed reactor

In the pebble bed reactor, the on-line refueling equipment is constantly operating and the impact of potential down time is a strong concern on plant availability. Furthermore, continuous operational pressure arising from on-line refueling presents less opportunity for planned maintenance which could possibly result in deferred maintenance of other plant systems

The pebble handling system is shown conceptually in Figure 6-2 [6]. Because it must operate continuously, the pebble handling system must highly reliable. It is also a complex system. These combined attributes may make it relatively costly to build and maintain. Furthermore, the pebble handling system build cost is repeated for every module needed.

Figure 6-2: Conceptual Pebble Handling System



The prospective advantage of online refueling is an increase in plant availability. PBMR claims that a 98% availability is possible with a maintenance outage every 5-years. However, the theoretical advantages of online refueling have not been realized over the longer term in other commercial power reactor designs (e.g., CANDU reactors). For pebble reactor, pebble handling equipment reliability is the key consideration. The pebble handling system at AVR worked well after improvements but did account for 3% generator unavailability [7]. PBMR's target unavailability for their pebble handling system appears possible in light of AVR experience.

On-line refueling demands that approximately 3000-5000 fuel spheres be processed each day. While refueling equipment problems of up to approximately a week in duration may be tolerable, longer term problems will force shutdown and reduce plant availability. Personnel access to refueling equipment to effect repairs may also be limited. Furthermore, extremely long operational runs will place increased demands on mechanical equipment and could result in higher forced outage rates, thereby further negating the advertised benefit of online refueling.

The negative impact of the higher forced outage potential cannot be understated. It is not simply the prospect of the plant being forced to shutdown to fix a problem. It is the unpredictability of the timing of the forced outage. The resources to do major maintenance work at a nuclear plant are not instantaneously available, making unplanned outages more costly simply because they are unplanned. Furthermore, the timing is critical because of the extremely costly prospect of having to buy replacement power to meet generation commitments, especially when the power demand is high. When normal generation costs approximately \$50/Mw-hr and replacement power on the spot market is running \$500/MW-Hr, the economic advantage of continuous refueling can erode very quickly.

An additional concern with the continuous refueling system is that of the potential to introduce another source of contamination in to the primary coolant system. Each pebble will make up to 8 or 10 trips through the fuel handling system before being discharged as spent fuel. In those transits, the pebbles will erode fuel handling system and could potential be a carrier of erosion products (e.g., iron, nickel) into the core for activation.

Finally, reflector replacement in the pebble reactor is problematic because there is no opportunity to do so with the on-line refueling equipment. PBMR will require a special mid-life outage to replace reflectors. It will also need special equipment to perform the replacement and, moreover, removal of the reactor vessel head (a major evolution for an HTR) is required.

6.3.1.2 Prismatic Reactor

The prismatic reactor (e.g., ANTARES) is refueled every 18-months. Refueling takes approximately 22 days. The reliability of the refueling equipment must be commensurate with meeting this refueling window. The prismatic core is refueled in 1/6 radial segments. With a 50% fuel management scheme, approximately 1500-2000 fuel blocks must be moved each refueling. This includes the complement of 510 fresh elements being introduced and the same number of spent elements being removed. The balance of the fuel moves are necessary to configure the core to the desired loading scheme. Additionally, refueling also offers the opportunity to replace graphite reflector blocks on a periodic basis. Typically, a reflector block adjacent to fuel will be replaced every 6 years or every 4-cycles.

The question of refueling equipment reliability is very important for the prismatic. Its design is also challenging. The system must be able to accurately move over three axes bearing a 150 Kg at the end of a long reach and perform many manipulations. Such prismatic fuel handling equipment has been successfully demonstrated (at Fort St. Vrain). It is reasonable to expect that this technology, update appropriately, can be readily developed for moderns HTRs.

An added advantage of periodic refueling of the prismatic reactor is that maintenance of the refueling equipment itself can be accomplished when the plant is operating. Furthermore, once commercialized and in a multi-module setting, maintenance of refueling equipment can be accomplished for all modules during non-refueling periods, optimizing use of the both the equipment and the refueling staff.

Finally, the refueling window is very advantageous to the prismatic option since it represents an opportunity to perform scheduled maintenance on other plant equipment, thereby lessening the chances of forced outages during operation. While this does not completely eliminate the potential for forced outages during plant operation, it certainly will contribute to reducing the frequency of forced outages.

6.3.2 Perspectives on Fuel Handling/Refueling Method

In conclusion to the foregoing discussion, it is difficult to declare one option having a clear advantage with respect to fuel handling and refueling issues. A significant difference in availability with online vs. offline refueling, as previously stated, is not strongly supported through prior experience; however, it must be recognized that mature commercial scale designs have not yet evolved for either approach to HTRs. In the end, the determining factor will be whether the unplanned unavailability associated with the more complex operational configuration of online refueling exceeds the marginal evaluated advantage of that approach versus periodic refueling. Effectively, this also represents a tradeoff between overall planned unavailability, which can be optimally timed to power generation requirements and unplanned unavailability, which is random.

The prismatic reactor periodic refueling approach is more advantageous and consistent with current operating philosophy (i.e., consistent with current day LWRs). This reasoning is supported by AREVA NP's choice of reactor type whereby the selection of the prismatic form for ANTARES was based on four key perceptions regarding the associated tradeoffs:

1. The prospective availability advantage associated with online refueling is remains to be demonstrated
2. The potential economic benefits associated with prospective availability advantages are outweighed by the higher power capability of the prismatic core (if high power is needed)
3. Planned maintenance can be schedule more uniformly throughout the life of the plant and timed more appropriately to utility planned outage requirements.

Finally, power level aside, it must be recognized that the end user requirements may impact the selection of reactor type. The details of specific process heat requirements of the systems supplied by the reactor may make the choice of refueling concept either vitally important or unimportant. That is to say that continuous refueling may be more amenable to certain process heat users than others and vice versa for period refueling. The point is that the plant can be used in other modes than the standard supply electricity only mode.

6.4 Economic Factors

The key economic discriminators are compared in the below table and discussed in detail afterwards.

Table 6-6: Economic Factors

ECONOMICS	Prismatic Reactor	Prismatic Rating	Pebble Reactor	Pebble Rating
Capital Cost	Lower	++	Higher	-
O&M Cost	Lower	++	Higher	-
Fuel Cost	Similar	o	Similar	o
Waste Costs	Lower	++	Higher	-
Decommissioning Costs	Similar	o	Similar	o
Overall Rating for Economic Factors		++		

6.4.1 Capital Cost

Both the prismatic and pebble bed reactors can be designed with similar secondary systems; hence, the only discriminators for the purpose of this study are those cost items comprising the envelope of the nuclear core. These are core, initial fuel and reflector elements, core internals, control rod systems, fuel handling and storage equipment, reactor vessel and related equipment.

A major cost item in the preceding list is the reactor vessel. The following table compares some of the key data for the prismatic and pebble reactor vessels:

Table 6-7: Reactor Vessel Data

Reactor Vessel Parameter	Prismatic Reactor	Pebble Reactor
Overall Height, Meters	25	30
Internal Diameter, Meters	7.2	6.2
Operating Pressure, MPa	5	9
Weight, Tonnes	965*	1000**
Material (design option)	9 Cr 1 Mo	SA508

*ANTARES vessel at 6 MPA; **PBMR vessel

Note that the prismatic reactor vessel is somewhat similar in geometric size to the pebble reactor vessel but almost identical in weight. A key discriminator is the choice of material. Hence, assuming the same material, the cost of these reactor vessels will be similar as well; however, cost of the prismatic reactor vessel per unit of energy produced will be less (if higher power is needed). However, the prismatic reactor needs to use an advanced vessel material which will serve to reduce the advantage of the prismatic reactor's larger power output.

Of the remaining cost items introduced above, the fuel handling and storage system is perhaps the most remarkable with respect to cost differential. Items such as core internals, control rods, reflector elements will bear similar costs in each option.

The pebble bed reactor has a very complex pebble handling system as shown previously in Figure 6-2. It will be very costly to design and build a system with the requisite high reliability that will continuously circulate the highly radioactive pebbles for the life of the plant. Furthermore, in a multi-module setting, the build cost is repeated in how ever many modules as needed.

The additional complexity shown in the PBMR design will add significant costs as well (fuel handling system). Significant additional storage capacity for pebbles not in the core will be needed, adding to cost. In conjunction with storage for pebbles not in use, a complex radiation measuring system will need to be created that can determine the needed information about each pebble that is removed from the reactor at 30-second intervals. All of these measuring and storage systems will need to be safety rated and operating at all times in order to prevent an unscheduled outage – further increasing costs

The pebble bed design will also likely require a large number of extra pebbles to be ready to circulate in the core when the plant first reaches criticality. While it is not an overall extra cost, it is an additional upfront cost that will need to be paid sooner than other reactor designs require.

6.4.2 Fuel Cost

Fuel fabrication costs are not readily available, but the main cost component of both designs is expected to be that of the TRISO fuel particles. Considering that the pebble reactor requires more particles per unit energy produced may be a cost discriminator; however, it is premature to judge this one way or the other without hard fabrication cost data. Additionally, the cost of fabrication of the fuel particles into the final fuel form (i.e., either fuel compacts/blocks or pebbles) is not expected to be much different for pebble or prismatic options.

In enrichment costs, there will be a difference because the prismatic reactor requires 14% enriched fuel while the pebble core requires 9.6% enriched fuel. Hence, more separative work units will be required for the prismatic fuel (enrichment factor of 30 versus 20). Nevertheless, on a unit energy basis, the natural uranium requirements are remarkably similar between prismatic and pebble bed fuel as shown in Table 6-3.

Enrichment costs are only one component of the cost of fuel fabrication, and it is generally agreed that the bulk of the cost of fuel will lie in the manufacture of the particles themselves. Hence, as shown previously in Table 6-3, the pebble reactor requires 33% more particles to be fabricated per unit of energy delivered than the prismatic reactor. This is an advantage for the prismatic reactor; however, given the more severe service conditions, this advantage may be offset by higher costs due to stricter quality requirements.

While enrichment costs may be higher, the prismatic offers significantly more capacity per module. Assuming a 47% electric conversion efficiency for the prismatic, a 600 MWt prismatic module offers 264 MWe. The PBMR module is designed for 400 MWt with a 41% electric conversion efficiency, offering 164 MWe. Therefore, a similar sized prismatic module offers 61% more power then the PBMR design. Assuming a 600 MWt for the PBMR, mentioned in the NGNP Point Design study [8],there would be 246 MWe produced, or still 11% more power in the prismatic design. The point here is simply that even with the same power capability, the prismatic reactor will still realize a fuel economy benefit.

The pebble bed design will also likely require a large number of extra pebbles to be ready to circulate in the core when the plant first reaches criticality. While it is not an overall extra cost, it is an additional upfront cost that will need to be paid sooner than other reactor designs require.

Enrichment differences, number of particles required and fuel utilization comprise key fuel cost components as discussed above; however, there are many other factors to be considered such fuel fabrication costs, cost impact of required quality control requirements, graphite costs, etc. must also be considered. At this time, there are simply too many parameters, many with offsetting cost components, and too many unknowns with respect to overall fuel cost to be able to declare that one reactor type has a fuel cost advantage over the other. The correct judgment is that fuel costs will be similar given that both the prismatic and pebble reactors face similar situations with respect to their fuel, its qualification, and its fabrication.

6.4.3 Operation and Maintenance Cost

The operation and maintenance of an HTR require fewer personnel than do light water reactors for the following reasons:

- Simpler, more compact design
- Fewer systems and components
- Smaller staff sizes

Considering that the scope of this study encompasses only the reactor vessel and its contents, any differential with respect to cost must addressed within that context. The main discriminator is the fuel handling system. In the pebble bed reactor (i.e., PBMR), the fuel handling system is a large and complex system and continuously operates (which requires that much maintenance be performed on it while “hot”). The fuel handling system in the prismatic reactor, on the other hand, is somewhat smaller and, while complex, it is not on the same level of complexity compared to the pebble handling system. Additionally, it is decoupled from the spent fuel storage mission which is an integral part of the pebble system. Hence, the prismatic reactor is considered as having the advantage of “less complexity” that will require less resources to operate and maintain.

6.4.4 Waste Costs

Because it does not have extensive water purification systems to maintain, it is expected that the low-level waste generate by an HTR would be lower than that of light water reactors. This expectation applies equally to both reactor types.

However, as discussed previously in Section 6.2.3, the prismatic reactor will have a waste cost advantage over the pebble core primarily due the ability to separate fuel related waste into constituent parts and lower storage volume requirements.

6.4.5 Decommissioning Cost

The smaller, simpler core of the pebble bed reactor appears to be easier to decommission, but the pebble circulation system adds significantly more complexity and cost. The decommissioning costs for the prismatic reactor core should therefore be significantly less because it does not have the additional radioactive systems. Since the rest of the reactor systems can be designed in similar ways for either reactor type, there does not appear to be any additional decommissioning cost distinctions.

6.5 Research and Development

Research and development risks are present in Table 6-8 below followed by explanatory discussion.

Table 6-8: R&D Difficulty

DEVELOPMENT & R&D	Prismatic Reactor	Prismatic Rating	Pebble Reactor	Pebble Rating
Fuel Development	Higher	-	Lower	++
High Temperature Materials (e.g., core internals)	Similar	o	Similar	o
Reactor Vessel	Higher	-	Lower	++
Graphite	Lower	+	Higher	-
Licensing Methods, Computer Code Development, and Qualification	Lower	++	Higher	-
Refueling Equipment	Similar	o	Similar	o
Overall Rating for R&D Difficulty		o		o

6.5.1 Fuel Development

The safety case for HTR technology wholly depends on the performance of TRISO particle fuel. Successful development of TRISO particle fuel technology requires a combination of knowledge and skill in order to establish an acceptable fuel particle design, develop a cost-effective fuel fabrication process, and demonstrate that the resultant fuel meets the required performance objectives. It is critical that any fuel development program obtain a detailed knowledge and understanding of each step of the process. Both “know how” and “know why” must be equally obtained before a valid basis for fuel design and fabrication specifications can be established.

As previously mentioned in Section 6.2.1.1, two different fuel development strategies are being followed. PBMR is qualifying their fuel to the previously qualified German fuel. Prismatic reactor proponents are developing an advanced fuel design to accommodate the large performance envelope shown in Figure 6-1. Each program, given enough time, should be successful.

The main risk, therefore, associated with fuel development for NGNP is primarily a schedule risk. Namely, will the advanced fuel design be available in time to meet the initial operation target date of 2018? The ongoing DOE-sponsored Advanced Gas Reactor (AGR) Fuel Development and Qualification Program is working to reestablish coated-particle fuel fabrication capability in the U.S. and to qualify a coated-particle fuel design for use in advanced gas reactors. However, the current AGR Program schedule does not match the NGNP need for fuel by 2018 and would have to be accelerated significantly to do so. To this end, AREVA and BWXT have jointly stated it is feasible to provide either UO_2 or UCO fuel on a schedule that is consistent with NGNP requirements.

Based on the above, it appears PBMR and the pebble technology has an advantage with respect to fuel R&D. This is not to say their strategy is not without risk because PBMR must successfully demonstrate the ability to manufacture fuel to the quality of the past German fuel but they also must demonstrate that they have mastered their understanding of its behavior.

6.5.2 High temperature materials

The Next Generation Nuclear Plant (NGNP) HTR will demonstrate the use of nuclear power for electricity, hydrogen production, and process heat applications. The HTR will have an average reactor outlet temperature of approximately 900 ºCº - 1000 ºCº. The design service life of the NGNP is 60 years.

The thermal, environmental, and service life conditions of the NGNP will make selection and qualification of some high-temperature materials a significant challenge. High temperature metallic materials, graphite, and SiC-SiC and C-C composites are being considered for use. Important materials issues that must be addressed include:

- High-temperature mechanical properties (e.g., tensile, creep, creep fatigue, stress-rupture, high and low-cycle fatigue, fracture toughness) in air and impure helium environments
- Environmental degradation processes from exposure to high-temperature helium with contaminants such as CO, CO_2, H_2, H_2O, and CH_4
- Long-term irradiation effects on mechanical properties (e.g., tensile, creep, creep fatigue, stress-rupture, high and low cycle fatigue, fracture toughness)
- High-temperature metallurgical stability (thermal aging effects)
- Development and validation of new sources of graphite materials
- Extension of ASME Code approval for metallic materials at the higher NGNP operating temperatures
- Development and ASME Code approval for 9 Cr-1 Mo steel, graphite, composite, and ceramic materials
- Development of component fabrication technologies for critical components such as the reactor pressure vessel (RPV) and control rods
- Emissivity of the RPV surfaces for cool-down under accident conditions

Because the average reactor outlet temperatures of either the pebble or the prismatic reactor are not significantly different, the R&D risk associated with the necessary high temperature materials for core internals, control rods, fuel handling equipment etc. is considered to be similar for both options.

More discussion relative to reactor vessel material is presented later in Section 6.10.

6.5.3 Graphite

Graphite is the foundation for HTR technology. Both reactor types, pebble bed or prismatic, need qualified grades of graphite for the key component in the reactor core: fuel, reflectors, core support structures. Significant R&D will be required to qualify the different grades of graphite that will be used. Does one option have an R&D advantage relative to graphite over the other?

The prismatic reactor has an advantage in this regard due to the fact that in the prismatic reactor, both inner and outer reflector blocks can be routinely replaced (target 6-year replacement frequency). In the pebble bed reactor, the outer and central reflectors will be replaced every 20-years during a special outage. Hence, the R&D for the pebble reactor must qualify their reflectors for significantly greater neutron fluence.

6.5.4 Licensing Methods, Computer Code Development, and Qualification

Both prismatic and pebble bed reactor technologies will require significant effort in the area of methods development and qualification. The spectrum to be covered is quite broad since every facet of the technology requires attention in this regard.

Again, as in the previous section, the question "Does one option have an advantage over the other?" needs to be addressed. In this case, the answer is clear – the prismatic reactor option, due to its static core geometry, has a significant advantage.

The stochastic nature of the pebble core simply adds another dimension of complexity on top of already complex issue area. This has not gone unnoticed – witness below the concerns raised in Reference [9] as paraphrased below:

> *"Core physics will be constantly changing as the pebbles flow through the core, necessitating some statistical bounding of key parameters. Differences between the center of mass and the center of gravity for individual pebbles, and surface defects and irregularities may result in non-linear conditions governing pebble flow through the core which can make it impossible to reliably predict the transit time for any particular fuel pebble, or even the fuel pebble packing density within the core."*
>
> *"Design analyses will require development of appropriate thermal-hydraulic codes to deal with the complex geometries and uncertain core configurations of the pebble bed design."*
>
> *"At present, there are gaps in the spectrum of internationally accepted codes and standards dealing with nuclear grade graphite and the fabrication of graphite components for use in HTGRs; with thermal-hydraulic codes for use in the complex geometries of a Pebble Bed Modular Reactor (PBMR); nucleonic codes that can accurately predict the transient and accident response of a loosely coupled, statistically bounded pebble bed core configuration."*
>
> *"The statistical nature of the distribution of the PBMR fuel could also conspire to make one section of the spent fuel array particularly reactive. Criticality control events at fuel fabrication facilities have shown that processes in place to exclude moderator from an area occasionally fail, as do geometry and quantity controls, thus, care will need to be exercised in the management of criticality during the storage, transportation, and disposal of the PBMR spent fuel pebbles."*

Both technologies need to develop their respective licensing and analytical methods and secure regulatory approval for them as a prerequisite for licensure; however, the main point is the pebble reactor faces a much more difficult challenge due to the stochastic nature of the pebble core. This not to say it cannot be done, which it can; but, it will probably be at the expense of having to provide additional margins the prismatic option would not have to give away.

6.5.5 Refueling Equipment

Both prismatic and pebble bed reactor technologies depend on highly reliable fuel handling equipment. The pebble bed reactor has its pebble handling system which has to work continuously whereas the prismatic reactor depends on its fuel handling system to work flawlessly during its refueling outage. Both systems are complex and require significant development work; however, this is not so much an R&D problem but more of an engineering problem. Hence, with respect to R&D, the associated risks are considered similar.

6.6 Core Design Issues

In this section, core design and capability differences are examined. The section is divided into three subsections; namely, general considerations, central reflector, and core stochastic subsections.

6.6.1 General Considerations

The performance capabilities of the pebble bed reactor and prismatic reactor are compared in this section as shown in Table 6-8.

Table 6-9: Principal Core Design Features

CORE DESIGN ISSUES – 1	Prismatic Reactor	Prismatic Rating	Pebble Reactor	Pebble Rating
Core Power Density, KW/liter	6.6	+++	4.7	-
Reactivity Control				
Excess Reactivity, %Δk/k	3-5	-	1-2	+
Control Rods in Fuel Region	Yes	++	No	-
Control Rods in Reflector Region	Yes	o	Yes	o
Alternate Shutdown Capability	Yes	o	Yes	o
Xenon Defect Override Capability	Yes	o	Yes	o
Xenon Stability/Oscillation Control	Yes	o	Yes	o
Flexibility/Adaptability				
Fuel Zoning	Yes	++	No	-
Burnable Poisons	Yes	o	Yes	o
Axial/Radial Shuffling	Yes	+	Limited	-
Pu & Actinide Burning Capabilities	Yes	+	Limited	-
Deep Burn Capabilities	Yes	+	Limited	-
Traceable limiting fuel location	Yes	+	No	-
Mis-loaded Fuel Possibility	Yes	o	Yes	o
Overall Rating for Core Design Issues -1		++		

6.6.1.1 Power Density

Prismatic cores can have higher power densities than pebble reactors. Compare ANTARES power density of 6.5 KW/L versus PBMR's 4.7 KW/L – approximately 50% higher. The maximum power density each option can achieve is a function of that option's acceptable performance in the limiting design basis event (i.e., depressurized conduction cool down) and thus demonstrating inherent safety.

6.6.1.2 Reactivity Control

Pebble cores, due to continuous refueling, require only a minimal amount of excess reactivity. This reactivity is managed with control rods that, by necessity, are located in the outer reflector region. As a result of THTR experience, control rod insertion directly into the pebble core is not considered a design option. While minimum excess reactivity is considered an advantage by pebble reactor advocates, its management becomes more difficult as the pebble core power is increased. Hence, increases in the pebble core power are usually accompanied by increased height rather than increased radial dimension. This is done to maintain the effectiveness of the control rods.

The pebble reactor core, due to its low reactivity margin, may have difficulty in performing load follow maneuvers or difficulty in overriding post-shutdown Xenon defect. Control rods would need to be able to add sufficient compensating reactivity in order to maintain power or restart the reactor. The situation could also be exacerbated by the unavailability of the pebble handling system which would preclude addition of fresh fuel and removal of spent fuel.

Prismatic cores require additional fuel material in order to be able to maintain full power throughout its cycle. The additional reactivity that this represents can be readily managed. First, the prismatic core can readily accommodate control rods or alternate shutdown channels directly in the fueled zone. Second, additional control rods can be located in the reflector region for further control. Third, prismatic fuel elements can accommodate fuel zoning and burnable poisons for long term reactivity control in a manner similar to current day LWR cores. Furthermore, the axial and radial shuffling of fuel elements is possible which facilitates core management. Finally, increases in both radial and axial dimensions can be considered in increasing core power level since control rods can be located within the fueled region.

The prismatic core's excess reactivity capability is an asset with respect to being able to override the effects of xenon, either following load follow operations or a full shutdown. The post-shutdown maximum Xenon defect is on the order of 4-5% $\Delta k/k$ (similar to LWR behavior). Sufficient excess reactivity is available to override this level of defect; hence no restart issues are envisioned. (This result has been confirmed by preliminary analysis results for ANTARES.) Further more, at the currently envisioned height, the prismatic core is not anticipated to be susceptible to xenon oscillations; however, the prismatic core does have many design options through which compensatory measures can be implemented should it become necessary.

6.6.1.3 Fuel Flexibility/Adaptability

The prismatic core design affords excellent fuel cycle flexibility whereas the pebble bed design, due to its stochastic core, is much more constrained both spatially and temporally. The reasons for the prismatic core advantages are as follows:

1. A fixed-core geometry allows for fuel zoning and burnable poison capabilities, both important to efficient fuel management

2. Axial and radial shuffling of fresh and exposed fuel elements also contributes fuel management flexibility

3. Due to finer control of core geometry, the prismatic core is more adaptable to other fuel types and (PuO, actinide burning, deep burn etc.)

4. Limiting core locations with respect to maximum fuel burn-up and maximum fuel temperatures are traceable throughout the cycle and, by inherent design, would never be concurrent.

The pebble bed core may offer a degree of fuel cycle flexibility as well; however, it will be difficult for it to achieve the same level of flexibility as the prismatic core because of the larger operating margins it must have to accommodate an ever changing core configuration. Furthermore, in the pebble bed core, the limiting core location is not as traceable, and given the random paths the pebbles take, the possibility of the maximum burn-up pebble being at the maximum temperature location is distinctly real.

6.6.1.4 Mis-Loaded Fuel

Refueling of prismatic cores requires the placement/replacement of many prismatic fuel elements to replenish the core with fresh elements. The possibility of a miss-loaded fuel element, though unlikely, cannot be ignored. The probability of mis-loading a fuel element is low because refueling is computer controlled and refueling algorithms are thoroughly verified on an element-by-element basis prior to refueling operations. Nevertheless, this unlikely event must be anticipated and its impact be demonstrated to be acceptable within operational limits.

The pebble bed core is not susceptible to fuel mis-loading in the same sense as in a prismatic reactor because the fuel is all the same. However, there is the potential to overcharge the pebble core with fresh fuel; however, this is very improbable and, given the individual worth of a pebble, probably of little impact. Hence, for the pebble reactor, this is judged as a non-event.

6.6.2 Core Physical Features

The prismatic and pebble core physical features are examined in this section as shown in Table 6-9 below.

Table 6-10: Core Physical Features

CORE DESIGN ISSUES-2	Prismatic Reactor	Prismatic Rating	Pebble Reactor	Pebble Rating
Annular Core	Yes	o	Yes	o
Top Axial Reflector	Integral / Regularly Replaced	++	Permanent	-
Central Reflector	Integral / Regularly Replaced	++	Semi-Permanent Free Standing Column	-
Outer Reflector	Integral / Regularly Replaced	++	Permanent	-
Bottom Axial Reflector	Integral / Regularly Replaced	++	Permanent	-
Overall Rating for Core Design Issues-2		++		

Both prismatic and pebble cores at the referenced power levels are designed as annular cores with a central reflector. However, the central reflector represents more of a design challenge for the pebble core versus the prismatic core.

In the prismatic core, prismatic reflector elements are essentially identical to prismatic fuel elements except they do not have fuel or coolant channels. Moreover, they are designed to be handled in similar fashion to the prismatic fuel element and do not require special handling equipment. Furthermore, prismatic reflectors in the central region, along with the annular core and outer reflector elements are all constrained within the core barrel and upper core constraints. Finally, reflector elements are periodically replaced including those located in the central reflector region.

In a pebble reactor with an annular core similar to PBMR, the central reflector is a 9-meter high column of graphite freestanding in a sea of fuel pebbles. The column consists of inter-locking graphite blocks. These blocks will need to be replaced at least once during the life of the pebble reactor. Separate handling equipment will need to be provided and, more importantly, a prolonged outage will be required to perform the replacement because the evolution will require removal of the reactor vessel head. Furthermore, demonstration of the seismic adequacy of this tall columnar design may prove very challenging.

6.6.3 Core State Issues

A major difference between the prismatic and pebble bed reactor is the latter's stochastic core (versus the static nature of the prismatic core). In the pebble core, the fuel pebbles are continuously removed from the bottom and replaced at the top; hence, the fuel pebbles flow down through the core. The key aspects of core state issues are summarized in Table 6-10 below and discussed thereafter.

Table 6-11: Core State Issues

CORE STATE ISSUES	Prismatic Reactor	Prismatic Rating	Pebble Reactor	Pebble Rating
Core Geometry State	Fixed	+++	Random	-
Statistical Core Design Req'd	No	++	Yes	-
Add'l Margin to Accommodate	No	++	Yes	-
Packing Fraction, Bridging Issues	No	+	Yes	-
Pebble flow path predictions	na	++	Difficult	-
Validation of Max Conditions (Temp, Bu, Power)	Easier	++	Harder	-
Overall Rating Core State Issues		++		

Pebble flow is difficult to predict. Even more so is the prediction of the spatial distribution of fresh and burnt pebbles and the corresponding power profiles and temperatures. Given this behavior, bounding core analysis techniques (physics plus thermo-hydraulics) must be used in assessing safety margins to ensure operational limits are not exceeded. This approach was thought adequate, however, testing at AVR [Test HTA-8] revealed unexpected hot spots that were significantly hotter than expected maximum coolant temperatures (i.e., > 1280 ºC). Furthermore, THTR experienced pebble flow distributions that were significantly different than predicted. Pebble flow distribution fluctuations impacted operational and safety-related core characteristics (e.g., power distributions, temperature distributions, nuclear shutdown margins).

USNRC staff involved with HTR licensing is well aware of flow distribution issues in the pebble reactor and will need to be ensured that flow distribution anomalies are adequately addressed in design and safety analyses. Additionally, the IAEA has prepared a report [9] identifying key issues relating to the safety and licensing of an HTR. With respect to the pebble reactor core, the report highlighted issues related to the stochastic nature of the pebble core. A synopsis of their concerns is summarized in the following points:

- The statistical distribution of PBMR spherical fuel results in additional uncertainties in the character of the core, uncertainties that will vary over time. These additional uncertainties will need to be addressed in the nucleonic, thermal hydraulic, and fuel performance codes in licensing the PBMR design.

- The statistical nature of the distribution of the PBMR fuel could also conspire to make one section of the spent fuel array particularly reactive. Criticality control events at fuel fabrication facilities have shown that processes in place to exclude moderator from an area occasionally fail, as do geometry and quantity controls, thus, care will need to be exercised in the management of criticality during the storage, transportation, and disposal of the PBMR spent fuel pebbles.

- Core physics will be constantly changing as the pebbles flow through the core, necessitating some statistical bounding of key parameters. Differences between the center of mass and the center of gravity for individual pebbles, and surface defects and irregularities may result in non-linear conditions governing pebble flow through the core which can make it impossible to reliably predict the transit time for any particular fuel pebble, or even the fuel pebble packing density within the core.

While both reactor types will certainly have a learning curve to follow upon initial start-up, the pebble core with its random core configuration has an additional level of complexity to deal with in ferreting out problems. Consider the following “teething” experiences at THTR: broken pebbles, higher than predicted core bypass flows, uneven pebble flow distribution between center and periphery, and larger than predicted temperature gradients at core exit.

This is not to say the prismatic core will not have its share of growing pains; however, it is easier to address problems in a static situation as opposed to a constantly changing one. Take for example the solution of core flow fluctuations which occurred during initial operations at Fort St. Vrain. The flow fluctuations were caused by the prismatic blocks shifting slightly. The problem was solved by the addition of core restraint devices know as “Lucy Locks” which prevented any further block movement [10].

Another issue faced by the pebble core is concerned with fuel pebble bridging. This phenomena occurs when a section of pebbles literally locks it self in place, allowing other pebbles to flow around it or, if severe, hold a part of the core stationary.

Finally, to ensure operational limits are adequately met, the pebble core has to operate with larger margins than the prismatic core. Hence, this has implications for the extent to which the pebble design can be optimized.

In summary, the fixed state of the prismatic reactor core is a significant advantage relative to licensing the technology and analytically demonstrating its safety case. The stochastic nature of the pebble core will serve as a “lightening rod” to regulators and will require a significant level of effort above that required for the prismatic reactor for the proponents of pebble technology to demonstrate its safety case. This is also likely to be a “confidence issue” with both likely end-users and the public in general.

6.7 Maintenance Issues

Both reactor types, prismatic and pebble, must be not only be designed to facilitate maintenance but must also be designed to be ALARA (as low as reasonably achievable) with respect to the potential radiation dose imparted to maintenance workers. Issues related to core component accessibility and replacement capability are judged to be roughly equivalent between the reactor types for items such as control rods, in-core instrumentation, ex-core instrumentation etc. from both maintenance ease and ALARA perspectives. However, there are several maintenance areas where the prismatic reactor has a clear advantage over the pebble reactor. These are summarized in the table below:

Table 6-12: Maintenance Issues

MAINTENANCE ISSUES	Prismatic Reactor	Prismatic Rating	Pebble Reactor	Pebble Rating
Dust & Particulate Generation Impacts				
Quantity	Lower	++	Higher	-
Erosion	Lower	++	Higher	-
Blockages of passageways	Lower	++	Higher	-
Spread of Contamination	Lower	++	Higher	-
Impact of flow control/Pressure Control	Lower	++	Higher	-
Release Potential	Lower	++	Higher	-
Dust Control Measures	Lower	++	Higher	-
Mechanical Failure	Lower	+	Higher	-
Defueling/Refueling Capability	Yes	o	Yes	o
Reflector Replacement	Easier	++	Harder **(if required)**	-
Component Replacement capabilities	Yes	o	Yes	o
Component Accessibility (refueling systems)	Easier/Low Dose	+	Harder/High Dose	-
ISI Requirements	Easier	++	Harder	-
Overall Rating Maintenance Issues		++		

Dust generation in the pebble core is a major concern. As the pebbles flow down through the core, they are constantly rubbing against themselves and against the inner and outer reflectors. The resulting abrasion produces graphite dust that will be dispersed throughout the system. THTR experienced graphite dust deposition of about 1 mg/cm^2 which correlates to the expected weight loss due to abrasion. Nevertheless, it did require the addition of an enhanced filtering arrangement [11] [12]. Furthermore, THTR experienced an off-site radiological release involving graphite dust.

The main problems arising from dust generation are identified in the preceding table. Circulating graphite dust acts as an abrasive on the pebbles and core internals, and, in direct cycles, the turbo-machinery. The higher turbulence of the flow regime in the core may also exacerbate the level abrasion experienced. Furthermore, critical flow passageways may become blocked. This is of special concern if more advanced heat transfer technology is used where passageways, with dimensions in millimeters, will be prone to blockage. The spatial distribution of the dust may also be unpredictable especially if the reactor is coupled directly to the power conversion system and core flow varies with power level. Fluctuating flow fields will relocate stagnation points and correspondingly, the dust will relocate as well. Finally, and perhaps most importantly, the dust inventory represents an additional radionuclide inventory that could potentially be released in accident.

Given the foregoing dust related issue, the pebble reactor must implement dust control measures to eliminate the problem. It also should be noted that the prismatic reactor also is susceptible to dust generation; however, the magnitude of the dust problem in the prismatic core is significantly less.

While reflector replacement is a core management activity, it does have maintenance impacts. In the prismatic reactor, the reflectors are moved using the fuel handling equipment which can be removed for maintenance. On the other the hand, PBMR requires a special plant shutdown to replace the central reflector.

Regarding ISI requirements, there is concern [9]with the pebble bed reactor that given its on-line refueling capability, the amount of shut down time to perform NDE and ISI will be reduced, and with fewer periods of time with the entire core off-loaded, the accessibility of some components (for inspection and repair) will be more restricted. This may require in-service inspection and techniques to shift to on-line, real time monitoring. To address this concern PBMR is developing on-line ISI methods. Also, NDE techniques for nuclear grade graphite will need to be developed and improved, as well as remote methods to assess the surface condition and structural integrity of pebbles as they are examined before permitting additional passes through the reactor.

With respect to ISI and subsequent NDE examinations, the prismatic reactor is in better standing. The prismatic reactor's regular refueling interval permits opportunity to perform inspections. The fuel handling equipment access ports on the vessel head permit the insertion of inspection equipment. Granted the inspection equipment remains to be designed, tested etc., nevertheless, the prismatic reactor has the potential to better accommodate ISI requirements.

6.8 Operational Considerations

Operational considerations are summarized in Table 6-13 below and followed by explanatory discussion.

Table 6-13: Operational Considerations

OPERATIONAL CONSIDERATIONS	Prismatic Reactor	Prismatic Rating	Pebble Reactor	Pebble Rating
Past Reactor Experience (AVR, THTR, FSV) (with respect to core only)	Similar	o	Similar	o
Operational Core Management	Easier	+	Harder	-
Plant Staffing	Lower	++	Higher	-
Overall Rating – Operational Considerations		+		

6.8.1 Past Reactor Experience

Past experience relative to pebble reactor and prismatic reactor technology within the boundary of the reactor (i.e., core only) has been positive. The Ft. Saint Vrain prismatic reactor in the US and the AVR and THTR in pebble bed reactors in Germany experienced "teething" problems upon initial operation that required resolution. As mentioned previously, FSV experience core flow fluctuations that were eliminated by the installation of a core restraint system. AVR core was very successful but its fuel handling system did require frequent maintenance during its initial years of operation. The system worked well after undergoing a series of improvements. Perhaps THTR operational experience was more remarkable – broken pebbles caused by the direct insertion of the control rods into the pebble bed, larger than expected core bypass flows, uneven pebble transit times, and a significant amount of dust generation. Each one of these problems was addressed and satisfactorily resolved. Nevertheless, THTR operation was considered a success.

The lessons learned from this past experience are very important. It is interesting to note that the scale up in size from AVR (49 MWth) to THTR (700 MWth) was significant – more than a factor of ten! It is not surprising then level of difficulty encountered initially in THTR's larger core. On the other hand, Fort St. Vrain (842 MWth) had no comparable prototype yet, from a core perspective, operated remarkably well after the flow fluctuation problem was solved.

Consideration should then be given to the "leap" in technology the HTR/NGNP core will represent. The prismatic annular core will be about 25% smaller than FSV, excluding reflectors; and based on FSV experience, the annular configuration should pose no difficulties. On the other hand, the annular pebble core will be smaller than THTR as well; however, there is the potential to encounter difficulties due to the annular configuration (e.g., pebble flow behavior, bypass flow).

Overall then, based on the above discussion, the past experience with both types of reactor cores is viewed as similar.

6.8.2 Operational Core Management

Prismatic core management should be similar to the core management of current day LWRs. Loading patterns are developed and implemented during refueling. Upon startup, reactor operations staff track core behavior through monitoring to verify/validate core performance.

Pebble core management is distinctly different. There are no loading plans because of continuous refueling. However, assuring acceptable core parameters will be a continuous job. Pebbles will need to be monitored for burn-up and structural integrity. Pebble flow patterns will need to be established. Periodic reactor physics testing to confirm core nuclear characteristics will be required. Witness the difficulty THTR encountered with pebble flow distribution which has to be constantly monitored.

Hence, the prismatic reactor is viewed as being more operational friendly which is a definite advantage.

6.8.3 Plant Staffing

A prismatic plant facility consisting of 4 x 600 MWth prismatic reactor modules requires an operating plant staff of 225 people on-site[13]. Approximately 25% of the staff is licensed operators (60) with the balance being attributable to remaining standard departments (Administration, technical support, maintenance, radiological protection, radwaste, QA/QC, and security). Of the standard departments, another 25% or 60 people will be required for continuous support coverage for all four modules on a 24hr/day, 7-day per week basis.

The pebble bed plant facility consisting of 8 x 400 MWth modules will require similar staffing requirements; however, the total number will be greater due to the 4 additional modules. This means an additional 60 operations staff and an additional 60 support staff. This would increase the total staff required to 345 people.

Hence, the prismatic reactor in a multi-module setting can claim an advantage with respect to staffing.

Note: The pebble bed operations staffing numbers are not in agreement with PBMR staffing estimates which rely on the acceptance of reduced licensed operational staff. The above numbers assume 1 shift-supervisor per 2 modules and 2 reactor operators per module. This is a safe assumption since it matches current staffing requirements.

6.9 Safety and Licensing

Licensing and safety aspects of the impact of reactor type are summarized in Table 6-14 below and discussed thereafter.

Table 6-14: Safety and Licensing

LICENSING & SAFETY	Prismatic Reactor	Prismatic Rating	Pebble Reactor	Pebble Rating
Licensability	**Somewhat Easier**	**+**	**Somewhat Harder**	**-**
Safety (Overall)	**Higher**	**+**	**High but more difficult to prove**	**-**
Overall Rating – Safety and Licensing		**+**		

Licensability

Since its conception, the very attractive safety aspects of HTR technology have allowed it to remain in various forms of development over the past 40-years or so despite being overshadowed by LWR technology. The key advantages of the HTR design are:

- Low power density (order of magnitude less than typical LWRs)
- High thermal capacity of the moderator (huge mass of graphite)
- Slowly developing accidents (results directly from the combination of low power density and high thermal capacity)
- Single phase coolant (Helium)
- Robust first fission product barrier (coated particle fuel)
- Reliance on passive decay heat removal
- Large negative reactivity temperature coefficient

These attributes culminate in HTR designs that are inherently safe – i.e., they make it highly improbable to have a catastrophic core damage (i.e., meltdown) and a corresponding release of a large amount of radioactivity. Both the prismatic and pebble reactor options share these attributes.

The issues of fuel, materials, safety, security, safeguards, analytical methods, waste, etc. present significant challenges to the licensability of HTR designs. The impact of reactor type on the level of difficulty in resolving these issues varies, understandably, with the given issue:

- fuel qualification (albeit somewhat harder for the advanced fuel design, in either case, the regulator must be satisfied with the level of qualification that assures fuel performance)
- material qualification (similar operational vectors – e.g., temperature, fluence, duty)
- safety (both options must meet established safety goals)
- security/safeguards (theft/diversion of material in pebble reactor is a concern - see Section 6.12 for a detailed discussion)

- analytical methods (random nature of pebble core adds an additional degree of difficulty)
- waste (relatively similar waste profiles in terms of material and volume)

If there is one thread that runs through several of the above issues, it is the random nature of the pebble core which was previously discussed in detail in Section 6.6.3. Because the fuel spheres in the pebble core are constantly moving, statistical methods need to be used to derive bounding parameters to demonstrate margins. The validity of the statistical methods will need to be demonstrated to the regulator. In particular, the regulator will need to be assured, most likely by direct demonstration, that calculated parameter values will be conservative. For example, the USNRC is well aware of the AVR pebble melt-wire tests which indicated calculated local maximum core temperatures were non-conservative. Hence, regardless of the sophistication of statistical methods and arguments, the fact that pebble core limiting locations cannot be accurately predicted will be problematic.

Source term

The radiological source term is the amount of fission product inventory that is postulated to be released following a design basis accident. The prismatic core (ANTARES) has approximately 4600 Kg U enriched to 14% U235 (644 Kg). The pebble core (PBMR) contains 9 grams U per pebble or 4140 Kg U enriched to 9.5% U235 (393 Kg). Since most of the fissions come from U-235, the prismatic core radionuclide inventory is approximately 60% (i.e., 644/393) greater that of the pebble core; however, on a per megawatt basis, the core radionuclide inventories are equivalent. Nevertheless, the same site boundary dose limits must be met which would favor the lower inventory of the pebble core; however, the release fraction is the critical variable. In this regard, the prismatic reactor would potentially have the lower release fraction because fission products would encounter additional barriers (fuel compact, fuel block) as opposed to the single protective layer on the pebble sphere. In addition, the pebble core dust inventory release has to be included as well.

Safety Overall

As said above, both the prismatic and pebble cores share the same key attributes that all contribute to the HTR's inherent safety case. Both reactor types must be designed to meet all regulatory requirements and be demonstrated to be safe to operate. In this regard, that demonstration will be more complex and difficult for the pebble core option than the prismatic option due to the stochastic nature of the pebble core. This makes the safety case for the pebble option slightly more difficult to prove.

6.10 Key Component Design and Fabrication Issues

The key mechanical components comprise a large share of the plant capital cost and can have major impact on plant construction and operation. The reactor vessel and the prime mover for core flow are the main mechanical components associated with reactor type.

Based on the discussion presented below and summarized in Table 6-15, both options are judged to face a similar level of overall difficulty with respect to these components. On one hand, the level of difficulty associated with the pebble reactor vessel is judged to be easier than that of the prismatic reactor. However, on the other hand, the prismatic reactor does have a significant advantage over the pebble reactor relative to the prime mover for core flow.

Table 6-15: Key Components

MECHANICAL COMPONENTS	Prismatic Reactor	Prismatic Rating	Pebble Reactor	Pebble Rating
Assessment of Level Of Difficulty Of Key Mechanical Hardware Design And Fabrication		o		o
Reactor Vessel Design	Harder	-	Easier	++
Core Power	600 MWt	n/a	400 MWt	n/a
Material	9 Cr – 1 Mo	-	SA-508	++
Weight	Similar	o	Similar	o
Design Pressure	Lower	++	Higher	-
Vessel Fabrication	Harder	-	Easier	++
Fabrication Location (for NGNP)	On-site	o	On-site	o
Prime mover for core flow	Easier	+++	Harder	-
Overall Rating for Mechanical Components		**o**		**o**

Reactor Vessel – Weight, Design Pressure

As discussed in preceding Section 6.4.1 and shown in Table 6-7, the weight of the prismatic reactor vessel is comparable to that of the pebble reactor vessel. The reason for this, even with a 1-meter larger internal diameter, is that the prismatic reactor vessel design pressure is 5 MPa versus 9MPa for the pebble reactor vessel. This is a significant advantage; especially considering the higher power capability of the prismatic core.

The somewhat smaller pebble reactor vessel dimensions do not provide a significant advantage in terms of transportation issues. Either option's reactor vessel would need to be significantly smaller (around 5 meters OD) to be able to deliver a complete package at INL site. This means that in both cases, on-site fabrication will be required.

Reactor Vessel - Fabrication

Compared in Table 6-16, as examples of potential vessel designs for NGNP, are the basic dimensions of the PBMR and ANTARES reactor vessels (excluding the closure head):

Table 6-16: Reactor Vessel Data – Lower Section

Reactor Vessel Parameter	PBMR	ANTARES
Internal Diameter, M	6.2	7.2
Flange External Diameter, M	7.7 est.	8.3
Thickness, mm @ core beltline	180	170
Thickness, mm @ nozzle ring	285	270
Height, M (lower section, flange to vessel bottom)	25	19

In both cases, issues associated to the fabrication are similar: (1) forgings are required at least for the flanges and the nozzle ring; and (2) due to the overall dimensions, only Japan Steel Works can provide such large forgings.

For the NGNP, we have proposed a multi-loop design with 4 cross vessels instead of one unique one. For such a condition, the thickness of the nozzle will be below 230 mm which means that rolled plate could be used for the nozzle ring instead of a big forging (limit is at 9 inches according to former experience in the US for BWRs).

It is also to be mentioned that for the prismatic design, the fabrication of the forged ring with one unique cross vessel will be difficult (ingot size would be too large). The problem is the same whatever the material is (SA 508 or mod 9 Cr 1 Mo).

Material Selection

What is the reference material for the reactor vessel?

SA 508 Steel

SA 508 grade 3 class 1 is the conventional steel for forgings of PWRs. This material is already covered by ASME. The use of this material at higher temperature than 700oF is covered by Code Case N-499 and can be summarized as follows:

- 3000 h maximum duration between 371 and 427 °C
- 1000 h and no more than 3 events between 427 and 538°C

These requirements are quite severe and could be hard to fulfill, depending on the assumption of availability of active or passive systems.

An issue raised during the pre-application phase with PBMR was the interaction between SA 508 and helium. The feedback from experience with the material is primarily with water and R&D is required to demonstrate that corrosion will not be a problem.

In terms of weld qualification, there are no issues related to welding anticipated. In terms of product size, the PBMR vessel is larger than PWRs (EPR ID is 4.9 m) but is comparable with the size of BWRs vessels and it is likely that no detailed qualification will be required.

9- Chrome 1- Molybdenum

For mod 9 Cr 1 Mo, the following issues need to be addressed.

- Welding-1: problems of hot cracking met at the beginning of weldability actions have been fully solved but further optimization of the welding process and welding products is still required

- Welding-2: post-weld heat treatment has to be performed at higher temperature compared to SA 508 and this complicates the fabrication (this is not however considered a major problem by AREVA)

- Corrosion: mod 9 Cr 1 Mo should have a much better behavior than SA 508 in He environment. This will have however to be showed by specific R&D action (but probably program should be very limited)

- Forgings: there is an issue associated to the availability of big forgings. It is expected that ingot sizes up to 200-250 T could be obtained from JSW to be compared to about two times more for SA 508. Not a real problem if the design is based on plates with a limited number of forgings. This is more a problem if it would be required to have a full forging design in which case mod 9 Cr 1 Mo would require more circumferential welds. The forging of the nozzle ring (with one unique cross vessel) is an issue as already discussed above.

- Code qualification: mod 9 Cr 1 Mo is covered by subsection NH since edition 2004 but this subsection has to be extended to heavy section products

- NRC approval: Generally speaking, high temperature sections of the ASME have never been approved by the NRC and the approbation is likely to take some time

- A qualification will have to be performed to qualify the behavior of representative material in the core beltline (irradiation should be performed on base material and weld but irradiation already carried out in Europe already indicate a good behavior and this shall not be an issue). The characterization of the material of the forging will have also to be carried to demonstrate that the material in the bulk of the forging is as good as the material elsewhere.

Based on the above, the fabrication of the prismatic reactor vessel is judged to be harder primarily due to the combination of its larger diameter and the need to use an advanced material (i.e., 9 Cr-1Mo).

Prime Mover for Core Flow

The primer mover for core flow will either be a helium circulator should the NGNP reactor be an indirect cycle plant or the main compressor should a direct cycle power conversion system be selected. It either case, the prismatic reactor has a significant advantage due to the core's relatively low flow resistance (55 kPa @ 264 kg/s). This translates into approximately 15 MWe of circulator power or about 30 MWth . The flow resistance of the pebble core is significantly greater by a factor of 2-3. This pressure drop translates into much greater circulator power requirements (about 30-45 MWe) or 60-90 MWth out of the direct cycle.

Relative to the question posed by this section, the prime mover which has to pump same amount of flow but develop a factor of 2-3 less head is more readily designed. A key factor is the ability to develop the required pressure head with as simple a machine as possible. In a prismatic reactor, citing the ANTARES circulator as example, a one stage machine is feasible. It is not clear if a multi-stage machine would be required for a pebble core circulator (assuming an indirect cycle configuration), nevertheless, the pebble circulator faces harder duty due the high pressure head it must develop. Should a direct cycle be employed, the same concerns remain, except that additional stages of compression will be required in the pebble option in order to overcome the higher core pressure drop.

6.11 Schedule, Plant Layout

Table 6-17 below and the discussion in the subsequent text examine schedule and plant layout aspects of the reactor types.

Table 6-17: Schedule /Plant Layout

PLANT LAYOUT/SCHEDULE	Prismatic Reactor	Prismatic Rating	Pebble Reactor	Pebble Rating
Schedule	Less Advanced (relative to PBMR)	o	More Advanced (due to PBMR Demo Plant Lead)	+ +
Plant Layout and Construction	Similar	o	Similar	o
Overall Rating Plant Layout/Schedule		o		+ +

6.11.1 Schedule

A main objective of the NGNP project is for the demonstration plant to commence initial operations by 2018 and to be fully operating by 2020. The operative question, for the purposes of this study, is this:

Does one reactor type offer a schedular advantage for NGNP over the other?

And, secondly:

Does this advantage apply to the commercial version of the reactor?

On a single unit basis, the pebble reactor plant (400 MWth) is smaller, hence, it should take less time to build than the larger prismatic reactor plant (600 MWth). This is, of course, a much too simplistic view. For example, as previously shown in Table 6-7, the PBMR vessel is comparable in size to the ANTARES vessel. Another differentiator is number of plant systems and, here, the prismatic reactor may have a slight advantage with fewer systems. However, counting systems is not very accurate; there are many factors that affect the schedule, system size and complexity included.

At this juncture, therefore, it is more appropriate to consider those long term items that will be responsible for driving the schedule. It is also important to note again that the scope of this study is limited to "the reactor vessel and within." Hence, the schedule issues discussed are therefore limited as well.

Licensing

The NGNP will by licensed by the US NRC. The corresponding licensing framework is still under consideration but the goal is to ultimately have non-LWR technology such as HTRs be licensed under the technology neutral framework (10 CFR 53). However, the NGNP, as part of licensing demonstration process for Part 53, may be licensed under 10 CFR 50 as a basis for future applicants to use Part 53.

Both the prismatic and pebble technologies are capable of being licensed by NRC; however, the level of difficulty may be more for the pebble reactor than the prismatic; the difficulties arising from the random nature of the core and the qualification of the fuel. However, there are other licensing issues that are common to both options, such as containment versus confinement or reduction in emergency planning requirements. It will be the resolution of these common issues that will drive the licensing part of the overall schedule. Hence, neither reactor type is judged to have a particular schedule benefit or disadvantage on the licensing front.

Fuel Development

As mentioned previously, PBMR's fuel development strategy (i.e., recreate German quality fuel) is perceived to have a schedule benefit; however, the development of prismatic fuel is the required timeframe for NGNP is also possible. Because the safety case for HTR technology is to a large extent, singularly reliant on the performance of TRISO particle fuel, the level of scrutiny to be afforded to the issue of fuel qualification by the regulator that is anticipated will belay any schedule advantage.

Reactor Vessel

The reactor vessel is the component which needs to most lead time in which to design, procure materials, fabricate and install. As seen in Table 6-7, both the prismatic and pebble reactor vessels are of comparable size; and, based on this size, shipment of a fully fabricated vessel to the INL site is not feasible. Hence, on-site fabrication will be required. At least for NGNP, it would appear that neither reactor type as currently envisioned (i.e., with respect to power level) offer neither a clear advantage nor a clear disadvantage.

With respect to commercialization, reactor vessel size does matter. For sites with water access, transporting a fully shop-fabricated vessel is possible; however, the slightly smaller flange diameter of the pebble reactor vessel may offer it access to more sites than the larger diameter prismatic vessel. Nevertheless, this is not viewed as an important differentiator.

Previous Licensing/Construction Experience

From a historical perspective, the pebble bed reactor may offer a slight advantage in schedule achievement because other countries (Germany) have more recent building experience (i.e., late 1970s/early 1980s vs early 1970s) that can be drawn on as opposed to Fort St. Vrain, assuming that the age of its construction experience will not probably render most of it moot. Both reactor types will have to rely on the resurgence of nuclear plant construction and take the appropriate lessons that will evolve from that experience. With respect to licensing, only Fort St. Vrain was licensed by the USNRC; however, because it was issued a Class 104 license (i.e., demonstration reactor), the value of its licensing experience with respect to NGNP is limited.

PBMR Demonstration Project

It would be remiss not to mention PBMR (Pty) Limited's ambitious program in South Africa. Under development since 1993, the PBMR project entails the building of a demonstration reactor project near Cape Town and a pilot fuel plant near Pretoria. PBMR's current schedule is to start construction in 2008 and for the demonstration plant to be completed four years later. The fist commercial PBMR modules are planned for 2016. Furthermore,

Furthermore, PBMR is actively pursuing licensing activities with the NRC and are planning to submit an application for design certification in the 1st quarter of 2008. In support of their project, PBMR has embarked upon a significant series of pre-application licensing interactions with NRC as witnessed by numerous public meetings; especially, the PBMR Technology Familiarization Sessions held with NRC in February and March of 2006.

Clearly, the fact that PBMR's design for its demonstration reactor is far along, its fuel development program well underway, and that actual construction experience may be gained prior to NGNP cannot be ignored. Hence, the pebble reactor option must be credited for the experience that PBMR will gain for it.

Based on the above, it is judged that, due to PBMRs program the pebble reactor option does have a moderate schedule advantage over the prismatic option at this time. That advantage may wax or wane depending on the progress of actual PBMR demonstration plant construction. This advantage applies primarily to NGNP development but does not apply in the commercial case.

6.11.2 Plant Layout and Construction

These following attributes are assessed in terms of impact on Plant Layout and Construction.

A. Construction Complexity

B. Constructability

C. Construction while Operating an Existing Plant

Construction Complexity

Are there features in the plant, which will create transportation issues during construction?

It is assumed that the reactor vessels for either type of reactor will have to be fabricated on site. The pebble bed has a slight advantage due to smaller its smaller diameter vessel. However, the pebble bed also has a disadvantage due to the complexity of the pebble handling system. The prismatic reactor has less equipment, and thus will be the least complex in terms of construction. This may also translate into quicker construction even though it may take longer to place the prismatic core blocks versus filling the pebble bed core with pebbles.

Constructability

Factors that affect constructability include: complexity, and number of pieces, of equipment required for the functioning of the plant; number of units constructed already; operational experience; potential for modular construction; and, estimate of bulk quantities required for construction.

Prismatic type reactors have less equipment, and thus are more likely to have a shorter procurement and construction duration. Since prismatic reactors have less equipment, they will also require less bulk quantities.

Although pebble bed reactors are smaller, they are more complex and an 8-unit pebble bed commercial plant is more likely to have longer construction duration than a 4-unit prismatic commercial plant.

Both the pebble reactor and prismatic vessels are comparably large components, which will require specialty cranes and equipment for transporting at the plant site during final installation. Otherwise, minimal transportation issues are expected during construction.

In general, there will be some difficulty in fabricating the reactor vessels of either type of reactor on site. The materials required to fabricate a vessel to operate at very high temperatures will be expensive and require long lead procurement items regardless of reactor type.

Construction while Operating an Existing Plant

The constructing activities at an operating nuclear plant will have to be analyzed and evaluated with respect to the safety of the operating reactor. The added scrutiny will undoubtedly complicate the work process and lengthen the construction schedule of additional reactors once the first reactor is operational. This difficulty will be present regardless of the type of reactor chosen. However, the pebble bed reactor will be more susceptible to these delays because the first plant will be operational sooner and would require the addition of 7 more units, versus 3 more for the prismatic. Furthermore, for the commercial plant, the prismatic will probably have an advantage because of the reduced complexity in building a 4-unit plant versus a 8-unit plant, assuming the result is the same power output.

Based on the above, with the exception of requiring less units in a multi-unit setting, it is judged that there is no distinct plant layout and construction advantage or disadvantage associated with either the prismatic or pebble bed reactor options.

6.12 Non-Proliferation, Safeguards, SNM Accountability

There are no 100% proliferation-proof nuclear systems but all nuclear systems feature a relatively high resistance to proliferation, provided that comprehensive and efficient international controls can be implemented. Institutional measures to address proliferation resistance are of key importance. Both prismatic and pebble technologies are no exception in this regard because within the context of internal and external controls, they are highly proliferation resistant. Table 6-18 below presents a summary followed by suitable discussion.

Table 6-18: Non-Proliferation etc

NON-PROLIFRATION SAFEGUARDS & SNM ACCOUNTABILITY	**Prismatic Reactor**	**Prismatic Rating**	**Pebble Reactor**	**Pebble Rating**
Material Diversion Risk	Lower	**+**	Low	**-**
Institutional Diversion Risk	Lower	**+**	Low	**-**
Material Tracking	Easier	**++**	Harder	**-**
Overall Rating Non-Proliferation etc.		**+**		

6.12.1 Non-Proliferation

Fuel Cycle Front End

Regardless of the technology, the front end of the fuel cycle is least resistant. The source at the uranium mine appears to be the weakest point largely because it is less difficult for a potential proliferant state to obtain covertly natural uranium as opposed to low enriched uranium (< 20 %) from a facility under international safeguards. In either case, the proliferant state needs anyway an enrichment step (a major obstacle) to obtain weapons grade material. Under these conditions, to start from more or less enriched uranium does not make much difference.

Fuel Cycle - Reactor Operations

Safeguarding fissile material from diversion will provide some different challenges for the PBMR. The small size of a fuel pebble makes theft easier, and the large number of pebbles makes inventorying spent fuel at the fuel pebble level very difficult. Once stolen, a pebble could easily be used as a radiological dispersion device (dirty bomb) or an improvised exposure device.

Theft of fissile material in a prismatic reactor is much more difficult. First, the fuel prisms weigh approximately 120 Kg each; hence, special lifting equipment would be required. Second, shielding the prism, due to its much great radionuclide content, would be very difficult. And, third, refueling is conducted every 18 months which severely limits the theft opportunity.

In a diversion scenario, extracting usable fissile material from fuel pebbles or prisms would be unattractive and very difficult. The initial enrichments are still low and the high burn-ups achieved result in much degraded plutonium concentrations. However, consideration would have to be given to alternate approaches of diverting fissile material such as cycling depleted uranium spheres through the core to breed plutonium. This situation would be unique to the pebble reactor more so than a prismatic reactor. A compensating fact, however, is that a significant number fully irradiated pebbles (~100,000) would be required to amass enough plutonium for a weapon and the time element to collect the diverted pebbles would also be significant. Should depleted uranium pebbles be used, the numbers to divert is much less (~10,000) but the time element factor would be about the same.

Fuel Cycle Back End

One of the main advantages of HTR with regard to the resistance to proliferation, is that there is no operational fuel reprocessing technology available to day. Therefore, a country wanting to proliferate with this kind of reactor fuel (if the plutonium route is chosen by this country) should have to develop a specific technology for that. This does not represent an insurmountable difficulty, but this would need a minimum of skill, knowledge and of course, enough time and money.

It is to be noted that for the case of pebble bed reactors these conclusions could be modified because of very specific characteristics of their fuel and because of the loading / unloading mode of this fuel. However, one can say that the apparent drawback of this on-line refueling mode (from proliferation resistance point of view), could be compensated by the fact that it would be necessary to divert or steal (and reprocess) several hundred thousands of pebbles to obtain enough weapon grade plutonium for the making of a nuclear explosive device.

6.12.2 Safeguards and SNM Accountability

The key differential between prismatic and pebble bed reactors regarding accountability it the level of tracking.

In the prismatic option, individual prisms are numbered and tracked. This also means that its constituent parts, particle fuel and compacts will also be tracked on a lot-wise basis similar to that used today in current LWR fuel manufacturing. Hence, the problem of tracking SNM at a prismatic reactor plant will be one of tracking several thousands of prisms at any given time.

Conversely, the situation for the pebble reactor is much different. Individual pebbles are not uniquely identified which makes tracking individual pebbles extremely difficult. Consider that at any given time, a pebble bed reactor will have 460,000 fuel pebbles in the core and a through-put of about 180,000 pebbles per year. For the envisioned 10-module site proposed by PBMR, this translates to an on-site inventory of nearly 5-million fuel spheres and 4-million pebbles in transit (fresh fuel in, spent fuel out). The logistics associated with tracking pebble inventory and demonstrating SNM accountability will be challenging.

From another perspective, it is also likely that a means of identifying and tracking individual pebbles will be required by the regulator. This is necessary to assure that fuel failures and nonconforming conditions can be tracked back through manufacturing so the extent of condition can be assessed and corrective actions taken.

6.12.3 Plant Security

Both reactor types will need to meet the plant security regulations (10 CFR 73). Normal security (gates, guards, & guns) requirements do not discriminate with respect to reactor type. The maximum credible design basis threat may impact the each option differently; however, without knowing DBT details and considering design status, each option must be viewed as equally capable of meeting survivability requirements.

6.13 Post Accident Behavior

As previously discussed in Section 6.9, HTR have many characteristics that make them inherently safe – i.e., they make it highly improbable to have catastrophic core damage (i.e., meltdown) and a corresponding release of a large amount of radioactivity. Both the prismatic and pebble reactor options share these attributes. It follows, therefore, that both options will display acceptable post-accident behavior with respect to both regulatory requirements and investment protection considerations. Obviously, each option’s power level has been optimized to meet these requirements.

Several items worth mentioning are shown in Table 6-19. These are air ingress, water ingress, reactivity excursions, and conduction cool-down response.

In the event of an air ingress event, the pebble bed reactor is more susceptible oxidation issues because of the lower graphitization temperature of the fuel (due to pebble fabrication process limitations). Conversely, the graphite blocks comprising prismatic fuel are fully graphitized because of the absence of fuel kernels at that stage of their fabrication. Hence, the prismatic blocks (which fully encompass the fuel compacts) are more resistant to oxidation.

As configured, neither reactor option as currently envisioned is coupled to a steam cycle; hence, there is little susceptibility to water ingress. This does not preclude coupling the NGNP to a steam cycle. Past experience with water ingress events at AVR, Ft. St. Vrain, and THTR are more of an operational than a safety concern. However, NRC’s review of Fort St. Vrain operational experience attributed chronic water ingress and the resulting corrosive atmosphere [10] as a potential cause of partial control rod insertion event.

Prismatic reactors need additional fuel material versus the pebble reactor in order to achieve an 18-month cycle length. The reactivity this additional fuel material represents is managed by the use of burnable poisons such that the net reactivity in the core is approximately constant through out the cycle. Prismatic reactors may also have control rods directly within the active fuel region; however, inadvertent rod withdrawals and rod ejection accidents are accommodated through design. Furthermore, some excess reactivity will be needed in both reactor types to be able to override post-trip Xenon buildup on restart.

Finally, the depressurized conduction cooldown event is the limiting design basis accident for both reactor types. Each option's power level has been optimized to meet fuel temperature limits following this event. Hence, both prismatic and pebble reactor responses are, by design, similar.

Table 6-19: Post-Accident Behavior

POST ACCIDENT BEHAVIOR	Prismatic Reactor	Prismatic Rating	Pebble Reactor	Pebble Rating
Behavior of reactor systems and fuel during and after key accident conditions				
Air Ingress/Oxidation Issues	Less Susceptible	**+**	More Susceptible	**-**
Water Ingress	Low Susceptibility	**o**	Low Susceptibility	**o**
Reactivity Excursion	Similar	**o**	Similar	**o**
Conduction Cooldown Events	Similar	**o**	Similar	**-**
Overall Rating Post-Accident Behavior		**o**		**o**

6.14 Comparison Summary Results

The results of the previous sections are summarized in Table 6-20. As previously mentioned, a simple, qualitative rating scheme was applied as follows:

- o No clear advantage or dis-advantage (neutral tone shading)
- \+ Weak or small advantage (light green shading)
- ++ Moderate advantage (bright green shading)
- +++ Strong advantage (dark green shading)

Additionally, the discriminators are listed in order of the degree of potential difference or remark-ability between options combined with the relative importance of the discriminator itself. The overall results show that for a number of discriminators, both options are perceived as equivalent or that the prismatic option has a small advantage. However, for six of the higher ranked discriminators, the prismatic reactor is considered to have a moderate advantage over the pebble bed option. The pebble bed option, wholly due to PBMR project status, is considered to have a moderate schedule advantage. Finally, for the highest ranked discriminator, Performance Capability, the ***prismatic reactor***, due primarily to its power capability, has a ***strong advantage*** over the pebble bed reactor option.

Table 6-20: Summary Results Comparison

Table No. / Discriminator	Prismatic Reactor	Pebble Bed Reactor
6-1 Performance Capability	+ + +	-
6-2 Fuel Service Conditions	+ +	-
6-3 Fuel Qualification & Fabrication	o	o
6.4 Spent Fuel Disposal & Reprocessing	+ +	-
6.5 Fuel Handling and Refueling	+ +	-
6-6 Economic Factors	+ +	-
6-8 Research and Development Difficulty	o	o
6-9, 10, 11 Core Design Issues	+ +	-
6-12 Maintenance Issues	+ +	-
6-13 Operational Considerations	+	-
6-14 Safety and Licensing	+	-
6-15 Mechanical Components	o	o
6-16 Plant Layout/Schedule	o	+ +
6-18 Non-Proliferation, Safeguards, SNM Accountability	+	-
6-19 Post-Accident Behavior	o	o

7.0 RECOMMENDATIONS

The DOE should select the ***prismatic reactor*** for the NGNP because it represents the best technological foundation for a commercially attractive, multi-use high temperature reactor concept.

Furthermore, a commercial GEN-IV HTR based on prismatic reactor technology is more likely to be embraced by the US Nuclear Industry because it represents less of a paradigm shift because it will be operationally familiar to prospective owners. The prismatic HTR is very much analogous to an LWR except that the coolant is helium instead of water.

Past experience in the US with the Fort St. Vrain reactor more that adequately demonstrated the feasibility of the prismatic core concept! The plant was licensed by the USNRC and the core itself operated satisfactorily.

In summary, the prismatic reactor offers the following key advantages over the pebble reactor alternative:

- Greater economic potential
- Higher power level and passive safety
- More useable power
  - i.e., less parasitic power loss
- Greater design flexibility
- Higher degree of license-ability
  - Concept previously licensed (FSV)
- Higher degree of predictability
  - Core performance
  - Scheduled outages
  - Less chance of forced outages

Based upon the above and the assessment provided in Sections 5.0 and 6.0, AREVA recommends the ***prismatic reactor*** for the NGNP.

8.0 REFERENCES

1. Idaho National Laboratory, Document ID No. 3963, “Statement of Work for Preconceptual Engineering Services for the Next Generation Nuclear Plant with Hydrogen Production,” Project No. 2384, July 26, 2006.

2. INEEL/EXT-03-01163, “Next Generation Nuclear Plant – High-Level Functions and Requirements,” Idaho National Engineering and Environmental Laboratory, September 2003.

3. Keuter, D.R., Vice-President, Entergy Nuclear, “NuStart Energy: Re-establishing the Nuclear Option,” presentation at the American Nuclear Society’s Annual Conference, San Diego, CA, June 2005.

4. Kemm, K, “Development of the South African Pebble Bed Modular Reactor System,” Uranium Institute, 24th International Symposium, 1999.

5. Sauerwein, J., “Fuel Design, Manufacturing, Fuel Specifications, and Performance,” presentation at the American Nuclear Society’s Advanced Gas Reactor Technology Course, San Diego, CA, June 2005.

6. Kadak, A. C. Modular Pebble Bed Reactor High Temperature Gas Reactor MIT, presentation at the American Nuclear Society’s Winter Meeting. Washington, D.C., 2002.

7. Pohl, P., Wahlen, E., AVR GmbH, “AVR Operational Experience, Overview,” July 5, 2001.

8. INEEL/EXT-03-00870 Rev.1, “NGNP Point Design – Results of the Initial Neutronics and Thermal-Hydraulic Assessments during FY-03,” Idaho National Engineering and Environmental Laboratory, September 2003.

9. INSAG Technical Report CGE 2005-01 Revision 2, May 4, 2006, “Key Issues Affecting Advance High Temperature Gas Reactors.” Ellis W. Mershoff, Cornfed George Enterprises, LLC. May 4, 2006. Prepared for: The International Atomic Energy Agency, Division of Nuclear Installation Safety,Vienna, Austria.

10. GA-A21925, A.M. Baxter et al, “FSV Experience in Support of the GT-MHR Reactor Physics, Fuel Performance, and Graphite,” November 1994.

11. Baumer, R., Kalinowski, I, Hochtemperatur-Kernkraftwerk GmbJ, “Eleventh International Conference on the HTGR, Dimitrovgrad, 19-20 June, 1989.

12. USNRC, Note to Commissioner Assistants, August 17, 2001, “Summary of the NRC Delegation Visit to Germany on Safety Aspects of High Temperature Gas Cooled Reactor Design Technology,” Attachment 2, “Safety Aspects of HTR Technology.”

13. GA Report No. 910720 Rev. 0, GT-MHR Conceptual Design Description Report, January 1995.

PRECONCEPTUAL DESIGN STUDIES REPORT

APPENDIX B2

(Issued Previously as 12-9045442-001)

NGNP with Hydrogen Production Power Level Special Study

April 2007

BEA Contract No. 000 60209

AREVA NP Inc.

AREVA

Record of Revisions

Revision	Date	Pages/Sections Changed	Brief Description
000	3/13/2007	None.	Original Issue
001	4/18/2007	Pages 8, 41 and 42	Corrected minor typographical errors on Pages 8 and 41. Pages 41 and 42 - added paragraph at the end of Section 6.3.2 to address NGNP hydrogen process loop test configuration.

Table of Contents

List of Tables

List of Figures

1.0 Summary

A Special Study has been conducted to determine and justify a recommended rated thermal power for the NGNP prototype reactor and its associated demonstration hydrogen production facility. This study has been accomplished by examining a collection of topics selected to provide insight into the many aspects and perspectives from which this question can be answered.

Results of this study indicate that the goals of the NGNP project can be best met by designing, licensing, building, and successful operation of the prototype reactor at full commercial scale, that is, with a rated thermal output of 565MW. In addition, the estimated thermal power requirements of a demonstration hydrogen production facility will be approximately ten percent of this value, 60 MW.

2.0 Introduction

The Power Level Study is one of the four preconceptual design studies which the AREVA NGNP Team is performing for INL. This study will establish the recommended rated thermal power for AREVA's preconceptual HTR design based on the NGNP requirements. It will also establish the recommended fraction of this power which will be supplied to the demonstration hydrogen production facility.

This document describes the issues addressed by this study, the approach used to resolve these issues, the key criteria considered, the results of evaluation of these criteria, and final recommendations based on a synergistic assessment of the individual criteria evaluations.

This study is closely related to the NGNP Primary and Secondary Cycle Concept Study which the AREVA team is also performing. These two studies are coordinated, since the outcome of each is influenced by the other. Toward this end, close communication was maintained between the leader of the power level study and the primary-secondary study with guidance from the AREVA NGNP Systems Integration lead who provides overall coordination for all of the special studies.

3.0 Applicable NGNP Goals

The Gen IV Next Generation Nuclear Plant (NGNP) is an advanced reactor optimized to produce both hydrogen and electricity. The NGNP will generate hydrogen without emitting greenhouse gasses or other air pollutants. It will also exhibit high thermal efficiency, attractive safety aspects, minimize waste, and be proliferation resistant. It will be suitable for efficient hydrogen production utilizing, for example, water-cracking by high temperature electrolysis or thermo-chemical decomposition. For prototyping the NGNP, the DOE selected a very high temperature gas-cooled nuclear (VHTR) reactor with the capability to produce process heat, electricity and/or hydrogen.

A key design objective of the NGNP is the elimination of the need for active safety systems to prevent fuel damage in the event of loss of forced cooling. In order to prevent fuel damage, the plant must be designed to passively remove core decay heat via conduction, natural convection, and thermal radiation in this event. The requirement for passive sensible and decay heat removal under accident conditions, places a limit both on core size and on power density for a given core size in order to both limit the stored energy in the core, and facilitate passive heat removal. This results in a low power density core design. In order to achieve significant levels of energy production, the VHTR design concept favors multiple moderate size power reactor modules, typically four or more, which could share common support facilities.

In order to provide a framework within which to make the recommendations called for in this study, it is important to understand the goals of each of the key contributors and users of the NGNP project results. In the sections below are examined the goals of each of these key groups as they relate to the power level of the NGNP demonstration reactor.

3.1 DOE Expectations

The NGNP High Level Functional Requirements[1] document contains several high level functional requirements for the NGNP which must be considered in an analysis of plant power level. These requirements, and a summary of key concepts to be considered for each, are summarized below.

3.1.1 Develop and Demonstrate a Commercial-Scale Prototype VHTR

A commercial-scale prototype will be built, tested, and operated to demonstrate the performance characteristics of future advanced high-temperature reactors. The primary function of this prototype plant will be to verify both operational and safety performance of high-temperature reactors over a range of normal and transient conditions. The NGNP will also demonstrate the ability to generate efficient and reliable process heat.

3.1.2 Obtain Licenses and Permits to Construct/Operate the NGNP

The NGNP will be licensed by NRC under 10 CFR 50 or 10 CFR 52. The licensing of the NGNP by NRC will also demonstrate the effectiveness of licensing future advanced high-temperature reactor concepts for commercial applications. In particular, it is anticipated that many of the current issues associated with NRC licensing of a non-LWR and the use of nuclear power for hydrogen production will be resolved during the licensing of the NGNP.

3.1.3 Develop and Demonstrate Hydrogen Production

Hydrogen production plant(s) will be included as part of the NGNP facility to demonstrate the capability of high-temperature reactors to produce hydrogen in a cogeneration mode and demonstrate hydrogen production. Hydrogen production will be demonstrated using high-temperature water electrolysis and a thermochemical process.

3.2 Commercial (Vendor/User) Expectations

The Design Features and Technology Uncertainties for the Next Generation Nuclear Plant[2] report, developed by the Independent Technology Review Group, identifies commercial expectations as:

> *Selection of the technology and design configuration for the NGNP must consider both the cost and risk profiles to ensure that the demonstration plant establishes a sound foundation for future commercial deployments. The NGNP challenge is to achieve a significant advancement in nuclear technology while at the same time setting the stage for an economically viable deployment of the new technology in the commercial sector soon after 2020.*

4.0 Study Requirements

The "Statement of Work - Preconceptual Engineering Services for the Next Generation Nuclear Plant with Hydrogen Production,"[3]describes the following requirements for the conduct of the NGNP Prototype Power Level Study.

> *The vendor shall prepare a study that evaluates and recommends a power level for the NGNP prototype nuclear system, which is scaleable and meets all the necessary requirements as a "commercial" prototype and is licensable as a commercial prototype. In addition, the subcontractor shall evaluate and recommend minimum optimal prototype hydrogen plant size that will be scaleable to a future commercial scale plant.*

From this requirement definition, three specific, though interrelated, questions are developed to focus this study. These questions are:

- What should be the rated power level of the Nth of a Kind (NOAK) commercial VHTR module?
- Given the desired power level of the commercial VHTR module, what should be the rated power level of the NGNP prototype plant?
- In order to demonstrate commercial scalability of an associated hydrogen production plant, what is the power requirement for a demonstration plant to be associated with the NGNP reactor? The power requirements for the sulfur-iodine and high temperature electrolysis processes should be considered.

5.0 Study Process

5.1 Assumptions

The NGNP design is to be adapted from AREVA's ANTARES commercial HTR design concept, which incorporates a prismatic core, as agreed with INL. For this adaptation, the basic system configuration will remain the same for electricity production, but a separate interface for a high temperature heat transport loop to the hydrogen process will be provided. The NGNP specific power level and system operating parameters will be adjusted based on the results of the relevant special studies performed by the AREVA NGNP team.

5.2 Study Approach and Decision Process

This study has been completed utilizing the basic approach and decision process defined for all of the special studies under AREVA's scope of work. Key aspects of this process are:

- **Establish decision hierarchy** – Define the specific questions to be answered by the study, and develop a list of study criteria and supporting questions.
- **Identify range of options** – Define the range of study parameters to be considered for each of the study criteria, as applicable.
- **Get the required initial input** – Obtain sufficient information to provide a high-level assessment of each of the study criteria.
- **Prioritize decision sequence** – Identify those study criteria that provide information or insights which are critical to answering the main study questions.
- **Assess options regarding each criterion** – Identify which study criteria require additional or more detailed information to provide a complete evaluation.
- **Synthesize results for optimum solution** - Develop an overall set of recommendations based on a synthesis of the evaluations of each of the study criteria.
- **Internal expert review of draft special study results** – Utilizing the AREVA team, obtain expert review of the study and associated recommendations.
- **Finalize special study report** – Document the results of the special study.

5.3 Study Limitations

AREVA team has no scope for H_2 process development, H_2 process plant design, high temperature heat transport loop design and development, H_2 plant R&D, schedule, and risks, and H_2 plant economics (prototype or commercial). This limits the information that AREVA can use regarding these areas and their impact on the questions addressed by this study. In order to address these limitations, the AREVA team will make limited efforts to gather relevant information from available sources. Reasonable assumptions will be made where appropriate.

5.4 Study Criteria

In order to provide the necessary data to answer the three main questions defined for this special study, several study criteria were developed. For each of these criteria, focused questions were provided to guide the evaluation. Answers to these questions were pursued to the extent necessary to develop an overall understanding of the impact of the specific study criteria on the main study questions.

Market View

- What are the projected markets for the commercial VHTR module?
- For each of these markets, what is the optimum power level of the VHTR plant?
- For those applications where modularity may be advantageous, what is the optimum power level of each reactor module?

In developing the answers to these questions, end uses considered included electricity generation, hydrogen production, process heat production, and appropriate combinations of these uses.

Core Neutronics

In order to fulfill the top-level requirement that the VHTR plant being developed meets the passive safety goals of a Generation IV reactor, operating core power densities will be limited and rated reactor power changes will require changes in core size and geometry. Given this:

- What are the practical limits on core geometry (diameter, height, configuration – cylindrical vs. annular) from a neutronics/reactor physics standpoint?
- Considering these limits and representative power density limits, what are the practical power limits of a VHTR core?

Licensing Issues

It is a top-level goal of the NGNP program that the prototype NGNP reactor will be licensed by the NRC using a process consistent with that used to license commercial nuclear power plants.

- What is the optimum power level for the prototype NGNP module, as a fraction of the desired commercial VHTR module power level, to maximize the portability of the NGNP prototype licensing experience to the commercial plant?
- As this power fraction changes, what is the expected impact on the portability of the licensing experience in key licensing areas? What areas are most and least impacted?

Demonstration of Passive Safety Features

One of the primary purposes for constructing the NGNP prototype is to provide an opportunity to demonstrate, and generate data regarding, the passive safety features of this reactor type.

- What are the key plant power-related operational parameters that influence the performance of the reactor passive safety features?
- What is the optimum value of these parameters for the prototype NGNP module, as a fraction of those of the commercial VHTR module, to generate useful and applicable data and operational experience with these safety features?
- As this power fraction changes, what is the expected impact on the usefulness and applicability of the data generated for each of these key parameters?

Fabrication Issues

- Which major reactor components present challenging fabrication issues?

- For those components and associated issues, which are functions of rated reactor module power and what is the nature of that functionality?

Component Feasibility

There are several reactor components, the performance of which may limit the power of a VHTR module. These components include the reactor vessel, the intermediate heat exchanger, and the helium circulator.

- What other components fit into this category?
- For each component, what is the limiting consideration and how is it related to rated reactor power?
- Must this component be unique within a reactor module, as with the reactor vessel, or may separate trains be utilized to mitigate the performance issues?

Plant Flexibility and Operability

- For those applications that utilize multiple power output types (i.e., production of electric power and hydrogen) what is the optimum relationship between the outputs to achieve acceptable operational flexibility and plant availability?
- For any power output type or combination, are there module rated power levels which present particular operational flexibility or availability challenges, for example plant startup or load following operations?

Plant Safety Limits

Several plant safety limits and related operational parameters are directly related to the rated power of the reactor. These limits include conduction cooldown limits, fuel temperature limits, core flow and flow bypass limits, and core mechanical performance limits.

- How does the reactor module rated power influence these operational and off-normal limits?
- In each case, what are the key parameters which relate the reactor power to the limits?
- Are there any threshold powers above which it is impractical to meet these limits?

Economic Considerations

One of the benefits of constructing the NGNP prototype is to provide an opportunity to benchmark key cost data for this reactor type, including construction costs, capital equipment costs, fuel costs, and operation and maintenance costs.

- What are the key plant operational parameters that influence the usefulness of this economic benchmark data?
- How are these parameters related to the commercial VHTR module rated power, the NGNP prototype module power, and the difference between these power levels?

Hydrogen Plant Process Heat Requirements

One of the top-level requirements for the NGNP prototype reactor is to support the operation of a hydrogen production plant. The design of this plant is to be sufficient to demonstrate scalability of the process to commercial size. For the purposes of this evaluation, the hydrogen production process is assumed to be either the sulfur-iodine chemical process or the high temperature electrolysis process.

- What are the expected power requirements for the commercial scale processes.
- What are the expected power requirements for the demonstration scale processes that must be supported by the NGNP prototype reactor?
- What is the current state of development of each of these processes, with respect to power requirements, and what is the prognosis for development to the required demonstration scale on a schedule consistent with deployment of the NGNP prototype reactor in 2018?

Research and Development

- What impediments exist to successful design, fabrication, and operation of the NGNP at the chosen power level?
- What R&D opportunities are presented by design, fabrication, and operation of the NGNP at the chosen power level?

6.0 Evaluation of Study Criteria

Upon review of the initial evaluations of each of the study criteria, several were identified as being critical to the establishment of the recommended commercial reactor and NGNP prototype reactor power levels. These are designated Key Discriminating Criteria and were subject to additional study and evaluation. The results of these evaluations are presented first in the following sections. The evaluation results for the remaining study criteria are presented following the Key Discriminating Criteria. The results of the Hydrogen Plant Process Heat Requirements evaluation are presented separately because it was determined that, while the relatively small power requirements for the demonstration hydrogen plant were not critical to determination of the NGNP power level, the results of the evaluation are required to answer the third of the power level special study questions.

6.1 Key Discriminating Criteria

6.1.1 Market View

The most likely commercial applications for the VHTR will entail the use of process heat and electricity in various combinations. A principle anticipated commercial use of the energy from the VHTR will be the production of hydrogen. However, AREVA internal studies have also identified a number of other potential industrial applications for the energy products from the VHTR. These applications are introduced below and more fully developed in following sections.

6.1.1.1 Hydrogen Production

The long-term vision for the VHTR is to be an integral part of the Hydrogen Economy. The primary reason for the US DOE sponsorship of the Next Generation Nuclear Plant Initiative is to support this vision. From a market standpoint, government funded initiatives will be the primary market driver in the near-term.

In framing the commercial VHTR's market potential, there is a need to gain a better understanding of its potential role in a hydrogen economy. A part of this task is to ascertain which hydrogen production technologies show the most promise. As the most potentially promising methods are currently at the research/development stage, this assessment, of necessity, remains somewhat speculative. However, it is possible to report the favored production processes and project the required energy characteristics, particularly temperature, to drive these processes. From this information, energy production requirements for hydrogen generation utilizing the VHTR can be postulated.

Power and Production Process Configurations for Hydrogen Production[4]

Hydrogen can be produced by thermo-chemical, electro-chemical, and hybrid (electro-thermo-chemical) processes using nuclear energy as the primary thermal energy source. The hydrogen production process properties determine the types of reactors that can appropriately be coupled to the relevant hydrogen production technology. Some processes require both electrical and thermal energy, and, therefore, for such applications, plants readily configured for co-generation are attractive.

An important design requirement for both thermo-chemical and electrochemical hydrogen production is the relatively *high temperature* needed for achieving high thermal-to-hydrogen energy efficiency. This is an important factor in the economics of the technologies. Furthermore, each hydrogen production process, and the nuclear system supporting it, has unique technological features that can significantly influence the economic compatibility of the system with the projected hydrogen markets.

Energy from the VHTR can be used in hydrogen production mainly in three ways:

- By using the electricity from the nuclear plant for conventional *liquid water electrolysis*.
- By using both the high-temperature heat and electricity from the nuclear plant for *steam electrolysis*.
- By using the heat from the nuclear plant for pure *thermochemical processes*.

Water electrolysis is already commercialized, however, is comparatively inefficient. It is unlikely that water electrolysis will become a favored commercial application of the VHTR for hydrogen production. However, high temperature steam electrolysis shows promise in that it has comparatively high efficiency and requires only intermediate range temperature –temperatures achievable by the VHTR. Large scale production capabilities, however, remain to be demonstrated.

Thermo-chemical water splitting is the principal process under study for a nuclear powered thermo-chemical hydrogen production. Hydrogen can be produced from nuclear power by thermo-chemical water splitting. (Heat plus water yields H_2 and oxygen.) Thermochemical processes have potentially higher efficiencies and lower costs than the electrolysis of water with electricity. High temperatures, in the range of 750-1000 C, potentially achievable by the VHTR, are required for economically viable production.

6.1.1.2 Industrial Applications Utilizing the VHTR

Commercializing the VHTR requires the identification of potential industrial applications that could utilize the energy output from the VHTR either in the form of process heat, electricity, or various combinations of both. AREVA NP studies have identified a number of such potential industrial applications for the VHTR. These applications (see below) utilize process heat and electricity in various combinations. The first few are ordered by the approximate timeframe of likely implementation, earliest first.

Hydrogen production, already discussed above, is included in this listing in order to suggest the time frame of likely implementation relative to the other industrial applications. (Note also that hydrogen generation may be a significant component of several of these industries.)

<u>Potential Industrial Applications of the VHTR</u>

1. Coal to Liquids

2. Oil Sands
3. Oil Shale
4. Coal Gasification – "Clean Coal"
5. Hydrogen Production
6. Petroleum Refineries
7. Electricity Production
8. Industrial Process Heat Applications
   8.1 Steel
   8.2 Alumina and Aluminum
   8.3 Chlorine VCM and PVC
   8.4 Ammonia and Fertilizers
   8.5 Chemical Platforms
9. Biomass
10. Water Desalination

A discussion of each of these potential applications for the VHTR is provided below. This material represents a summarized composite of AREVA and industry research. Hence, individual citations may not always be provided.

Assessment of Industrial Applications

Coal-to-Liquids

Coal-to-Liquids (CTL) is a promising concept for converting existing large supplies of coal to forms that can be substituted for current petroleum products. Coal is one of the most abundant sources of energy on earth but it suffers from a high cost for transporting it to needed locations, high environmental impact from burning coal and the difficulty of using coal to meet transportation energy needs. A CTL process addresses all of these issues.

A strategy for using nuclear heat in the CTL process is to identify conceptual approaches to using that energy, then to identify specific points in the process where those approaches could be applied. It was determined that the optimum application for the VHTR would be to displace chemical energy. One way is to use the VHTR to produce hydrogen. This offers the following CTL process improvements:

- The water-gas shifter reactors are eliminated
- The carbon monoxide that had been converted to carbon dioxide is now available to make more Fischer-Tropsch feedstock
- The size of the CO_2 removal equipment is reduced

For the purpose of this assessment, the process for producing hydrogen was assumed to be electrolysis which also produces oxygen. This assumption results in a further process simplification in that the oxygen producing equipment otherwise required in the CTL process can be eliminated.

The second nuclear heat application is to recover the tail-gas from the Fischer-Tropsch synthesis rather than to burn it for process heat. Also, the CO_2 can be recovered and converted to carbon monoxide for use in the reaction. These process changes increase the carbon utilization to 95.7%.

Plant sizes consistent with current plans would require approximately 3,000 WMth of nuclear generated heat.

At present, it appears that the nuclear heat option would be competitive with the fossil heat source at a CO_2 penalty of about $100/ton.

Oil Sands

The oil sands in Canada present a sound potential application for the VHTR. Production is expected to increase from today's level of about 1 million barrels per day to a level between 4 and 6 million barrels over the next few decades. The current in-situ methods require about 1,000 standard cubic feet (scf) of natural gas per barrel of bitumen. This natural gas is used only for process heat. An additional 80scf/barrel is used for processing and 250scf/barrel is used for hydrogen production for refining. Using the heating requirements only, one 600 MWth VHTR module could supply the requirements for a 40,000 to 50,000 barrel per day facility. The current oil sands production facilities are being built in 35,000 to 50,000 increments so this matches well with the VHTR capability. Most current oil sands facilities transport steam for heating about 10 km but two transport steam up to 17 km to support a single field. With this range, an oil sands field is expected to be productive for 40 to 60 years. Each of these characteristics matches well with VHTR conceptual design characteristics. The Canadian oil sands could possibly support a large number of VHTRs, providing significant environmental benefits by displacing the use of natural gas, the current plan.

Oil Shale

Current estimates indicate the oil shale deposits in the United States have 800 billion barrels of recoverable oil. While oil shale is found in many places worldwide, by far the largest deposits in the world are found in the United States in the Green River Formation, which covers portions of Colorado, Utah, and Wyoming. The oil resources in place within the Green River Formation are estimated to range from 1.2 to 1.8 trillion barrels. Not all resources in place are recoverable; however, even a moderate estimate of 800 billion barrels of recoverable oil from oil shale in the Green River Formation is three times greater than the proven oil reserves of Saudi Arabia.

Studies have shown that the heat from one 600 MWth VHTR unit can provide the needs of a 100,000 barrel/day facility for 40 years with heat transport no greater than 1,200 m. The oil shale is heated to 370°C which requires a heat source of 450° to 500° C. These requirements match the design profile for the VHTR.

Oil shale production is a significant potential industrial application for the VHTR.

Coal Gasification-"Clean Coal"

Power generation from coal emits significant amounts of sulfur dioxide (SO_2), nitrogen oxides (NOx), mercury and carbon, contributing to numerous health and environmental concerns. The Clean Air Act of 1970 set emission standards, but existing plants were grandfathered. Today, 850 of those plants are still operating, exempt from the 1970 emission standards. Clean coal initiatives will boost businesses involved in efforts to reduce emissions from coal.

Using heat, steam, pressure, and oxygen, coal can be broken down into a relatively clean gas, and a handful of other chemical byproducts. Coal gasification offers one of the cleanest, most versatile ways to convert coal into electricity and other forms of energy. Rather than burning coal directly, gasification breaks down the coal into its basic chemical components. The gasification facility can then co-produce a wide range of products, including electricity, high-value chemicals, and synthetic fuels. It is also important to note that Hydrogen can be produced in the coal gasification process.

The use of the VHTR for the coal gasification process is similar to that described above in the Coal-to-Liquids application above. The CTL process produces a fluid output that is more easily transported than bulk coal. A gaseous product can also be produced and transported but is more likely to be used at the source to generate, for example, "clean coal" electricity. For this reason, initial projects would be more likely to be CTL. A successful CTL application would also lead to interest in using VHTR for coal gasification. It is likely that these facilities would be sized similar to the CLT facilities, requiring on the order of 3,000 MWth for each installation.

If the trend to limit the production of green house gases both continues and, likely accelerates, coal gasification is viewed as a strong potential future market for the VHTR.

Hydrogen Production

One energy vision for the future is to create a stable, non-polluting hydrogen economy. The nuclear role in this economy is to produce the hydrogen that will then be used to fuel the transportation industry. The time scale for achieving this economy is debated but, with a finite hydro-carbon supply and no viable alternatives, a hydrogen economy will come into being someday. The timing will be based on the relative economics of using current petroleum resources compared to hydrogen production costs.

At present, Hydrogen demand is expected to grow at a rate of 4-10% annually for the foreseeable future. The current and future hydrogen market can be characterized as follows:

Current and near-term

- Oil Refining
- Ammonia (fertilizer) industries

  - Methanol industry
  - Merchant H_2 Customers
  - Oil Sands
- Mid-term
  - Oil Shale
  - Coal-to-Liquid
  - Electricity Power Peaking
- Far-term
  - Transportation
  - Remote electricity production

The most likely hydrogen production applications for the VHTR in the near term (next few decades) are as a supplement to coal-to-liquids, oil sands or oil shale production. In each of these processes, the crude product needs to be refined which requires hydrogen.

The demand for "stand alone" hydrogen production, e.g. for the transportation market, is more difficult to predict. This market is unlikely to develop a substantial demand until beyond 2030, which suggest it will not present a significant near term application for the VHTR, but may present a significant industrial application in the long term.

A recent study by Savannah River National Laboratory[5] demonstrated the feasibility of stand-alone nuclear hydrogen production facilities utilizing 600 MWth of process heat with an additional electric demand of approximately 192 MWe from either an additional reactor of from the electric grid.

Electric Production

Historically, economy of scale advantage has generally favored larger plants for production of electricity. However, the VHTR has several attributes that could make it attractive for electric power generation:

- Modular construction leading to shorter construction schedules and reduced construction costs

- Greater inherent safety permitting siting closer to load demand

- Smaller added increments of power to better match load growth and minimize capital outlay

An outline of the market potential based on each of the primary benefits follows:

<u>Short Construction Schedules</u>

The benefits from short construction cycles come from two sources:

1. Reduced interest during construction and

2. Delayed decisions on capital investment.

Scoping calculations suggest that a modular VHTR can compete with a large Gen III plant, on construction costs alone, provided the over-night construction costs do not exceed the costs for the Gen III plant by more than about 10%.

The second benefit of short construction schedules is to provide utilities the ability to delay capital investment decisions. Capital decisions made close to the need date are more likely to closely match generation with demand than decisions with a longer time horizon.

Location Close to Demand

Finding routes for transmission lines is becoming increasingly difficult. This is especially true in densely populated areas. One strategy currently employed is to construct the power generator as close as possible to the demand in order to reduce the need for new transmission lines. The inherent safety of the VHTR could facilitate such siting.

Smaller Increments of Power

One of the advantages of modular VHTR is that power can be added to the grid in smaller increments than with a larger base load plant. These smaller increments better match the load growth.

Modular construction also contributes to the option of progressing incremental capital investment to match electricity demand, i.e, adding individual power modules only when justified by demand. In fact, the market may even be willing to pay a premium for this flexibility.

For the purposes of this study, it is assumed that most electric generation facilities will be constructed in the range of 1000 MWe.

Industrial Process Heat Applications

Five industrial applications that use significant amounts of process heat were identified as potential applications for the VHTR: Steel; Aluminum and Alumina; Chlorine, VCM and PVC; Ammonium and Fertilizers; and Chemical Platforms.
A summary of each potential application is provided below.

Steel

The most viable concept for applying nuclear energy to steel making combines two well-known processes: direct reduction in a shaft furnace and refining in an electric furnace. In this process, iron ore is reduced in the solid condition by a synthesis gas ($CO+ H_2$) derived from steam reforming of natural gas to a product known as sponge iron. The reaction requires high temperatures and heat. The VHTR could he used to provide the heat needed to produce the reducing as for the direct reduction of iron ore and the electricity needed to refine the resulting sponge iron to steel in an electric-arc furnace. Production of steel by electric-arc furnaces is a long-established commercial technology. Electric-arc refining uses about 650 KWH/ton of steel. In a steel making system involving direct reduction and refining in an electric-arc furnace, nuclear energy can be used to:

a. Provide high-temperature heat for the production of a gas suitable for the reduction of iron ore to iron.

b. Produce electricity for operation of electric-arc furnaces to refine the sponge iron.

The most serious competition for the VHTR in this application is presented by fossil fuel. Absent strong pressure to reduce green house gases, if low cost fossil fuel is available close to iron ore reserves, supplying heat from the VHTR may not prove cost effective.

However, it is anticipated that a potential market may well exist in countries which take steps to limit the generation and release of green house gases.

Recent work by the Japan Nuclear Steelmaking Project[6] has focused on a nuclear reactor with a 500 MWth thermal output as a base case.

Alumina and Aluminum

The comparatively low temperature requirement (150°C) for the aluminum production process suggests heat sources used in alumina facilities will based on technologies less sophisticated than VHTR. Therefore, at present, aluminum and alumina manufacturing and processing facilities are not viewed as a strong potential market for VHTR.

However, economic pressure to reduce the generational of green house gases in industrial production processes would likely alter this conclusion. It is estimated that a 600 MWth facility, supporting a 1.2 Mt/y output may be feasible in this case.

Chlorine VCM and PVC

Chlorine is produced from the electrolysis of sodium chloride, using three methods: the mercury cell, the membrane cell (Best Available Technology) and the diaphragm cell. Vinyl Chloride Monomer is produced in two steps by chlorinating ethylene and by its oxychlorination (250°C) into dichloroethane, which is decomposed at 500°C into VCM. VCM is polymerized into PVC, mainly in suspension in water (50-70°C).

These process energy requirements would not fully utilize the capabilities of the VHTR. There may prove to be some advantage if the production of Chlorine-VCM-PVC is part of an integrated chemical platform. However, it is not clear that production of chlorine VCM and PVC presents a significant future commercial application.

Economic pressure to reduce the generational of green house gases in industrial production processes may alter this conclusion.

In order to support a 600 MWth VHTR facility, many related functions would have to be consolidated onto one site.

Ammonia and Fertilizers

Most ammonia is produced by the steam reforming of natural gas. However, it may also be produced by steam reforming of other hydrocarbons. For example, China produces 80% of its ammonia from coal, naphtha and refinery gas through reforming.

About 80% of the manufacturing plants use the catalytic steam reforming of natural gas. Primary methane reforming is highly endothermic and takes place at between 750 and 800°C in the presence of steam. It is fueled with natural gas. The secondary reforming is autogenic and takes place in the presence of air at around 1000°C. This eliminates the remaining methane and introduces into the system the nitrogen necessary for ammonia synthesis. The ammonia is synthesized from the catalytic conversion of hydrogen and nitrogen in an exothermic reaction at temperatures of between 350°C and 550°C.

Even though there is a reasonable match with the process energy requirements, the limited market size suggests that the production of ammonia and fertilizers will not present an attractive opportunity for the VHTR. It is estimated that facility requirements would limit feasible reactor outputs to 200 MWth.

The conclusion may be altered should production of these products be integrated as part of a chemical platform or if there is economic pressure to reduce the production of greenhouse gases.

Chemical Platforms

Base chemical production is generally endothermic whereas complex chemical syntheses are exothermic. Ethane and naphtha supplied from oil refineries are the source of major intermediates such as ethylene and propylene, which are produced in steam crackers using large amount of heat but also produce large amounts of steam. Natural gas is also a major feedstock to produce hydrogen and syngas.

The process heat demand is driven by the accumulation of processes which allow energy optimization and by-product recycling, as well as improved risk management in a given location. Most of the chemical platforms which will develop will be built in transition economies, some in OECD countries, but they will not all reach a demand in the region of 600 MWth.

Absent economic pressure to reduce processes that produce green house gases, chemical platforms appear to present a limited future market for the VHTR.

Biomass

Biomass, in the energy production industry, refers to living and recently living biological material (lignocellulosics) which, after some level of processing can be used as “bio” fuel or for industrial production. Biofuels include bioethanol, biobutanol, biodiesel & biogas. Biodiesel and biobutanol are direct biofuels and can be used in petroleum engines.

The most promising possibility for integration of the VHTR may be through indirect biomass-to-liquids approaches utilizing gasification. Thermochemical processes, including pyrolyis and gasification, employed in processing lignocellulose utilize heat in the range 400 - 850°C and are the best candidates for such integration.

It should be noted that the general economic viability of biomass fuel is controversial with experts in disagreement. For example, some believe that biomass-to-ethanol via processing lignocellulose results in a net energy deficit for the conversion process.

Government incentives may be necessary to attract business investment if there is, in fact, a net energy deficit in converting biomass-to-ethanol for fuel. This application, absent other incentives, appears to afford little opportunity for the VHTR.

However, should such incentives develop the most likely potential commercial application of the VHTR is that of indirect biomass-to-liquids approaches utilizing gasification. Other options include nuclear energy-supported integrated bio-refineries that utilize reduction of the byproduct CO_2 to liquid fuels to displace petroleum and generate additional carbon credits. Future developments in energy densification of the feedstock that allow larger processing facilities will improve the options for efficient integration of a 600 MWth VHTR with biomass conversion processes.

Water Desalination

Desalination technologies have achieved commercial, world wide application. While many are fossil powered, there are numerous nuclear powered desalination applications as well.

The International Atomic Energy Agency (IAEA) has studied, and continues to study, the nuclear desalination option. The IAEA results generally show that nuclear seawater desalination yields costs in the same range as fossil options. However, this conclusion is generally derived from data for large base loaded nuclear plants.

Large-scale deployment of nuclear desalination on a commercial basis will depend primarily on economic factors. Such economic factors may be very much region specific. For example a market may exist in an arid region with high water demand but with limited access to low cost fossil fuel.

Perhaps a water starved region enjoys (1) plentiful sunshine, (2) a high electricity demand during day light hours and (3) low electricity demand at night. In this instance, solar conversion systems could provide fresh water during the day, with the nuclear plant providing electricity in the same period. However, in darkness the “off-peak” (excess) electrical power could flow to the desalination plant during the night when the solar desalinization plant would otherwise be idle.

The modularity of the VHTR would favor this design for construction in certain remote locations, though required modules would likely be in the rage of 100-400 MWth. The high thermal efficiency of the plant, particularly in the co-generation configuration, would also favor the VHTR in arid regions with its reduced cooling water requirements over the Gen III plant designs. However, unless it were possible to uniquely match the advanced design capabilities of the VHTR (particularly the high temperature outputs) to a unique combination of electrical and thermal demands, it is improbable that it would prove cost effective for water desalination.

6.1.1.3 Market View Summary

The Gen III commercial impetus for very large individual reactors to optimize the investment in plant and fuel, and minimize electric power production cost is substantially altered by the Gen IV objective of passive core cooling post accident. Even given the improved thermal efficiency of the VHTR, the passively safe design may compromise the cost of electric power production when contrasted with the Gen III water cooled reactors. Thus cost optimization for the VHTR must take a different path -modularity.

The Gen IV safety objectives encourage the development of multiple “modular” reactors within a single physical facility. An important ancillary benefit of the modular concept is a reduction in the high initial capital investment typically demanded for the large Gen III units. With the VHTR, individual reactor modules can be added to match energy demand, pacing the capital outlays.

Modular construction can also take advantage of the economies of production, relying less on the economies of scale to reduce costs. It is envisioned by some that a large contribution to the cost effectiveness of modular facilities is the ability to manufacture the major component parts in a “factory” environment, shipping these subassemblies to the site to be assembled. The factory environment facilitates both careful control of the manufacturing process and reduction of production cost.

Commercial applications thus far identified for the VHTR do not appear to establish constraints on reactor module size. That is, there does not appear to be a significant advantage to producing modules with power levels below that which is limiting based upon other criteria. Thus the optimal size for a commercial module is the largest capacity permitted within the design constraint of passive safety and, if “factory built,” it is also constrained by the largest pre-assembly structures that can be cost-effectively transported to the plant site. There is an additional assumption built into this conclusion, namely, that existing experience regarding economies of scale are applicable and it is not cheaper to build many smaller units than one large unit of comparable size.

Table 6.1 – Summary of Market View Evaluation

Market	Standard Plant Heat Input MWth	Output	Comments
Coal to Liquids	3000	26,000 b/d	Competitive with fossil fuel options at approximately $100/ton carbon credit
Oil Sands	600	40,000 /d	Study configuration for 1 VHTR module
Oil Shale	600	100,000 b/d	Study configuration for 1 VHTR module
Coal Gasification	3000		Based on coal to liquid results
Hydrogen	600	100,000 t/y	SRNL study configuration for 1 VHTR
Petroleum Refineries	1800	15 Mt/y	
Electricity	2400	1000 MWe	Assumed default configuration
Steel	500		Japan Nuclear Steelmaking Project
Alumina/Aluminum	600	1.2 Mt/y	Low temperatures limit VHTR market
Chlorine/VCM/PVC	600	1 Mt/y PVC and 0.6 Mt/y Cl	Marketability requires consolidation of multiple functions on one site
Ammonia	200	0.75 Mt/y	
Chemical Platforms	200-600		Based on combination of processes to gain efficiency advantages
Biomass	600	1 Mt/y feed	Market depends on Carbon credit
Water Desalinazation	100-400		Low temperatures limit VHTR market

6.1.2 Economic Considerations

It is argued in the preceding section of this study that the optimum size VHTR module from an applications view point is the maximum module size imposed by design constraints (passive cooling, for example) not the size of the industrial facility that it is intended to supply with electricity or heat. The demand size can always be scaled so as to fully utilize the supply size.

There are several additional economic considerations that apply specifically to the planned size of the NGNP reactor. These considerations can be viewed from the perspective of the reactor vendors who will eventually participate in the NGNP project and the end-users who will purchase the commercial units built using the NGNP experience.

From a reactor vendor perspective, participation in the NGNP project, particularly the latter stages where significant sharing of the costs is anticipated, depends on a favorable balance of these cost with the benefits gained through such participation. Some of these benefits are difficult to assess from an economics standpoint, such as the perception of industry leadership gained through participation. Others are easier, particularly those related to the applicability of costs incurred for the NGNP that are directly transferable to the commercial fleet. Chief amongst these benefits is the ability to complete first-of-a-kind engineering tasks in a cost share manner. This benefit is maximized if the power level of the NGNP reactor is equal to the power level of the eventual commercial plant.

Any difference in power level will reduce this benefit. The reduction in benefit will increase dramatically as the power levels diverge. Thus, from a reactor vendor standpoint, there is considerable incentive to have the NGNP built as a full size demonstration plant.

Deployment of the NGNP provides an opportunity for eventual end-users to benchmark key cost data that will aid the decision making process. These costs may include capital cost data, construction costs, costs of operation and maintenance, and fuel cycle costs. Design of the NGNP at any power level other than 100 percent of the commercial plant will make these benchmarks less directly applicable, thus less useful.

From this argument, it follows that the NGNP prototype should be a full-rated design, though it may initially be licensed to some fraction of that design capability. While distortions in pricing may result from a first of a kind vs. N^{th} of a kind installation, this approach will nonetheless present the best opportunity of achieving these cost benchmarks. These benchmarks will play a key role in assuring the eventual commercial acceptability of this reactor technology.

Note that any first-of-a-kind prototype will not be able to capture the potential benefits that could accrue from some form of "mass production" of modular components in a factory environment – for subsequent shipment and assembly a plant site. It has been argued[7] that it is modularity of component construction that will encourage the acceptance and commercial utilization of the GEN IV nuclear plant designs. In this conceptual model, modularity takes advantage of economies of production, not the economy of scale (i.e., the GEN III very large base loaded plants) to both reduce construction capital at risk, and to reduce overall costs, thereby encouraging commercial acceptance.

In summary, economic considerations from both the reactor vendor and end user standpoints support the construction of the NGNP as a full sized demonstration plant.

6.1.3 Plant Safety Limits

The key parameters that influence the performance of reactor passive safety features (in order of importance) are the

- power level (including the axial profile),
- decay heat,
- thermal conductivity of graphite (especially including the effects of irradiation on the conductivity and the effects of annealing at higher temperatures),
- amount of bypass flow and its effect of cooling the reflector graphite,
- amount of power generated outside the active core, which has an influence that is similar to, but opposite in effect of, the bypass flow,
- inlet temperature, and
- outlet temperature.

The power level and inlet and outlet temperatures are project influenced parameters and are set by the particular design. Similarly, design decisions can be used to control the amount of bypass flow to within certain limits. It is possible that uncertainties in the graphite conductivity and uncertainties in the effect of annealing can be reduced through additional materials R&D.

The effect of bypass on cooling the reflector graphite has a rather large influence on the peak temperature during depressurized conduction cooldown (DCC) events. Parametric calculations, internal to AREVA, with the unrealistic assumption of no bypass flow had peak fuel temperatures during DCC that were approximately 77 °C higher than the corresponding reference cases. Other parametric cases show that the bypass flow through the central reflectors has a larger effect than bypass flow in any other region of the core. Nevertheless, beyond a minimum amount (a few percent of the total flow) required to cool the reflectors, additional bypass flow has little effect on the peak temperatures.

Modular HTR's rely on conduction and thermal radiation in their passive safety features for decay heat removal. Therefore, the selection of geometry, materials, and power level are all direct factors in the ability of the design to avoid exceeding limits after a loss of active cooling. This differs significantly from LWR's for which these plant-level decisions primarily influence the size of supporting safety systems.

It is difficult to assign an "optimum value" for the specific parameters without sufficient consideration of the savings in cost of the NGNP prototype resulting from these design decisions and an equal consideration of the costs of additional R&D that would be required to scale the data acquired from the prototype to the level of a full-size commercial plant. Nevertheless, the following comments can be made.

The thermal performance of the plant during a loss of active cooling is dominated by four items: the geometry of the plant, the thermal energy stored in the core at the beginning of the event, and energy (the decay heat) that is generated inside the core, and the heat transfer properties of the core (graphite). In the case of Pressurized Conduction Cooldown, which is less challenging for the fuel, the movement of heat through natural circulation of the helium coolant is also important. These four items are influenced by the operating parameters listed above as follows:

Power level and decay heat - The decay heat is directly related to the power level and is a strong factor in determining the peak temperatures reached during a DCC transient. Thus, reducing the linear power level (thermal power per unit height) would result in lower temperatures and is less challenging to the passive heat removal features of the design, but could still provide some useful data and experience. One possibility to compensate for the reduced power would be to decrease the reactor's physical size; however, this has serious consequences for other data and operational experience provided by the prototype -- such as core layout, (perhaps) block size, control rod locations, and neutronics.

Outlet temperature - The outlet temperature influences the maximum temperature of the fuel during normal operation, but has a much smaller effect on the safety features for decay heat removal and the peak temperatures during DCC.

Furthermore, the selection of outlet temperature is primarily determined by the target application of the nuclear heat source. All other considerations being equal, the higher the outlet temperature the more power is stored in the core and the more severe the impact of events that either increase the power level, locally or globally, or decrease the coolant flow either locally or globally.

Inlet temperature - Assuming sufficient bypass flow, the inlet temperature determines the temperature of the majority of the reflector graphite in the core and thus most of the thermal energy stored in the core at the beginning of the event. This has a large effect on the ability of the reflector graphite to absorb additional energy during a DCC event and thus a relatively large effect on the peak fuel temperature.

A change in linear power level for the prototype would have the following effects on the key parameters listed above:

The thermal properties of graphite - A reduction in linear power level will result in lower temperatures in the core during a DCC event. Thus, any data obtained for the thermal response of the core during conduction cooldown will be for a temperature regime that is lower than the temperatures that would occur in a full-scale reactor.

Bypass flow - A reduced power level will result in a lower mass flow (assuming that T_{out} - T_{in} is unchanged), which would result in a change in the pressure drop across the core and a change in the amount and distribution of the bypass flow for a given core geometry. The effect of this change in bypass flow on passive decay heat removal would most likely be small or negligible, however.

Power generated outside the active core - To first approximation, this should vary directly with the core power.

Inlet temperature / Outlet temperature - These are additional design parameters of the reactor, which can be altered independently of the core power and which could potentially be adjusted to compensate for the effects of a power reduction.

From the results of the parametric studies above, it is possible to quantify the sensitivity of the DCC results to these key parameters in terms of an equivalent change in reactor power. The impact of the graphite thermal conductivity on decay heat removal is very non-linear and depends greatly, not only on the temperature of the graphite, but its irradiation history as well. The results given here were obtained from calculations that used conductivity values that differed by ±25% from the thermal conductivity data for irradiated graphite, such as would be found in the core blocks at end of life.

These results are expressed in terms of an equivalent increase in reactor power level required to achieve the same peak fuel temperature during a DCC event.

- -5.9 MWt / % increase in residual power

- 1.0 MWt / % change in graphite conductivity - 27.7 MWt if power generated outside the active core is included in the calculation
- -0.23 MWt / °C increase in inlet temperature
- -0.11 MWt / °C increase in outlet temperature

Sensitivity to the initial power level of the peak temperatures during normal operation and during DCC were determined for conditions that were limiting for the fuel and for conditions that were limiting for the reactor vessel. Taking an average of the two results, these calculations indicate that the sensitivity of the maximum temperature during normal operation is 0.229 °C per MWt core power and the sensitivity of the peak temperature during DCC is 1.771 °C per MWt core power.

The mass flow through the core varies directly with the core power, assuming that T_{out} - T_{in} remains the same. The relationship between the mass flow, the amount of bypass flow, and its distribution in the core is non-trivial and requires further study.

The answer to the question, “Are there any threshold powers above which it is impractical to meet the applicable DCC limits?” depends on the safety philosophy that is used and how one combines uncertainties. Best estimate calculations, conducted internally by AREVA, remain below the 1600 °C guideline for a core power of 600 MWt. If uncertainties are "stacked", that is, if the most conservative value is assumed for every parameter influencing the peak fuel temperature, then a core power of 400 MWt (and a reduction in both inlet and outlet temperatures) is required to remain below the 1600 °C guideline. However, this approach is extremely conservative.

A less conservative calculation was conducted internally by AREVA eliminating the uncertainties in some of the less important parameters and choosing "reasonable values" for the rest. The results demonstrated that a power level of 540 MWt is sustainable without exceeding the 1600°C fuel-temperature guideline during DCC.

A more recent AREVA internal attempt to combine the uncertainties that contribute to this safety calculation, which has taken a more realistic approach for combining uncertainties, while maintaining a reasonable level of conservatism, has suggested that 565 MWt is a more accurate limit for the power level that is able to keep the fuel temperature below the safety guideline. Figure 6.1 presents the calculated reactor power level limits as a function of core inlet temperature for a 102 column prismatic core with an outlet temperature of 900°C and a limiting accident fuel temperature of 1600°C.

It should be noted that these results are very preliminary in nature, based upon the current level of understanding of the NGNP reactor core configuration and operational parameters. The uncertainties included in these analyses include both real calculational uncertainties and added margins due to our current lack of detailed information. It was designed to provide for a realistic approach to combination of appropriate uncertainties while maintaining a reasonable level of conservatism. Once more concrete input parameters are available, addressing many of the concepts briefly discussed above such as graphite thermal response and core bypass flow values, a more refined analysis can be conducted. Such a reanalysis may allow some increase in rated reactor power.

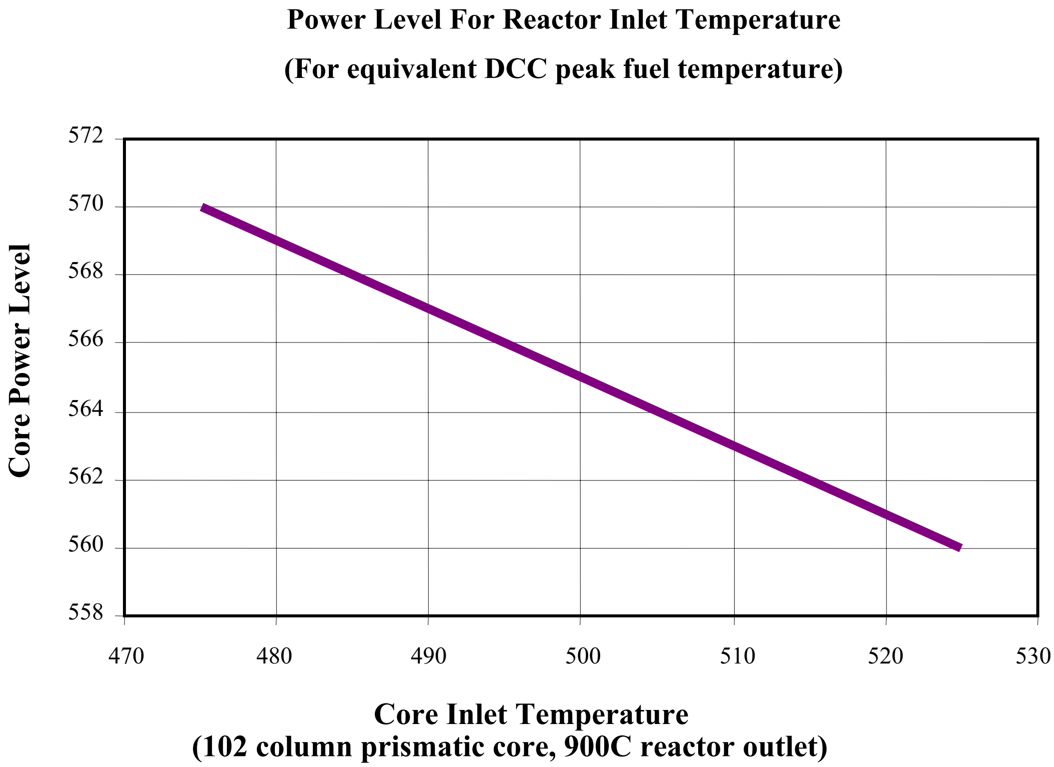


Figure 6.1 Depressurized Conduction Cooldown Analysis Results

Based on the considerations presented here, a maximum reactor thermal power rating of 565 MWth should be considered for NGNP the preconceptual design. This power level will allow some margin for uncertainties as the reactor design process continues.

6.1.4 Licensing Issues

The current commercial; nuclear plant licensing rules in the United States have created and evolved over the past 40 years to license light water reactor (LWR) technology. These regulations are generally ad-hoc, prescriptive, and deterministic. The regulations have generally evolved as operational experience level increased.

Thus all currently operating nuclear power reactors in the United States were licensed using the two-step licensing process of 10 CFR Part 50 where a construction permit (CP) is granted based on preliminary reactor design and site environmental impact statement. The license to operate the plant (OL) is later granted after years of review during the plant construction phase. This process naturally leads to regulatory delays and potential expensive construction rework.

The uncertainty associated with this licensing process was among the key factors that practically stopped the construction of nuclear plants in the United States in the last 25 years. The Nuclear Regulatory Commission (NRC) recently provided an alternative to the Part 50 licensing to remove this uncertainty. The so call 10 CFR 52 one-step licensing process, was created to grant a combined construction and operating license

(COL) to an applicant before the plant construction began. Of course, the COL application must include a plant design and site environmental impact statement or reference a previously reviewed and approved early site permit (ESP) and certified design (DC). The new Part 52 COL process has yet to be fully exercised. It is now undergoing trail usage by several applicants. However, the ESP and the DC portion of this licensing process have been tested by several utilities and reactor designers. Nevertheless, the basic nuclear plant regulations still reside in old and evolving 10CFR Part 50 and they are for the most part specific to light water reactor technologies.

The gas cooled reactor technology being considered for the NGNP prototype reactor is conceptually different from the traditional LWR technologies in most aspects of reactor design, safety, and operations. Both the pebble bed and the prismatic technology being contemplated for the NGNP prototype are specifically designed to include passive safety and inherent characteristics that are required by the Generation IV reactor requirements. Therefore, the deterministic licensing regulations developed for the LWRs do not directly apply to the modern gas cooled reactor technologies being considered for the NGNP prototype.

Through various gas cooled reactor licensing attempts in the recent history starting from DOE's efforts in 1990s to licenses the MVHTR, GA's application to license the GT-MHR, Exelon's interactions with the NRC on the pebble bed design, and the current Westinghouse efforts seeking certification for their PBMR, the US-NRC has recognized that new regulatory framework and subsequent licensing regulations are necessary to license non-LWR reactor technologies in the United States. The NGNP will most likely be a candidate that will utilize this new regulatory framework and it could be one of the first test cases that will exercise the new regulations.

In order to efficiently commercialize the VHTR, it would be clearly advantageous that a full scale NGNP prototype reactor is designed and reviewed by the NRC. This would assure that all aspects of the framework including the technology neutral and the subsequent technology specific regulatory guidance portion of the regulation are exercised such that most if not all safety issues have been reviewed and resolved through this prototype licensing effort.

Under the new regulatory framework a design phase PRA will be prepared and utilized to determine the licensing bases events (LBEs) and a subsequent design bases and beyond design bases events (DBEs) and (BDBEs). These event families and the subsequent design bases accidents (DBAs) must be deterministically analyzed with approved safety analyses computer codes. These analyses performed for the prototype are most useful for future commercial application if they are performed for a full scale plant design.

The risk informed regulator nature of the new licensing framework requires that the design phase PRA should evolve into the plant as built PRA and will subsequently become the living plant PRA. As a living PRA, it will serve as a tool that the plant designer, utility/operator, and the regulator can utilize to assess plant performances and sensitivities. The value of such a tool is most beneficial if it is developed full scale where many semi-scale plant performance characteristics of a commercial reactor designed with passive safety features may not be readily relevant.

Therefore, from the licensing point of view it is recommended that the power level of the NGNP prototype plant should at the commercial scale NOAK plant.

6.1.5 Demonstration of Passive Safety Features

Modular gas cooled reactors are designed for safety. This results in specific design decisions that provide such reactor characteristics. The NGNP, as a prototype of such reactor design for subsequent commercialization, must be capable of providing technical evidence of the performance of such features as they were postulated by the designers. The following are major design decisions for any modular gas cooled reactor type which must be demonstrated:

1. Annular core performance characteristics
    a. Neutronic
    b. Thermal
2. TRISO particle fuel performance characteristics
    a. Neutronic
    b. Thermal
3. Graphite characteristics
    a. Moderator and reflector neutronics
    b. Heat transfer properties
    c. Large stored heat capacity
4. Large negative reactivity coefficient
5. Passive residual (decay and stored) heat removal characteristics

The path to commercialization of modular gas cooled reactor is through the NGNP prototype demonstrating passive safety features to the licensing authorities in addition to the potential customers. The performance-based component of the new regulatory framework demands technical evidence of the performance and safety claims assumed or postulated by the design. The data necessary to provide this proof can only be provided by individual full scale test facilities or an integrated test facility. For passive features the dynamics of the required proof test demand full scale models. Extrapolation from scaled test facility is possible but the true dynamics of the system interactions can only be demonstrated with a full scale facility.

In consideration of these observations, the NRC, in 10 CFR 52.47(b)(2)(i)(B), promulgated the following requirement.

> *(2)(i) Certification of a standard design which differs significantly from the light water reactor designs described in paragraph (b)(1) of this section or utilizes simplified, inherent, passive, or other innovative means to accomplish its safety functions will be granted only if:*

(B) There has been acceptable testing of an appropriately sited, full-size, prototype of the design over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions. If the criterion in paragraph (b)(2)(i)(A)(4) of this section is not met, the testing of the prototype must demonstrate that the non-certified portion of the plant cannot significantly affect the safe operation of the plant.

In other words, successful licensing of a commercial VHTR reactor depends on the successful safety testing of a full-scale prototype reactor. This is a critical role for the NGNP to fulfill.

Beyond the technical, regulatory, and licensing considerations which drive a full scale safety demonstration test, acceptance of this technology by reactor vendors, potential end users, and the general public would be greatly enhanced by a full-scale, integrated demonstration of the plant passive safety features.

6.2 Remaining Study Criteria

6.2.1 Core Neutronics

In order to maintain passive cooling capability, which typically limits the power density of the core, increases in core power level typically result in changes in core geometry, either increasing core diameter or height. These geometric changes will change the neutronic behavior of the core, particularly with respect to xenon stability at larger core sizes.

The diameter of current reactor designs in the 600 MWth range are well within the acceptable range of core widths from a neutronic stability standpoint. However, increases in core diameter are precluded by other considerations, particularly related to reactor vessel feasibility. The 600 MWth plant designs are essentially at the vessel diameter limit for all practical purposes.

Previous work by GA and INL[8] indicates that a core height of 10 blocks or less, consistent with current 600 MWth plant designs, are neutronically stable and require no active Xenon control measures. There are preliminary indications that it may be possible to utilize 11 and possible 12 block core heights without active xenon control, but no decisive studies have been conducted to date. Beyond these heights, at some point, there is a need for active control measures, which result in significant complication in rod movement strategy, particularly considering the large temperature variation along the core and the impact that has on neutronics. One of the reasons that no significant work has been done on these higher cores is that other considerations, including higher core pressure drops requiring greater circulator power requirements and reactor vessel embedment issues have supported a core height of no more than 10 blocks.

Based on these observations, it is concluded that neutronics concerns will not be a limiting factor in the determination of the recommended power level for the commercial and NGNP reactors.

6.2.2 Fabrication Issues

One of the advantages of the modular VHTR design is the ability to fabricate major components within a factory setting and ship the completed components to the plant site. The one component which presents some logistical issues is the reactor vessel. It must be a single component within each module, where other large components can be split into multiple trains should field fabrication be not feasible. The reactor vessel fabrication question can be divided into two groups: Fabrication of the reactor vessel for sites with barge access, and for those sites restricted to land access.

Table 6.2 – Representative Reactor Vessel Parameters

	MVHTR 350 MWth[9]	**MVHTR 450 MWth**[10]	**Antares 600 MWth**
Outside diameter at the flange level	7.36 m (24.1 ft)	≈ 9 m (29.5 ft)	8.24 m (27.0 ft)
Outside diameter in the cylindrical part	6.8 m (22.4 ft) upper 7.0 m (22.9 ft) lower	7.55 m (24.8 ft) upper 7.64 m (25.1 ft) lower	7.54 m (24.7 ft) upper 7.74 m (25.4 ft) lower
Reactor vessel height	22 m (72.0 ft)	23.5m (77 ft)	25 m (82 ft)
Core inlet temperature	259°C (497°F)	288°C (550°F)	400°C (752°F)
Core outlet temperature	687°C (1268°F)	704°C (1300°F)	850°C (1562°F)
Primary pressure	6.4 MPa	7.07 MPa	6 MPa
Core concept	annular	Annular	annular
Active core equivalent outer diameter	≈ 3.5 m (11.5 ft)	4.17 m (13.7 ft)	4.84 m (15.9 ft)
Minimum reflector thickness	≈ 1.1 m (3.6 ft)	1.0 m (3.3 ft)	≈ 0.82 m (2.7 ft)
Number of fuel element columns	66	84	102
Average power density	5.9 MW / m^3	6.0 MW / m^3	6.5 MW / m^3
Material	LWR steel	LWR steel SA533, Grade B Class 1 and SA508 Class 3	Mod 9 Cr 1 Mo

Table 6.2 presents representative parameters for various reactor sizes. It should be noted that the active core outer diameter is a function of the power and also of the fuel element size. The increases from 350 to 450 then from 450 to 600 are each accompanied with the translation of the annular core by one fuel element outward.

The ANTARES design (based on GT-MHR) corresponds to a larger active core outer diameter. At the same time the outer diameter of the reactor vessel remains the same as that of MVHTR 450 MWth in the upper cylindrical part which means that internals design has been optimized. The ANTARES design is also based on modified 9 Cr 1 Mo as reference material which is more creep and irradiation resistant compared to conventional LWR vessel steel.

Sites With Barge Access

Many of the sites that are likely to support initial deployments of NGNP technology, particularly those related to existing hydrogen usage or petrochemical refining, are located in areas that support barge shipment of the reactor vessel to the plant site, such as the Gulf Coast of Texas and Louisiana. There is at least one domestic vendor capable of fabricating and barge shipping vessels of a size that approximates the size of the ANTARES reactor vessel to these locations[11]. This situation will likely prove the most cost effective method for delivering a completed reactor vessel.

Sites Without Barge Access

Both the proposed NGNP site and many sites that represent the largest potential deployment of VHTR technology, including tar sands and oil shale extraction, are located in areas that preclude shipment of a large reactor vessel by barge. Representative railroad size and weight limits[12] in the United States of 4.6 m wide and 5.6 m high with a weight limit of 800 tons clearly preclude shipment of any of the reactor vessels listed on Table 6.2. As such, deployment of an NGNP reactor of any reasonable size will require on-site fabrication of the reactor vessel. This situation may prove advantageous, in that the required techniques can be developed and demonstrated before commercial use.

Experience indicates that final assembly of the reactor vessel can be performed on site. There are questions as to whether the best option would be to perform the welding in the reactor cavity or in a dedicated on-site workshop. The major determinant will be the cost. Workshop fabrication is probably reasonable when the intent is to built 4 modules or more. It is probably an expensive option for one unique module. The final assembly in the reactor cavity is probably also technically possible. However, difficulty will be linked to the qualification of the processes (welding, etc) on site. Local Post Weld Heat Treatment will also have to be performed. In addition, radiographic examinations and final machining of the vessel are significant field fabrication issues. The radiographic examinations are an important schedule issue in that a wide range of surrounding activities can be affected due to personnel protection. Field final machining of the large diameters can also be challenging.

Based on this information, the fabrication method for the reactor vessel will not be a determining factor for the selection of the commercial or NGNP reactor power level, in that sites will be either capable of receiving a full sized reactor vessel or will require on-site fabrication of reactor vessels for any reasonable reactor power.

6.2.3 Component Feasibility

With respect to reactor power level, the component of interest is the reactor vessel, and the issue of interest is the material from which it is constructed. There are two material choices which are typically considered, modified 9 Cr 1Mo or LWR vessel materials. Each has potential benefits and drawbacks.

Modified 9 Cr 1 Mo Material

The availability of modified 9 Cr 1 Mo heavy section forgings in the dimensions required for HTRs is in question, more so even than LWR vessel materials which also have some schedule issues in this area. So far, the capabilities of Japan Steel Works for instance are not compatible with the required ingot size. A back-up solution would be to use plates instead of forgings. This is the current practice for BWRs (even modern ones) which reactor sizes and pressure conditions are close to those of HTRs. It is however to be noted that BWR vessels operate under fluences even lower than those of HTRs (which are already reduced compared to those met in LWRs). In any case, due to NGNP schedule limitation of 2018, it is likely that recommendations will be made to base the design on plates and to keep the forging option as a target for the N^{th} of a kind commercial reactor.

This material is also more costly than the alternate LWR material.

LWR Vessel Materials

The other alternative would be to select LWR material (SA 508 grade3 class1 for forgings and SA 533 grade B class1 for plates) instead of mod 9 Cr 1 Mo. This solution would have two drawbacks:

1. Need to decrease the core inlet temperature so that the temperature of the vessel would fall under the current limits of ASME Code Case N-499 defined as follows:
   - Normal service temperature < 371°C
   - Limitation on transients:
     - 3000 hours maximum duration between 371 and 427°C
     - 1000 hours and no more than 3 events between 427 and 538°C

   The reduction of the core inlet temperature would increase the temperature rise of the coolant through the core and this combined with the increase of the core outlet temperature envisioned for the NGNP is likely to be unacceptable for the fuel under normal operation

2. Need to operate at a power that supports acceptable conditions for the vessel under conduction cooldown situations. Table 6.3 indicates that all the cases from Ref. 14 that are conservative for the vessel give temperatures in DCC greater than the 538°C limit of Code Case N499. The maximum vessel temperature during DCC is slightly dependent on the core inlet temperature and core outlet temperature. The comparison of cases (1) and (2) on the one side and (1) and (3) on the other side show a decrease of the maximum vessel temperature by 2.4% when the core inlet temperature is decreased by 100°C and an increase by 0.9% when the core outlet temperature is increased by 100°C. Cases (1) and (4) can therefore be used to show the influence of power level for the NGNP base line. Assuming that the effect would be linear, the temperature achieved for a power of 550 MW would be 535°C. This would be theoretically acceptable in meeting this one limit, but does not consider other, time at temperature related limits that are also part of this code section, for example, it is also necessary to check that the cumulative duration

spent above 427°C is below the maximum allowed duration of 1000 hours. The consequence is therefore that the power level below which LWR material could be envisioned, subject that the temperature of the vessel can be reduced enough during normal operation (by implementation of a thermal insulation or by the modification of the flow path) would more likely be in the 450 MW range to provide some margin to these limits and assure some degree of operational success.

Table 6.3 – Key DCC Temperatures

Event	DCC	PCC	DCC	DCC	DCC	DCC
Assumption	Best estimate	Best estimate	Conserv. Vessel (1)	Conserv. Vessel (2)	Conserv. Vessel (3)	Conserv. Vessel (4)
Power (MW)	600	600	660	660	660	612
Core inlet	400°C	400°C	480°C	380°C	480°C	480°C
Core outlet	850°C	850°C	880◦C	880◦C	980◦C	880◦C
ΔT core	450°C	450°C	400°C	500°C	500°C	400°C
Mass flow (kg/s)	256.9	256.9	317	317	317	317
Pressure (MPa)	6	6	5	5	5	5
Bypass flow (%)	10	10	5	5	5	5
Fuel	1475°C	1374°C	1574	1527	1599	1494
Core barrel	710°C	627°C				
Main vessel	477°C	402°C	574	560	579	557
Core support structure	612°C	661°C				

Based on this data, operation of the NGNP at 565 MWth will require the adoption of a vessel made of modified 9 Cr 1 Mo material. This does not present a limitation on the reactor power level selected.

6.2.4 Plant Flexibility and Operability

The flexibility and operability characteristics of a particular reactor/production plant system are a strong function of the specific processes to which the reactor supplies power, the power split amongst multiple uses, and the fraction of reactor power supplying each individual process train. Overall system configurations will determine allowable power change rates and will dictate optimum operational strategies.

One area that will need to be addressed for each operational scenario is the safety impact of the load characteristics of the secondary systems. Operation of the NGNP hydrogen process plant at only 10% of the total reactor power may not fully reflect some of the system feedback effects that may impact both operational and accident performance in plants that utilize a significant fraction of reactor power for hydrogen production or other process heat. Such feedback effects will need to be investigated for configurations representative of various commercial applications. The potential transient power behavior of the system must be reviewed to ensure that the most limiting cases are addressed in the plant safety analyses, including any potential power feedback effects that may exist. It is not the purpose of this study to attempt to answer these questions, but only to document them as important to ask at some point.

Though various power and temperature control strategies may need to be implemented within the process loops to reduce reactor fluctuations, these characteristics do not appear to be strongly impacted by the overall power of the reactor module, and, therefore, have little impact on the decision regarding the power level of the NGNP and commercial reactor units.

Preliminary research activities on VHTR-based hydrogen generation systems[13] indicates that, if the power requirements of the hydrogen plant are a significant fraction of the reactor thermal output, a "thermal absorber" device is needed in the secondary power system to minimize the impact of process temperature fluctuations on the reactor. In the referenced research, a small steam generator is used to perform this function. Since the demonstration hydrogen production plant associated with the NGNP reactor will only utilize approximately ten percent of the total reactor power output, it is not anticipated that such a system would be required.

6.2.6 Research and Development

The investigations conducted in the course of completing this special study have identified three areas that require additional development activities to support the conclusions reached. These areas are:

1. Development of large-scale forging capabilities for modified 9 Cr 1 Mo vessel material.

2. Improvement of the ASME Boiler and Pressure Vessel Code to incorporate all of the required operational conditions anticipated for the modified 9 Cr 1 Mo vessel material.

3. Development and qualification of fuel designs to allow operation at the desired power level for the desired durations. (Current fuel designs should support operation at the desired power level, though for shorter cycles or larger reload batch sizes.)

There does not appear to be sufficient risk associated with these activities, in terms of schedule or cost impacts, to change the recommendations made by this special study.

6.3 Hydrogen Plant Process Heat Requirements

6.3.1 Hydrogen Production Systems

The purpose of this section is to provide a high-level description of the hydrogen process options considered in the determination of process heat power requirements since the AREVA scope of work does not include design or detailed analysis of the hydrogen processes. The candidate hydrogen processes to be considered include the sulfur iodine (S-I) thermochemical process and the high temperature electrolysis process.

Sulfur Iodine Thermochemical Process

The Sulfur-Iodine (SI) process is a classic thermochemical cycle. Thermochemical cycles combine a net endothermic series of linked chemical reactions to achieve a desired overall reaction while regenerating all intermediate reactants. Thermochemical hydrogen cycles split water into hydrogen and oxygen with heat and water as the only system inputs.

The SI cycle consists of three chemical reactions, coupled in two process loops. The process involves thermal decomposition of sulfuric acid and hydrogen iodide, followed by regeneration of these reagents using the exothermic Bunsen reaction. Process heat is supplied at temperatures greater than 800°C to concentrate and decompose sulfuric acid. The exothermic Bunsen reaction is performed at temperatures below 120°C and releases waste heat to the environment. Hydrogen is generated during the decomposition of hydrogen iodide, using process heat at temperatures greater than 300°C. The General Atomics SI process flowsheet with reactive distillation of hydrogen iodide as the third step is assumed for illustrative purposes.

Section 1 carries out the exothermic Bunsen reaction,

$$I_2 + SO_2 + 2\,H_2O \rightarrow H_2SO_4 + 2\,HI$$

This primarily takes place around 120°C in a heat exchange reactor and, to a lesser extent, in two oxygen scrubbers and a sulfuric acid boost reactor. The output from the heat exchange reactor consists of three phases, which are separated and processed separately. The gas phase contains primarily O_2, which is scrubbed to remove residual SO_2 and withdrawn as a co-product. The sulfuric and hydroiodic acids split nearly completely into two, distinct liquid phases. The lighter of the two liquid phases contains sulfuric acid, which is concentrated in a boost reactor to about 20 mole % and passed on to Section 2 for decomposition. An aqueous solution of hydroiodic acid and iodine comprises the heavier phase, which is passed on to Section 3.

Section 2 carries out the high temperature, endothermic decomposition of sulfuric acid into water, SO_2, and O_2,

$$H_2SO_4 \rightarrow H_2O + SO_2 + \tfrac{1}{2}\,O_2$$

The sulfuric acid stream from Section 1 is heated to about 550°C, pressurized to about 70 bar, and concentrated to about 50 mole % H_2SO_4 in a series of flash steps, after which it is vaporized (550-700°C) and decomposed (700-850°C) in a catalytic reactor using high temperature heat provided by the intermediate heat transfer loop. The 850°C decomposition products are cooled by interchange with the sulfuric acid being concentrated as well as by interchange with process streams in Section 3. Cooled and partially condensed product from Section 2 (40-120°C) is returned to Section 1 to close the sulfuric acid loop of the SI cycle.

Section 3 carries out the intermediate temperature decomposition of hydroiodic acid into hydrogen and iodine,

$$2\ HI \rightarrow H_2 + I_2$$

The aqueous solution of hydroiodic acid and iodine from Section 1 is heated by interchange with other process streams and fed to a reactive distillation column in which hydrogen iodide is taken overhead along with water and simultaneously decomposed to hydrogen and iodine. This column is operated at 40 bar and 265-290°C. The vapor overhead product is primarily water, hydrogen, and some unreacted hydrogen iodide, while the bottoms are mostly iodine and water. Since much of the water that comes with the aqueous solution of hydroiodic acid and iodine from Section 1 is vaporized, the heat of vaporization must be recovered for the process to be efficient. Heat pumps, with steam as the working fluid, are used to recover heat from water condensation. A novel feature of this process is that the heat of solution obtained by mixing the overhead and bottoms products is also recovered using a heat pump. Hydrogen is separated from the reactive distillation effluents and removed as product, while all of the remaining streams are cooled by interchange and returned to Section 1 to close the hydroiodic acid loop.A simplified SI process flowsheet is shown in Figure 6.2 below.

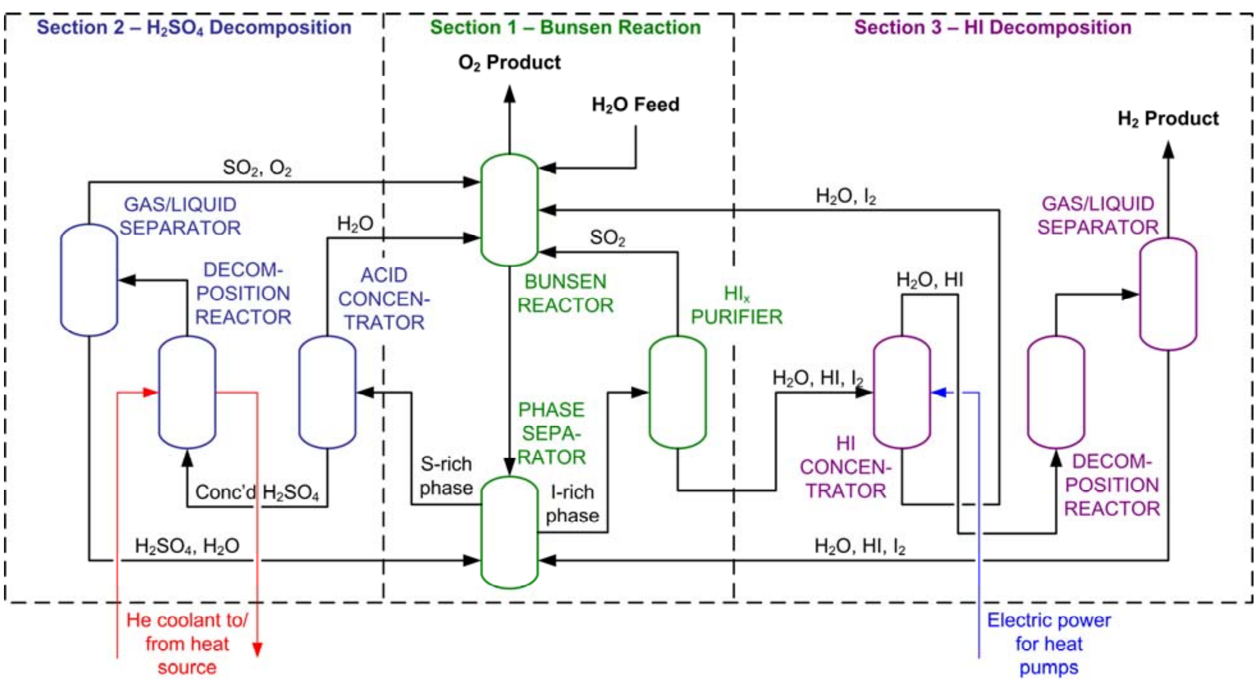


Figure 6.2 - Simplified Sulfur-Iodine Process Flowsheet

High Temperature Electrolysis

High Temperature Electrolysis (HTE) refers to the process of electrolyzing water (actually steam) at elevated temperature in a ceramic-type electrolyzer. The process is essentially the reverse of solid oxide fuel cell operation, in which hydrogen and oxygen (from air) are electrochemically reacted to produce water, heat and electric power. In the case of HTE, steam is reacted over a catalyst in a solid oxide electrolyzer at 800-1000 °C to produce hydrogen at the cathode of the cell and oxygen at the anode of the cell. The advantage of a high temperature HTE versus conventional low temperature water electrolysis is that a portion of the energy can be supplied as heat rather than electricity. This results in a substantial improvement in overall plant efficiency. Figure 6.3 shows how the thermal and electrical requirements vary for water electrolysis as a function of temperature.

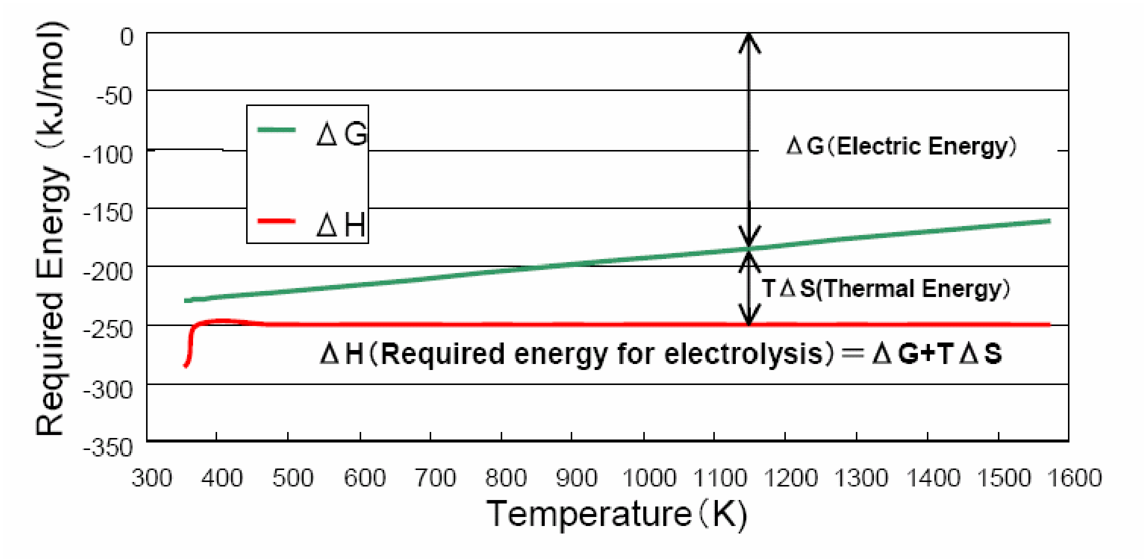


Figure 6.3 - Energy Requirements for High Temperature Electrolysis

The solid oxide electrolyzer operates by the conduction of oxygen ions through a ceramic electrolyte, typically yittria-doped zirconia. In order to obtain sufficient electrolyte conductivity, the cells must be operated at temperatures 800 °C. At the HTE operating temperature of 800-900 °C approximately 80% of the energy needs to be supplied as electricity and 20% as thermal energy in the form of superheated steam.

In actual operation, the HTE also requires a steam sweep gas be supplied to the anode portion of the electrolyzer in addition to the steam/hydrogen mixture supplied to the cathode. The cathode feed is expected to consist of 90% steam and 10% hydrogen. Heat from the nuclear reactor (via a helium heat transport loop) is used to generate steam, superheat the cathode steam/hydrogen mixture, and preheat the anode steam sweep gas. A hydrogen production efficiency of 55.5% (HHV basis) has been estimated for an HTE plant using heat from a modular helium reactor[14].

Since the HTE process is based on an electrochemical reaction, it is modular in nature due to limitations on scaling-up the electrochemical cell. A recent paper by General

Atomics, Idaho National Laboratory and others[14] proposed that a four reactor plant based on 600 MWth MHRs with an HTE hydrogen process plant would require 300 trailer-size HTE units containing 8 modules per trailer. Each HTE trailer unit would require 4.0 MWe of power to drive the electrolyzers. An additional 1.0 MWth of thermal energy would be required for the heat duty requirements of each trailer unit. The module HTE concept is shown in Figure 6.4.

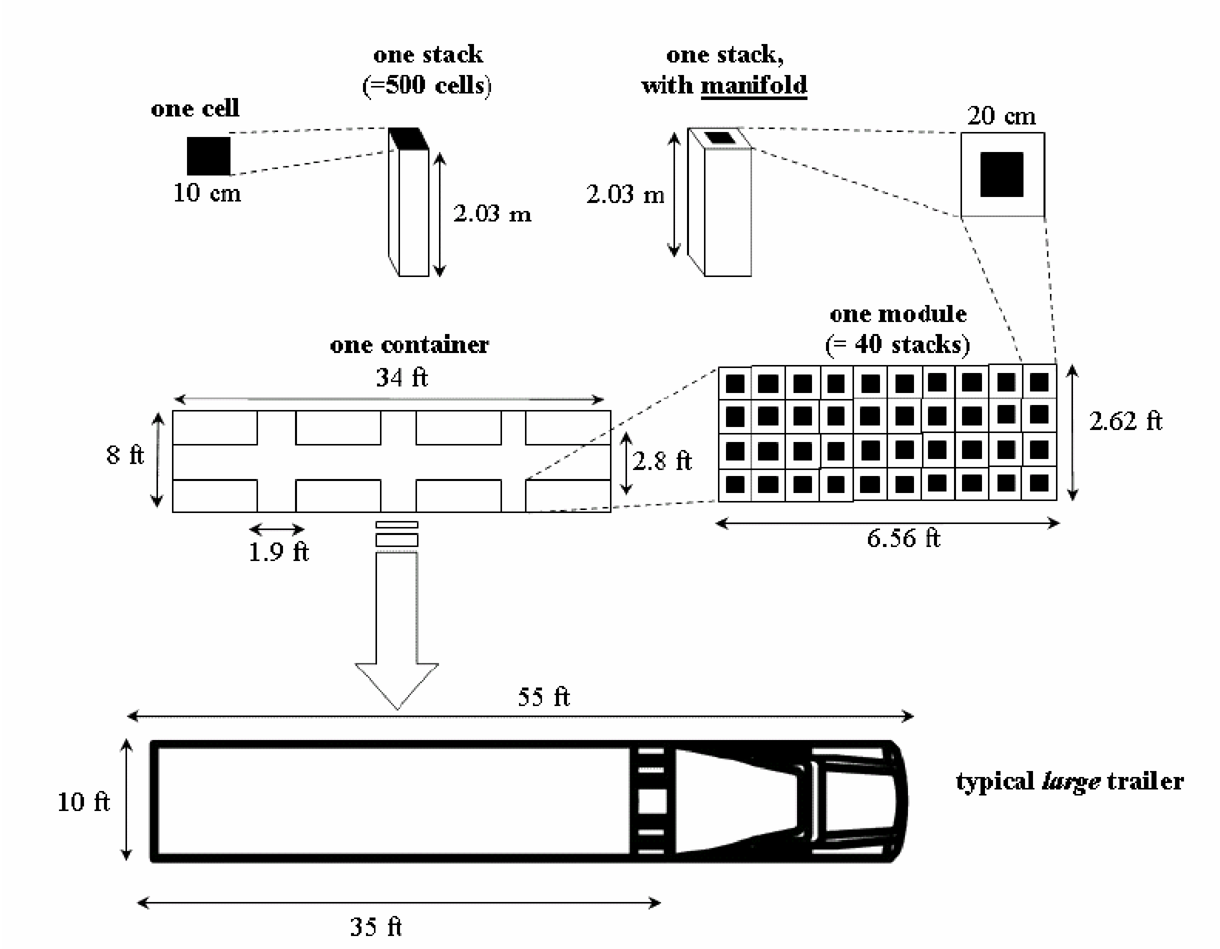


Figure 6.4 - Potential HTE Module Concept[14]

6.3.2 Hydrogen Production System Power Requirements

The hydrogen process requirements impact the power level trade study by establishing the thermal and electrical needs for the hydrogen engineering demonstration. In order to determine these requirements, an assessment was made of the expected power requirements for a commercial-scale hydrogen process. By considering the likely modular design of such a plant, and well-known chemical engineering scaling procedures, one can then determine the necessary power requirements for the NGNP demonstration scale hydrogen process.

Expected Power Requirements for Commercial-scale hydrogen process

A commercial-scale nuclear hydrogen plant will likely utilize the entire reactor thermal output for hydrogen production. A recent study by Savannah River National Laboratory[5] determines that a 600 MWth helium-cooled nuclear reactor combined with a sulfur iodine hydrogen production process could produce approximately 280 metric tonnes per day of hydrogen, which is sufficient for a large ammonia plant or oil refinery. This design used a total of 192 MWe from the grid. If the plant were self-sufficient, generating both thermal and process electric power, the hydrogen output per reactor would be less.

The 280 TPD SI plant described in the referenced report utilizes three process trains for Section 1 (Bunsen reaction), a single large train for Section 2 (sulfuric acid decomposition) and ten process trains for Section 3 (HI decomposition). The Section 3 units were based on the largest shippable components.

Since the HTE process is based on an electrochemical reaction, it is modular in nature due to limitations on scaling-up the electrochemical cell. Commercial applications would thus be scalable to the required size by adding modules.

Expected Power Requirements for Demonstration-scale Hydrogen Process

Well-known chemical process design requirements permit scale-up from smaller equipment sizes to full-size commercial units. A scale-up factor of 10:1 is usually achieveable, and larger scale-up factors are possible in many cases. For new equipment with several materials and design features that have not been demonstrated previously at large scale, it is often advisable to demonstrate a full-scale unit. For the S-I Process the most developmental and challenging portion of the process is the HI Decomposition (Section 3). It is therefore recommended that a full-size Section 3 process train (one of ten in a commercial plant) be demonstrated. Sizing the balance of the SI Process to this capacity would result in a hydrogen process energy requirement of 60 MWth and 20 MWe. Since the Section 3 process train for the commercial plant was based on the largest shippable unit size, it would also be permissible to test this unit at a slightly reduced size and still be large enough such that the key engineering questions can be answered. In this case, the SI hydrogen plant would require thermal input in the 30-60 MWth range and electric power of 10-20 MWe.

It is anticipated that the HTE process demonstration would consist of one train of modules, requiring 4 MWe and 1 MWth of process heat.

NGNP Hydrogen Process Test Configuration

It is assumed that hydrogen process testing would be conducted sequentially; nevertheless, given the fairly low power requirements of the HTE process, both HTE and SI processes could be run in parallel.

The power level chosen for the HTE process was chosen to permit deployment of a commercial scale module, including associated systems that could be tested using nuclear heat. Should it be desired to test the HTE process on a scale similar to that of the SI demonstration process, additional HTE modules could be added; however, doing so may impact the ability to run both SI and HTE hydrogen production systems in parallel due equipment limitations (i.e., IHX sized for 60 MWth).

While the development of the final operating strategy of the hydrogen process loop is beyond the scope of this study, it is important to note the level of testing flexibility that could be accommodated in the final design configuration.

6.3.3 Current State-Of-The-Art and Prognosis for Development

The SI thermochemical process has been demonstrated in an integrated system at the small scale of 100 liters by the Japanese Nuclear Energy Agency. A somewhat larger 200 lph SI plant is under construction in the United States as part of the DOE-NE Nuclear Hydrogen Initiative (NHI). It is being constructed by General Atomics in La Jolla, CA in conjunction with Sandia National Laboratory and the French national laboratory CEA. The 200 lph hydrogen output will have an energy content of approximately 600 watt(th). Assuming a process efficiency of 45% (not possible for such a small plant but representative of a commercial-scale unit), the equivalent hydrogen process thermal requirement is 1.3 kWth. This is obviously a long way from the 30-60 MWth required for the NGNP engineering demonstration.

The DOE's NHI program plan calls for the next stage of development of the hydrogen process to be a MW-scale pilot plant beginning in FY11. If a pilot plant of this scale can be built and operated in the 2011-2015 timeframe, it should be feasible to scale-up to the NGNP size of 30-60 MWth by 2018. The biggest challenge is perhaps the design and construction of the MW-scale pilot plant beginning in FY11.

The HTE process has been demonstrated at the component level for 2000 hours with a hydrogen output of 900 liters per hour[15]. This is equivalent to an electrolyzer electrical input of approximately 3 kW(e). The INL program plans for the HTE include completing an integrated lab-scale model at a nominal power level of 15 kW(e) by 2008; a pilot scale module of 50 kW(e) by 2010; a multi-module experiment at 200 kW(e) by 2012; and an engineering scale demonstration of a 5 MW section by 2015. Based on achieving this plan, the HTE development should be sufficiently advanced to meet the NGNP requirements in 2018. One of the major challenges will be the design and operation of a pressurized electrolzyer, since all the current and planned near-term development work has been with atmospheric pressure units.

7.0 Integrated Study Results and Recommendations

Table 7.1 presents a summary of the evaluations documented in the preceding sections. For each study criteria, an indication of the recommended commercial and/or NGNP reactor powers is stated based on the evaluation results. In addition, a qualitative assessment is provided of the potential impacts of a reduction in the NGNP rated power as a fraction of the projected commercial plant. These assessments are based largely on the expertise of the personnel involved in the evaluations. Comments are also provided which summarize the key findings of the individual evaluations.

Results of the evaluations of the key discriminating criteria indicate the following answers to the three study questions:

1. What should be the rated power level of the NOAK commercial VHTR module?

   The commercial VHTR module should be designed to operate at 565MWth.

This recommendation is based on commercial applications that are expected to support large module sizes and an evaluation of plant safety limits which indicate that the maximum power for NGNP initial conditions that provides acceptable results for the DCC accident is 565MWth.

2. Given the desired power level of the commercial VHTR module, what should be the rated power level of the NGNP prototype plant?

   The NGNP prototype plant should be designed and operated at 100% of the planned commercial power level, that is, 565MWth.

   This recommendation is made to support demonstration of plant passive safety features, portability of licensing experience, and sharing of first-of-a-kind engineering costs.

3. In order to demonstrate commercial scalability of an associated hydrogen production plant, what is the power requirement for a demonstration plant to be associated with the NGNP reactor?

   The demonstration Sulfur-Iodine plant will require 60MWth of process heat and 20MWe from the power conversion system.

   The demonstration High Temperature Electrolysis plant will require 1.2 MWth of process heat and 5MWe from the power conversion system.

   These recommendations are based on an examination of the current state of the art for these tow systems and the expected development progress between now and NGNP plant startup in 2018.

Evaluations of the remaining study criteria identified no concerns which would preclude or challenge the use of these recommendations.

Table 7.1 – Summary of NGNP Power Level Special Study

	Study Criteria	Recommended Power Level	Impact of NGNP as a Fraction of Full Size					Comments
			100%	75%	50%	25%	20MW	
Key Discriminating Criteria	Market View	Max. Practical	Base	Moderate	Major	Major	Major	Markets can support module powers as high as achievable within Gen IV constraints.
	Economic Considerations	100%	Base	Major	Moderate	Major	Major	100% power NGNP provides best commercial demonstration.
	Plant Safety Limits	Max. of 565MW	Base	Minor	Minor	Minor	Minor	The coupled limits imposed by Gen IV concerns and nuclear design constraints make this the practical upper power limit.
	Licensing Issues	100%	Base	Moderate	Major	Major	Major	Applicability of licensing experience and techniques will decrease significantly as the power fraction decreases.
	Demonstration of Passive Safety Features	100%	Base	Major	Major	Major	Major	Effective demonstration of the systematic response of the passive safety features requires a full-size test.
Remaining Criteria	Core Neutronics	No Limitations	Base	Minor	Minor	Minor	Minor	Neutronics impacts occur at plant sizes beyond those which would impact the practical upper power limit.
	Fabrication Issues	>50%	Base	Minor	Minor	Moderate	Moderate	Reasonable powers will require on-site reactor vessel fabrication for INL site. Though many commercial sites allow shipment, on-site fabrication experience may be beneficial.
	Component Feasibility	No Limitations	Base	Minor	Minor	Minor	Moderate	Key variables, including temperature and neutron exposure are not directly controlled by power level.
	Plant Flexibility and Operability	No Limitations	Base	Moderate	Major	Major	Major	Operational lessons learned from the NGNP will be applicable only at power levels near that of the commercial plant.
	Research and Development	100%	Base	Minor	Minor	Minor	Moderate	As stated above, key component limits, thus R&D opportunities, are not directly power related. As an overall R&D project, an NGNP plant of 100% power would provide most R&D return.
	Hydrogen Plant Process Heat Requirements	No Limitations	Base	Minor	Minor	Moderate	Major	This topic not directly related to NGNP power at expected power requirements of 60 MW or so, unless it becomes a significant fraction of total plant power.

8.0 References

1. “Next Generation Nuclear Plant – High-Level Functions and Requirements”, INEEL/EXT-0301163, Idaho National Engineering and Environmental Laboratory Bechtel BWXT Idaho, LLC, September 2003.
2. “Design Features and Technology Uncertainties for the Next Generation Nuclear Plant”, INEEL/EXT-04-01816, Independent Technology Review Group, June 30, 2004.
3. “Statement of Work – Preconceptual Engineering Services for the Next Generation Nuclear Plant with Hydrogen Production”, SOW 3963.
4. Adapted (with modification) from: “Configuration and Technology Implications of Potential Nuclear Hydrogen System Application,” Yildiz, et.al,, ANL-05/30, Nuclear Engineering Division, Argonne National Laboratory, July 31 2005.
5. W. A. Summers, “Centralized Hydrogen Production from Nuclear Power: Infrastructure Analysts and Test Case Design Study, Final Project Report, Phase-B Test Case Preconceptual Design”, NERI Project 02-0160, WSRC-MS-2005-00693, Rev.0 April 28, 2006.
6. “Hydrogen Markets: Implications for Hydrogen Production Technologies,” Charles W. Forsberg, Oak Ridge National Laboratory File Name: AICHE05.HydrogenMarkets.Paper February 8, 2005.
7. “A Future for Nuclear Energy – Pebble Bed Reactors,” Andrew C. Kadak, MIT, April 25, 2004.
8. “NGNP Point Design – Results of the Initial Neutronics and Thermal-Hydraulic Assessments During FY-03”, INEEL/EXT-03000870 Rev. 01, Idaho National Engineering and Environmental Laboratory Bechtel BWXT Idaho, LLC, September 2003.
9. “Preliminary Safety Information Document For The Standard MVHTR” VHTR-86-024, General Atomics.
10. “Concept Description Summary Report Modular VHTR Plant 450 MWt Steam Cycle Modular High Temperature Gas-Cooled Reactor Plant” DOE-VHTR-90408 Rev 0, General Atomics.
11. BWXT presentation to Exelon, January 8, 2007.
12. Pacific Railroad data.
13. Y. Inagaki et al., “Research and Development on System Integration Technology for Connection of Hydrogen Production System to an VHTR”, Nuclear Technology Vol. 157, February 2007.
14. M. Richards et al., *Int. J. Nuclear Hydrogen Production and Applications,* Vol. 1, No. 1, 2006.
15. S. Herring et al., 2006 DOE Hydrogen Program Peer Review, Washington, DC, May 17, 2006.

PRECONCEPTUAL DESIGN STUDIES REPORT

APPENDIX B3

POWER CONVERSION SYSTEM STUDY

Note (1): **Previously Issued as an AREVA Contractor-Supplied Document, No. 38-9049582-000.**

Note (2): **Pagination is same as original document. Page numbering was altered to be consistent with appendix numbering requirements.**

Pratt & Whitney
A United Technologies Company

Team Product Document

GO Number	S/A Number	Page 1 of	Total Pages	Rev. Ltr/Chg. No. See Summary of Chg.	Number
98936	10000	33	33	C	EID-09388

Program Title Next Generation Nuclear Plant (NGNP) with Hydrogen Production

Document Title NGNP with Hydrogen Production Power Conversion System (PCS) Special Study

Document Type Engineering Information Document	Related Documents N/A
Original Issue Date: April 19, 2007 Release Date: RELEASE 04-27-07 NL	Approvals: Michael W. McDowell Date:
Prepared By/Date: Gregory A. Johnson, Ph.D., P.E. Dept.: 938 Mail/Addr: RLA13	AREVA NP Inc. an AREVA and Siemens company 38-9049582-000
IR&D Program? Yes ☐ No ☒ If Yes, Enter Authorization No.	

Distribution

*	Name	Mail Addr.
	Terrence H. Murphy	RLA-11
	Ricky L. Howerton	RLA-11
*	George M. O'Connor	RLA-11
*	William R. Determan	RLA-11
*	Cheng-Yi Lu	RLA-13
*	R. Dale Rogers	RLA-11
*	Robert Z. Litwin	RLA-11
*	Andrew J. Zillmer	RLA-13
*	Nathan J. Hoffman	RLA-11

* Complete Document
No Asterisk, Title Page/Summary or Change Page Only.

Abstract

A study was performed to examine power conversion system (PCS) options for the next generation nuclear plant (NGNP), a very high temperature gas-cooled reactor (VHTGR). The purpose of the study was to provide insight into which PCS should be used and how should it be coupled to the reactor: direct or indirect. Seven PCSs were examined: direct helium Brayton, indirect helium Brayton, supercritical CO2 (SCCO2), cascaded SCCO2, combined cycle gas turbine (CCGT), subcritical steam-Rankine, and supercritical steam-Rankine with double reheat. The results of the study show that the SCCO2 cycles a very promising and warrant further development, but the relative immaturity precludes it as a short term option. Further, the results call in to question the wisdom of further pursuit and development of a helium Brayton cycle. The best short term option were the steam-Rankine cycles. The supercritical steam-Rankine cycle gave the best performance of the two. The CCGT was the most costly and provided little performance advantage over the supercritical steam-Rankine cycle. Issues associated with closed loop operation, high-temperature compressor inlet temperature and potential nitriding from the He/N2 working fluid cast uncertainty on the maturity of this cycle.

Reserved for Proprietary/Legal Notice

Form 651-F REV. 8-29-05

EXPORT CONTROLLED

Supporting Document Summary of Change

No.
Page 1 of 1

Rev.	Summary of Change	Approvals and Date
C	Grammatical correction made to section 2.1.3. Added potential CO2 ingress issue to section 2.3.5. Slight rewording of sections 2.4.5 and 2.5.4. Added a List of Tables to the Table of Contents.	
B	Update Table 17 and section 3.2 to change the CCGT schedule rating based upon additional technical information.	
A	Update section 3.0 to make explanation consistent with Tables 16 and 17. Numerous minor editorials.	

Form 734-C-1 Rev. 4-97

Note: TOC page numbering key: page 7 = B3-7 etc

TABLE OF CONTENTS

LIST OF FIGURES

LIST OF TABLES

1.0 Introduction

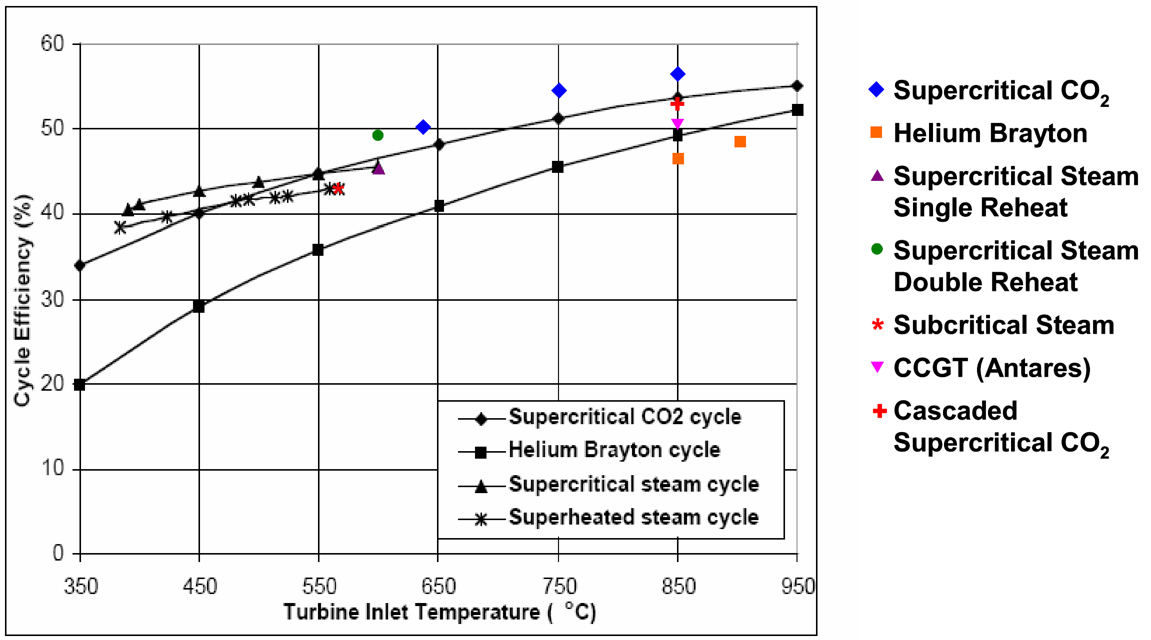
The power conversion system (PCS) special study will examine two closely related questions: what type of PCS should be used, a Brayton cycle, a steam-Rankine cycle, a combined cycle gas turbine (CCGT), or a supercritical CO_2 (SCCO2) cycle?; and how should it be coupled to the reactor, direct or indirect? Considerations driving this study are: system performance; flexibility and operability; adaptability of existing technology; technology maturity; deployment schedule, system costs including development, capital, and operation and maintenance; reliability; availability; maintainability; etc.

Further, the relationship between the NGNP and a commercial plant must be considered. The NGNP must serve both electricity (PCS) and the hydrogen plant. The NGNP conditions are driven largely by hydrogen process. However, the commercial electricity plant would likely have different conditions. Also, the optimum PCS for the commercial plant may not be the same as the optimum PCS for the NGNP.

The general approach taken for this study is to develop models of each of the candidate power conversion systems using two software packages: ChemCad and GateCycle. Start with previously optimized cycles when available. Size components; develop costs; determine technology maturity; identify issues such as integration, materials, safety, flexibility, etc; then to evaluate and report.

1.1 Cycle Performance Overview and Trends

Figure 1 is an overlay of PCS performance calculated during this study on top of PCS performance reported by MIT[1]. This figure is included to show the trends of the PCS cycles considered during this study. Note how the steam-Rankine system performance calculated in this study agrees with the MIT values. The performance of the supercritical CO_2 cycle calculated during this study tracks with but is greater than the MIT values. This is most likely due to different assumptions in turbomachinery efficiencies and recouperator effectiveness. For the helium Brayton cycles, the performance calculated during this study is less than, but still tracks with the MIT values. Also included in this figure are performance estimates for a supercritical steam-Rankine cycle with double reheat, a CCGT which is the baseline for the AREVA ANTARES concept, and a cascaded supercritical CO_2 cycle which will be discussed in greater detail later in this report.



Plot from: Driscoll, M.J., Report No: MIT-GFR-019, "Interim Topical Report Supercritical CO2 Plant Cost Assessment", September 2004, Center for Advanced Nuclear Energy Systems, MIT Nuclear Engineering Department

ChemCad used to model supercritical CO_2 and helium Brayton cycles

GateCycle used to model steam cycles

Figure 1. PCS Cycle Performance Trends.

1.2 Ground Rules and Assumptions

The following ground rules and assumptions were used during this study and are illustrated in **Figure 2**.

- 565 MWt helium gas cooled reactor power
- 900 °C reactor outlet temperature
- 500 °C reactor inlet temperature – desired
- 55 kPa pressure drop across core
- 55 kPa pressure drop across intermediate heat exchanger (IHX) – when used
- 5 MPa reactor inlet pressure – desired
- 1% heat loss
- 98% Generator Efficiency
- 1% BOP loads

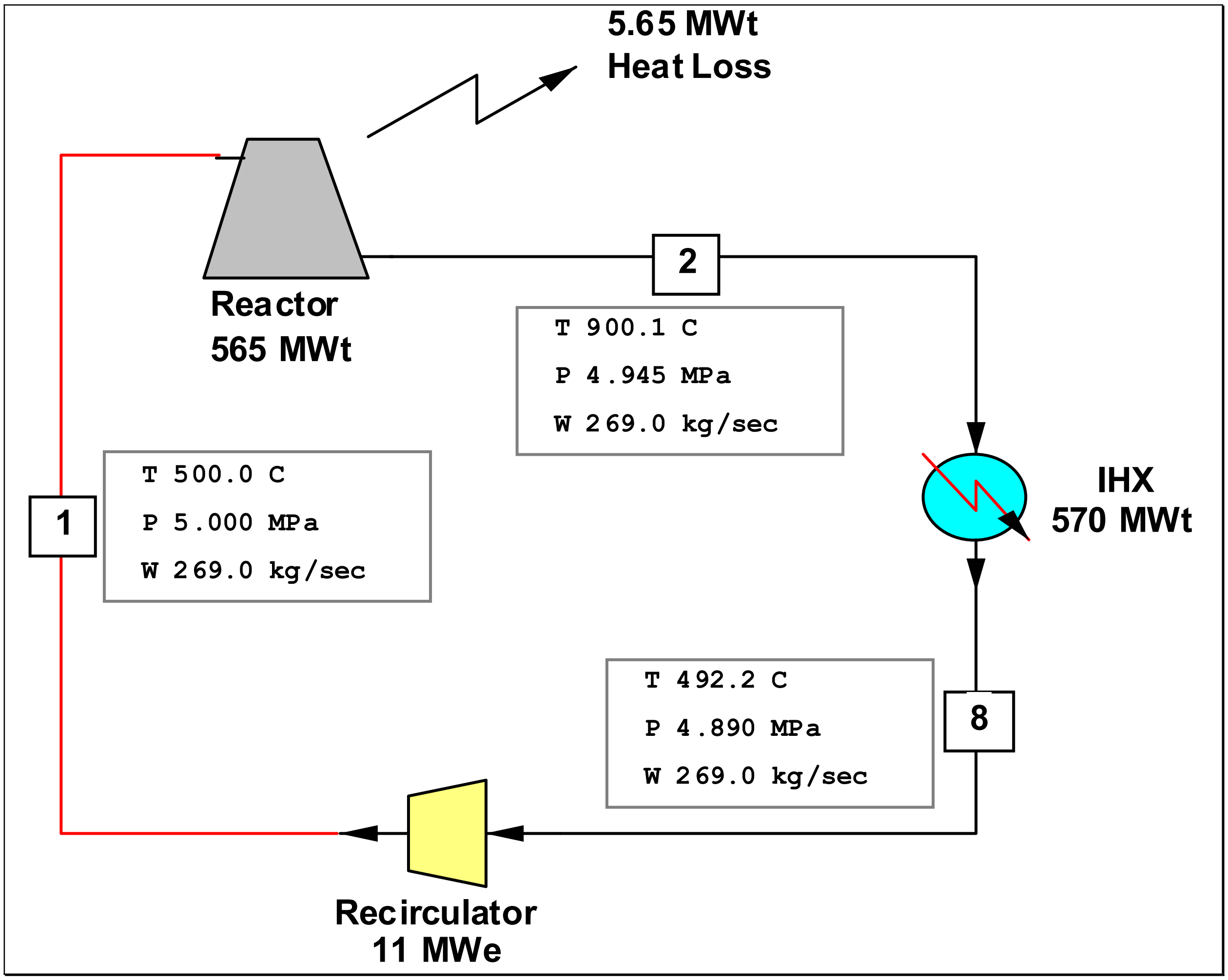


Figure 2. Ground rules and Assumptions for PCS Study.

1.3 PCS Power Balance and Efficiency Assessment

A common set of power balance and efficiency definitions were used throughout this study for cycle comparisons. These are defined as follows:

- Net Power to cycle = Prxtr - Qlosses + Precirc
- Gross Cycle Pwr = STurbine – SCompressors & Pumps
- Net Cycle Pwr = Gross Cycle Pwr - Genlosses- BOPlosses – Otherlosses
- Carnot Efficiency = (Thot – Tcold) ÷ Thot
- Gross Cycle Eff. = Net Cycle Pwr ÷ Gross Pwr
- % of Carnot = Gross Cycle Eff. ÷ Carnot Eff.
- Net Cycle Eff. = Net Cycle Pwr ÷ Reactor Pwr

1.4 PCS Maturity Assesment

A technology readiness level (TRL) scale such as that used in the aerospace industry was adapted for use in assessing PCS technical maturity[2,3]. This is illustrated in **Table 1**, where the very high temperature gas cooled reactor (VHTGR) PCS technology readiness level is mapped to the traditional aerospace technology readiness level.

Table 1. VHTGR PCS Technology Maturity Mapped to Aerospace Technology Maturity Assessment.

TRL	Aerospace Technology	VHTGR PCS Technology
9	Has an identical unit been successful on an operational mission (space or launch) in an identical configuration?	Has an identical unit been successful on a commercial operation in an identical configuration?
8	Has an identical unit been demonstrated on an operational mission, but in a different configuration/system architecture?	Has an identical unit been demonstrated on a commercial operation, but in a different system/configuration architecture?
	Has an identical unit been mission (flight) qualified but not operationally demonstrated (space or launch)?	Has an identical unit been successful on a pilot plant?
7	Has a prototype unit been demonstrated in the operational environment (space or launch)?	Has a prototype unit been demonstrated in the operational environment with demonstration of safety features?
6	Has a prototype unit been demonstrated in a relevant environment, on the target or surrogate platform?	Has a prototype unit been demonstrated in a relevant environment, on the target or surrogate platform?
5	Has a breadboard been demonstrated in a relevant (typical; not necessarily stressing) environment?	Has component/breadboard been demonstrated in a relevant (typical; not necessarily stressing) environment?
4	Has a breadboard been demonstrated in a laboratory (controlled) environment?	Has component/breadboard been demonstrated in a laboratory (controlled) environment?
3	Has analytical and experimental proof-of-concept been demonstrated?	Has analytical and experimental proof-of-concept been demonstrated?
2	Has a concept or application been formulated?	Has a concept or application been formulated?
1	Have basic principles been observed and reported?	Have basic principles been observed and reported?

1.5 Component Sizing and Cost Estimating Approach

The component sizing and cost estimating approach was to size and cost major components only. For heat exchangers, the type of heat exchanger (i.e. shell and tube, compact, etc.), the required heat transfer area and heat exchanger volume were roughed-out by a Rocketdyne heat exchanger expert. Heat exchanger costs were then estimated based on material requirements. For turbomachinery other than steam turbines, turbine and compressors sizes (diameters and number of stages) were roughed-out by Rocketdyne turbine and compressor experts. Volumes and costs were then inferred from gas-turbine cost and size data. For steam turbines, sizes were inferred from Siemens steam turbine sizes and costs were inferred from informal costs received from a steam turbine vendor. For building costs, floor space was estimated for the major components and a cost factor was applied based on recent Rocketdyne factory erection experience.

2.0 Power Conversion System Assessment

2.1 Direct Brayton Cycle

2.1.1 Description

The direct Brayton cycle is shown schematically in **Figure 3** and the temperature-entropy diagram is shown in **Figure 4**. Helium (He) working fluid is circulated through the reactor core picking up heat. The hot helium leaving the reactor is expanded through a turbine producing power. Turbine exhaust gas is passed through a recuperator where it transfers untapped thermal energy to the second stage compressor exhaust. Next, waste heat is rejected by passing the low pressure helium leaving the recuperator through the heat rejection heat exchanger (HRHX). The low pressure helium is partially compressed in the low pressure compressor; the heat of compression is removed by passing the intermediate pressure helium through an intercooler. The helium is then brought to full pressure by the high-pressure compressor, whereupon it passes through the high-pressure side of the recuperator where it captures thermal energy from the turbine exhaust. After leaving the recuperator, the high-pressure helium is circulated back through the reactor completing the cycle.

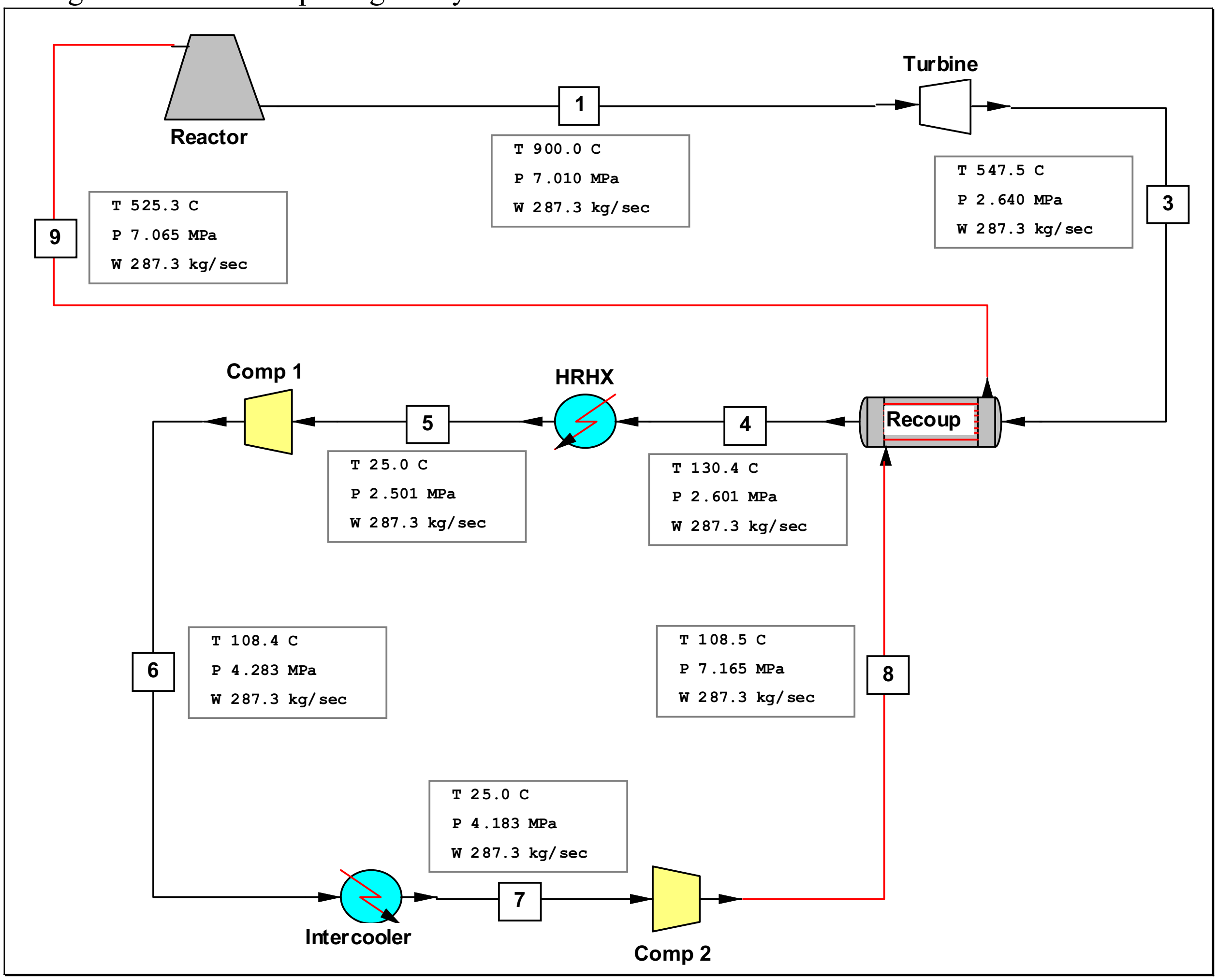


Figure 3. Direct Brayton cycle

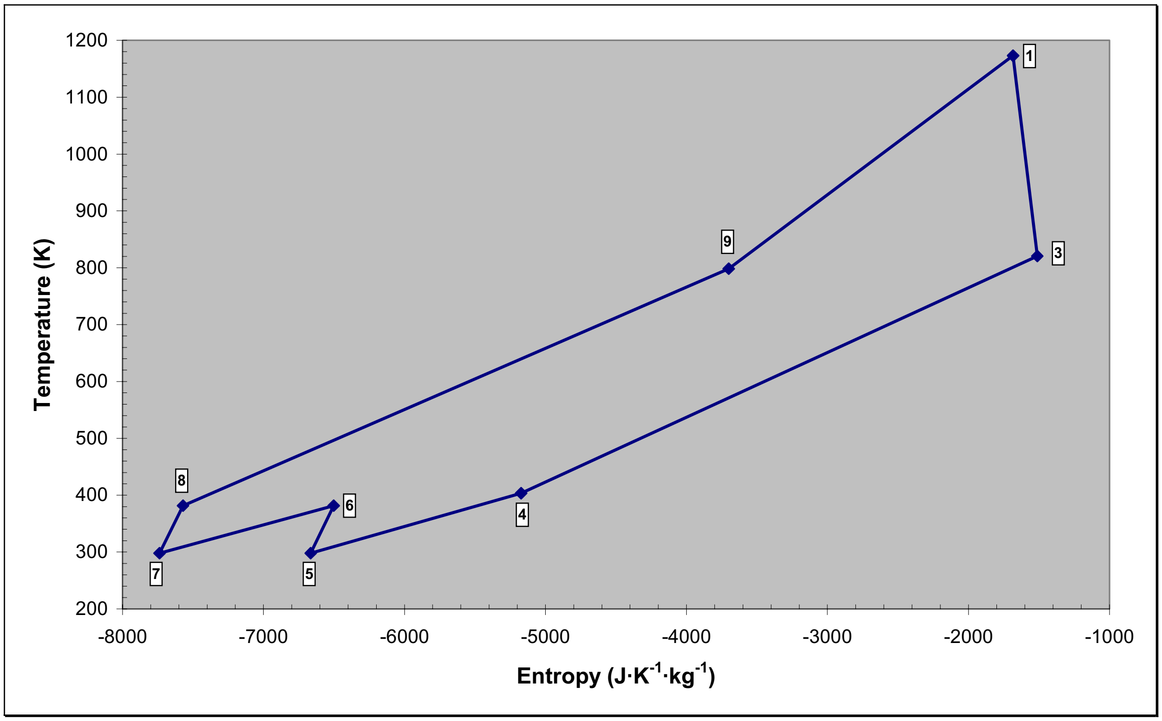


Figure 4. Direct Brayton Cycle T-S Diagram

2.1.2 Power Balance

The power balance and efficiencies for this cycle is shown in **Table 2**. This cycle produces a net plant output of 268.9 MWe from 565 MWt of reactor power for a net plant efficiency of 47.6%.

Table 2. Direct Brayton Cycle Power Balance.

	Power (MW)		Efficiency (%)
Reactor	565.0	Carnot Efficiency	74.6%
Heat Losses	-5.65	Gross Cycle Efficiency	49.6%
Net to Cycle	**559.4**	% of Carnot	66.5%
Turbine	530.1	**Net Cycle Efficiency**	**47.6%**
LP Comp	-125.9		
HP Comp	-127.0		
Gross Cycle Power	**277.2**		
Generator Losses	-5.5		
BOP Losses	-2.77		
Net Cycle Power	**268.9**		

2.1.3 Component Sizes and Costs

Estimates for the Component sizes and costs for the direct Brayton cycle are given in **Table 3**. The cost drivers are the recouperator and the turbomachinery. Because the cycle is highly recouperative, a large amount of heat transfer area is required. Volume is kept low by using a compact heat exchanger. A recognized difficulty with a helium Brayton cycle is the large size of the turbomachinery[4]. This is due to the low molecular weight of the working fluid.

Table 3. Direct Brayton Cycle Component Sizes and Costs.

Component	Heat Transfer Area (m^2)	Size (m^3)	Cost (Rel.)	Floor Space (m^2)
Heat Exchangers	**54,920.7**	**80.2**	**21.4**	**142.7**
Recouperator	46,253.4	53.6	15.8	89.3
HRHX	4,876.9	14.8	3.1	29.6
Intercooler	3,790.4	11.9	2.5	23.8
	Stages	**Size (m^3)**	**Cost (Rel.)**	**Floor Space (m^2)**
Turbomachinery		**564.4**	**57.3**	**130.6**
Turbine	6	419.3	42.6	57.9
LP Compressor	14	61.3	6.2	30.8
HP Compressor	19	83.8	8.5	41.9
Building Costs			**2.3**	
Total		**644.7**	**81.0**	**273.3**

2.1.4 Technology Readiness

The technology readiness assigned to the direct Brayton cycle is level 4 for the following reasons: Multiple small closed cycle Brayton engines have been demonstrated in laboratory environments. A 30 to 40 MWe size helium Brayton engine was demonstrated at Oberhausen, Germany. Large closed cycle Brayton engines have been analytically studied.

2.1.5 Issues

The primary issue or concern with using a Brayton cycle directly coupled to the reactor is spread of radioactive contamination throughout the PCS. This will result in potential increased radiation exposure to personnel around the power conversion equipment and will also present significant difficulties when the PCS needs to be opened for maintenance reasons.

The next concern with this PCS is that the high reactor outlet temperature will either require protection of the high-temperature ducting, perhaps with thermal liners.

Still another concern is the higher return temperature of 525 °C for the optimized cycle verses the desired return temperature of 500 °C. This will affect reactor vessel cooling. Another parameter affecting the reactor vessel is the higher operating pressure of 7 MPa for the optimized cycle verses the desired lower operating pressure of 5 MPa. This will result in a thicker, heavier vessel and thicker walls of the ducting increasing system costs. Efforts to correct these result in lower performance. For instance, by lowering the operating pressure to 5 MPa results in a loss of power of about 10 MWe and a loss of system efficiency of about 1.5%. This lowering of operating pressure will also result in larger PCS components. Likewise, the lowering of the return temperature to 500 °C by increasing the cycle pressure ratio also results in degraded performance. Loss in performance has been estimated to be about 4 MWe of power and 0.5% in efficiency.

Lastly, the turbomachinery of the helium Brayton cycle is rather large due to high specific energy of low molecular weight gas resulting in packaging issues and difficulties.

2.2 Indirect Brayton Cycle

2.2.1 Description

The indirect Brayton cycle is shown schematically in **Figure 5**, and the temperature-entropy diagram is shown in **Figure 6**. The cycle is exactly like the direct Brayton cycle previously discussed except that helium working fluid circulated through the reactor core picking up heat is sent to an intermediate heat exchanger (IHX) where thermal energy is transferred to the PCS. On the secondary side of the IHX, heat is transferred to He working fluid where it is then expanded through a turbine producing power. Turbine exhaust gas is passed through a recuperator where it's untapped thermal energy is captured by the second stage compressor exhaust. Next, waste heat is rejected by passing the low pressure helium leaving the recuperator through the heat rejection heat exchanger (HRHX). The low pressure helium is partially compressed in the low pressure compressor; the heat of compression is removed by passing the intermediate pressure helium through an intercooler. The helium is then brought to full pressure by the high-pressure compressor, whereupon it passes through the high-pressure side of the recuperator where it captures thermal energy from the turbine exhaust. After leaving the recuperator, the high-pressure helium is circulated back through IHX completing the cycle.

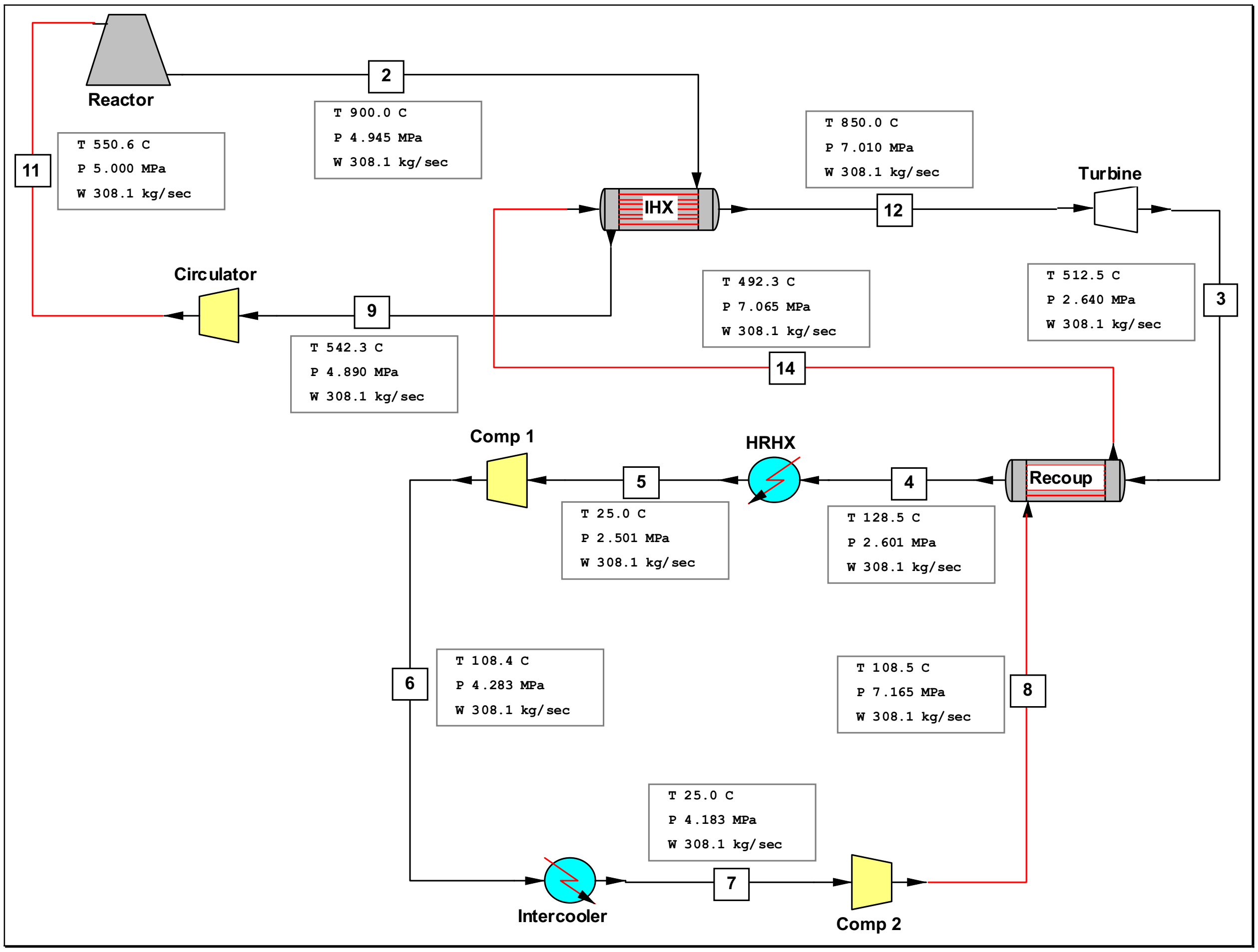


Figure 5. Indirect Brayton Cycle

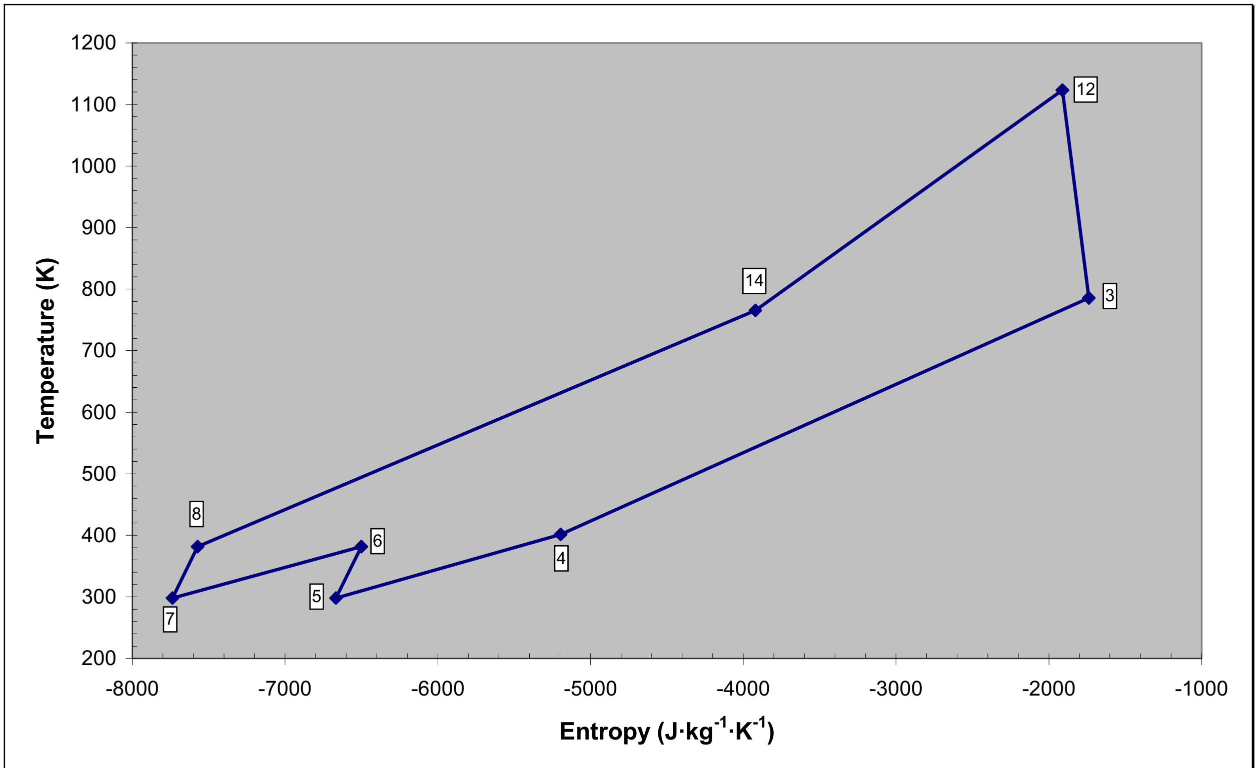


Figure 6. Indirect Brayton TS Diagram

2.2.2 Power Balance

The power balance for this cycle is shown in **Table 4**. This cycle produces a net plant power of 251.6 MWe from 565 MWt of reactor power for a net plant efficiency of 44.5%. This cycle is less efficient that the direct Brayton cycle due to the lower turbine inlet temperature and more importantly, the power consumed by the circulator in the primary loop.

Table 4. Indirect Brayton Cycle Power Balance.

	Power (MW)		Efficiency (%)
Reactor	565.0	Carnot Efficiency	73.5%
Heat Losses	-5.65	Gross Cycle Efficiency	47.7%
Circulator Power	13.39	% of Carnot	64.9%
Net to Cycle	**572.7**	**Net Cycle Efficiency**	**44.5%**
Turbine	544.4		
LP Comp	-135.0		
HP Comp	-136.2		
Gross Cycle Power	**273.2**		
Generator Losses	-5.46		
BOP Losses	-2.73		
Circulator Power	-13.39		
Net Cycle Power	**251.6**		

2.2.3 Component Sizes and Costs

Estimates for the Component sizes and costs for the indirect Brayton cycle are given in Table 5. Like the direct Brayton cycle, the cost drivers are the recouperator and the turbomachinery. The costs are similar but slightly greater than the direct Brayton cycle costs due to the slightly larger equipment required.

Table 5. Indirect Brayton Cycle Component Sizes and Costs.

Component	Heat Transfer Area (m^2)	Size (m^3)	Cost (Rel.)	Floor Space (m^2)
Heat Exchangers	**63,988**	**220.2**	**27.1**	**160.4**
IHX	8,677	138.4	5.4	14.4
Recouperator	46,037	53.3	15.8	88.9
HRHX	5,177	15.7	3.3	31.4
Intercooler	4,097	12.8	2.7	25.7
	Stages	**Size (m^3)**	**Cost (Rel.)**	**Floor Space (m^2)**
Turbomachinery		**580.5**	**58.9**	**133.9**
Turbine	6	427.6	43.4	58.7
LP Compressor	14	64.6	6.6	31.8
HP Compressor	19	88.2	9.0	43.4
Building Costs			**2.5**	
Total		**800.7**	**88.5**	**294.3**

2.2.4 Technology Readiness

Like the direct Brayton cycle, the technology readiness assigned to the indirect Brayton cycle is level 4 for the identical reasons previously discussed.

2.2.5 Issues

One concern with this PCS is that the high turbine inlet temperature will require protection of the IHX shell and the high-temperature ducting, perhaps with thermal liners.

Still another concern is the higher return temperature of 550 °C for the optimized cycle verses the desired return temperature of 500 °C. This will affect reactor vessel cooling and also increase circulator power demand from 11 MWe to 13.4 MWe. Further, the higher operating pressure of 7 MPa for the optimized cycle verses the desired lower operating pressure of 5 MPa for the primary circuit raises fabricability concerns for the IHX. Raising the primary pressure to 7 MPa will result in a thicker, heavier reactor vessel and thicker walls of the ducting increasing system costs. Efforts to correct these result in lower performance. For instance, by lowering the operating pressure to 5 MPa results in a loss of power of about 10 MWe and a loss of system efficiency of about 1.5%. This lowering of operating pressure will also result in larger PCS components. Likewise, the lowering of the return temperature to 500 °C by increasing the cycle pressure ratio also results in degraded performance. Loss in performance has been estimated to be about 8.5 MWe of power and 1.5% in efficiency.

Lastly, the turbomachinery of the helium Brayton cycle is rather large due to high specific energy of low molecular weight gas resulting in packaging issues and difficulties.

2.3 Supercritical Carbon Dioxide (CO_2) Cycle

2.3.1 Description

The supercritical CO_2 (SCCO2) cycle is shown in **Figure 7**, and the T-S diagram for this cycle is shown in **Figure 8**. Here, high-pressure carbon dioxide is passed through the secondary side of the IHX picking-up thermal energy from the primary. The hot, high-pressure CO_2 is expanded through a turbine producing power. The turbine exhaust is passed through two recouperators transferring thermal energy to the compressor and recompressor exhaust. The flow of the low pressure CO_2 leaving the second recouperator is split. This is done in such a manner to better match the heat capacities so that the maximum amount of recouperation is achieved. Some of the low pressure CO_2 is sent through a HRHX giving-up cycle waste heat. The CO_2 at this point in the cycle is very near the critical point and behaves much differently than an ideal gas. It is easier to compress which in-turn results in a much more efficient cycle and much smaller turbomachinery. This stream is then compressed and passed through the high pressure side of the second recouperator. The second split stream is sent directly to a recompressor where it is pressurized and mixed with the high-pressure stream exiting the second recouperator. This mixed stream is passed through the high-pressure side of the first recouperator extracting more energy from the turbine exhaust. After leaving the recouperator, the CO_2 is returned to the IHX completing the cycle.

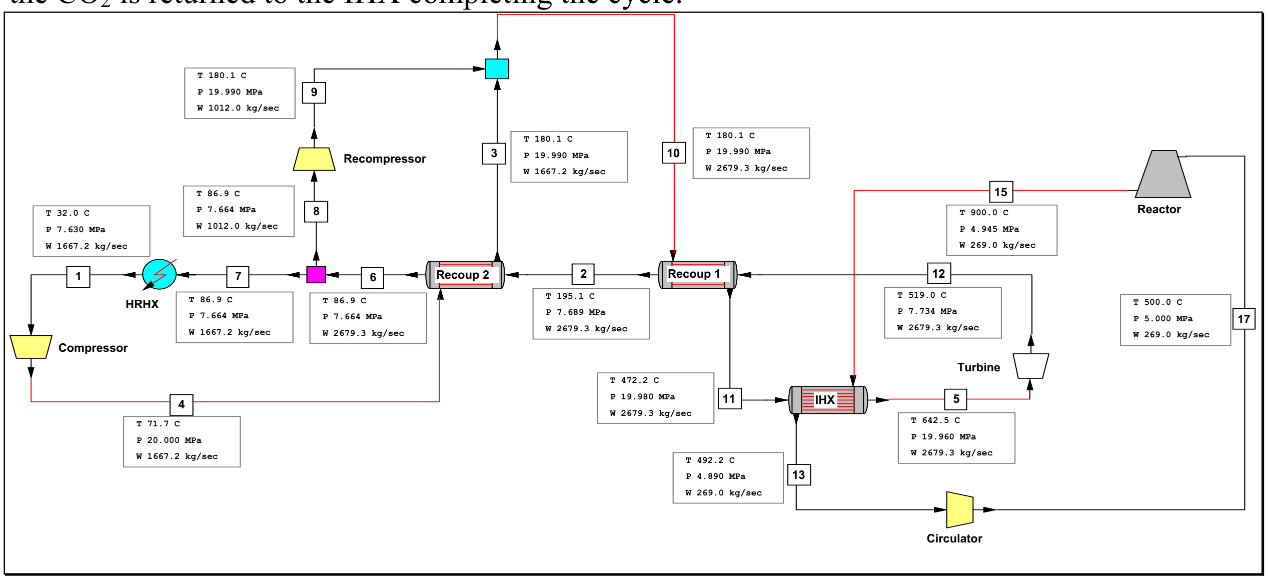


Figure 7. Supercritical CO_2 Cycle

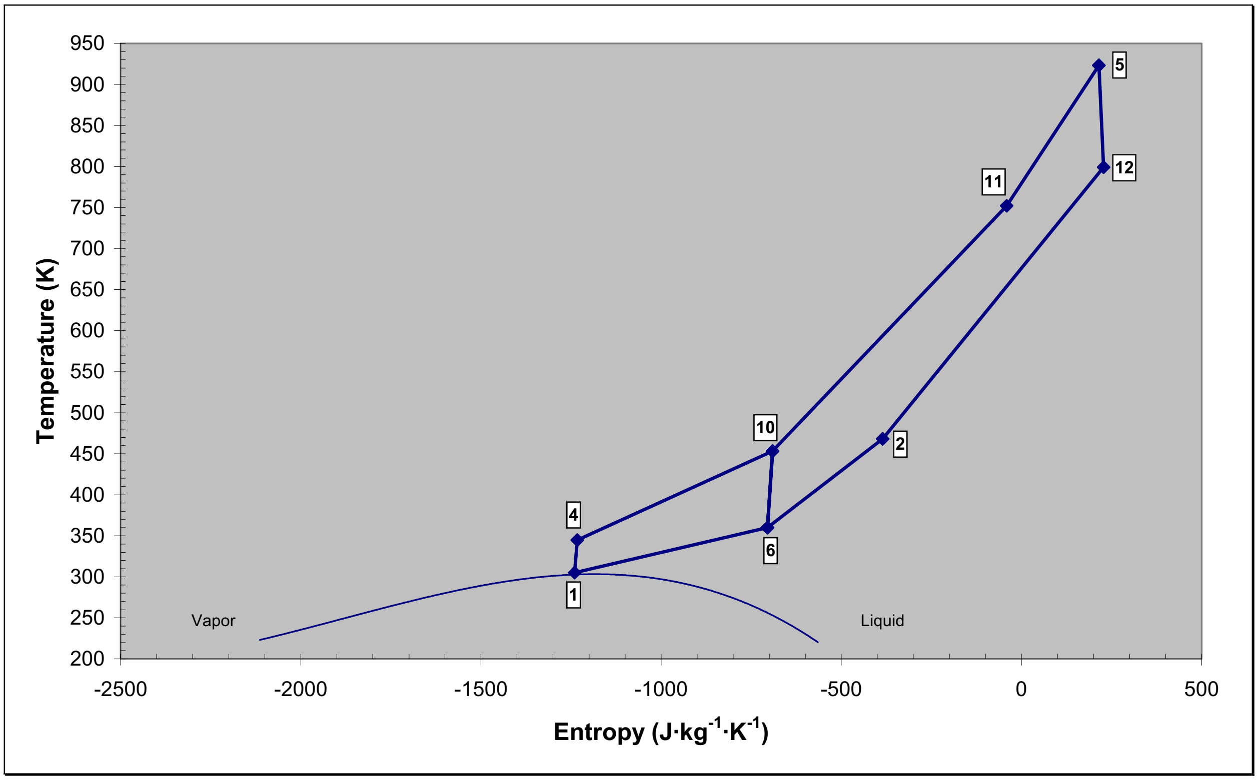


Figure 8. Supercritical CO_2 Cycle TS Diagram

2.3.2 Power Balance

The power balance for the supercritical CO_2 cycle is shown in **Table 6**. This cycle produces a net plant power output of 267.8 MWe from 565 MWt of reactor power for a net plant efficiency of 47.4%.

Table 6. Supercritical CO_2 Cycle Power Balance.

	Power (MW)		Efficiency (%)
Reactor	565.0	Carnot Efficiency	66.7%
Heat Losses	-5.65	Gross Cycle Efficiency	50.4%
Circulator Power	10.96	% of Carnot	75.6%
Net to Cycle	**570.3**	**Net Cycle Efficiency**	**47.4%**
Turbine	397.7		
Comp	-42.8		
Recomp	-67.5		
Gross Cycle Power	**287.4**		
Generator Losses	-5.75		
BOP Losses	-2.87		
Circulator Power	-10.96		
Net Cycle Power	**267.8**		

2.3.3 Component Sizes and Costs

Estimates for the component sizes and costs for the supercritical CO_2 cycle are given in **Table 7**. The cost drivers are the IHX and the recouperators. Of particular note is the very low costs of the turbomachinery. This is due to the nature of the supercritical CO_2 cycle of compressing the working fluid near the critical point and also due to the high molecular weight of the working fluid. Both of these factors keep the turbomachinery compact and relatively inexpensive.

Table 7. Supercritical CO2 Component Sizes and Costs

Component	Heat Transfer Area (m^2)	Size (m^3)	Cost (Rel.)	Floor Space (m^2)
Heat Exchangers	**60,976**	**144**	**30.7**	**136.8**
IHX	4,296	69.2	9.2	7.2
HP Recouperator	34,400	41.0	12.1	68.4
LP Recouperator	17,200	18.7	5.5	31.2
HRHX	5,080	15.0	3.9	30.0
	Stages	**Size (m^3)**	**Cost (Rel.)**	**Floor Space (m^2)**
Turbomachinery		**4.7**	**0.5**	**4.3**
Turbine	3	1.7	0.2	1.2
Compressor	2	0.6	0.1	0.8
Recompressor	4	2.4	0.2	2.3
Building Costs			**1.2**	
Total	**60,976**	**148.7**	**32.4**	**141.1**

2.3.4 Technology Readiness

The technology readiness assigned to the supercritical CO_2 cycle is level 3 for the following reasons: It could be argued that the supercritical CO_2 cycle is a variant of the Brayton cycle. However, Brayton systems using CO_2 have not been operated but have only been studied analytically. Further, compressing the fluid near the critical point has not been demonstrated. However, similar type hardware has been demonstrated with different fluids. For instance, turbo-pumps used on liquid propellant rocket engines are typically pumped or compressed through (near) the critical point.

2.3.5 Issues

The first observation regarding the supercritical CO_2 cycle is that the temperature differentials between the primary and the PCS are mismatched. The primary temperature difference is 400 °C by design. However, the PCS temperature difference optimizes to 170 °C. This results in a low turbine inlet temperature of 642 °C, preventing the PCS from achieving its full thermodynamic potential.

Second, CO_2 has a lower thermal conductivity than helium; 20% of that of helium. This results in large heat exchangers.

Another issue that has been raised, but might be inconsequential, is the potential for alloy ignition from free O_2 from CO_2 decomposition. At elevated temperatures, CO_2 will partially dissociate to CO and O_2 (see **Figure 9**). This decomposition will be suppressed at elevated pressures. The potential for this free O_2 to react and ignite molten metal (e.g. from a wiped bearing) should be considered.

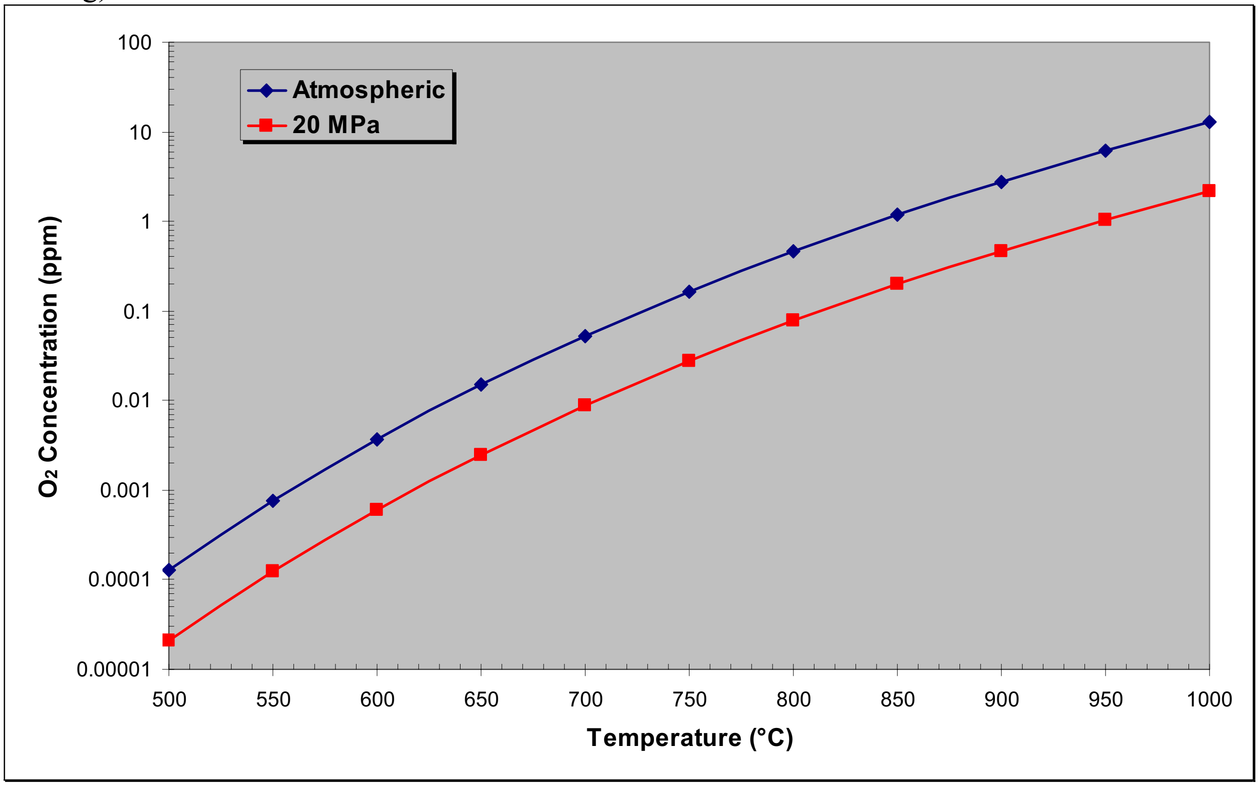


Figure 9. Oxygen Concentration due to CO_2 Dissociation.

Further, because the compressor operates near the critical point, a specialty compressor design will most likely be required. Similar type machines have been developed for other applications such as high power density rocket engine turbo-pumps (see **Figure 10** and **Figure 11**).

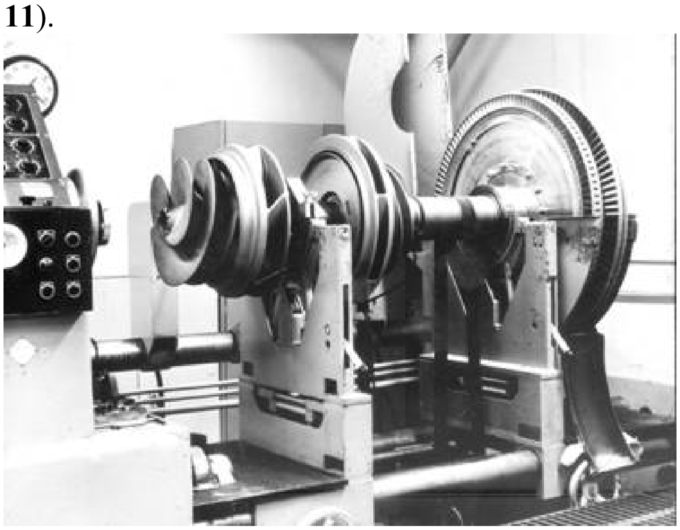


Figure 10. Turbo-pump for Saturn V main engine 36 inch pitch diameter, 77,000 HP (57 MW) Similar to supercritical CO_2 power generation system

Figure 11. High Power Density Rocket Engine Turbo-pump Experience

Also at issue is the unknown operational behavior of a supercritical CO_2 system. How it behaves during start-up, shutdown and transients has yet to be determined. And finally, the large pressure difference between the primary coolant and the PCS working fluid (approximately 15 MPa) at high temperature will make building an intermediate heat exchanger (IHX) a very major challenge.

Further, the potential of CO_2 ingress from an IHX leak into the primary system should be addressed during system design.

2.4 Cascaded Supercritical CO_2 Cycle

2.4.1 Description

In order to utilize the promising high efficiency of the supercritical CO_2 cycle and maintain the 400 °C primary circuit temperature difference, two supercritical subsystems are coupled together as shown in **Figure 13**. The hot reactor coolant passes through the first IHX raising the temperature of the CO_2 entering the turbine to 850 °C. The reactor coolant then proceeds to the second IHX were the remainder of the heat is transferred to the second supercritical CO_2 cycle subsystem raising its turbine inlet temperature to 609 °C. In this manner, the PCS efficiency can increase while maintaining the primary circuit temperature differential at 400 °C.

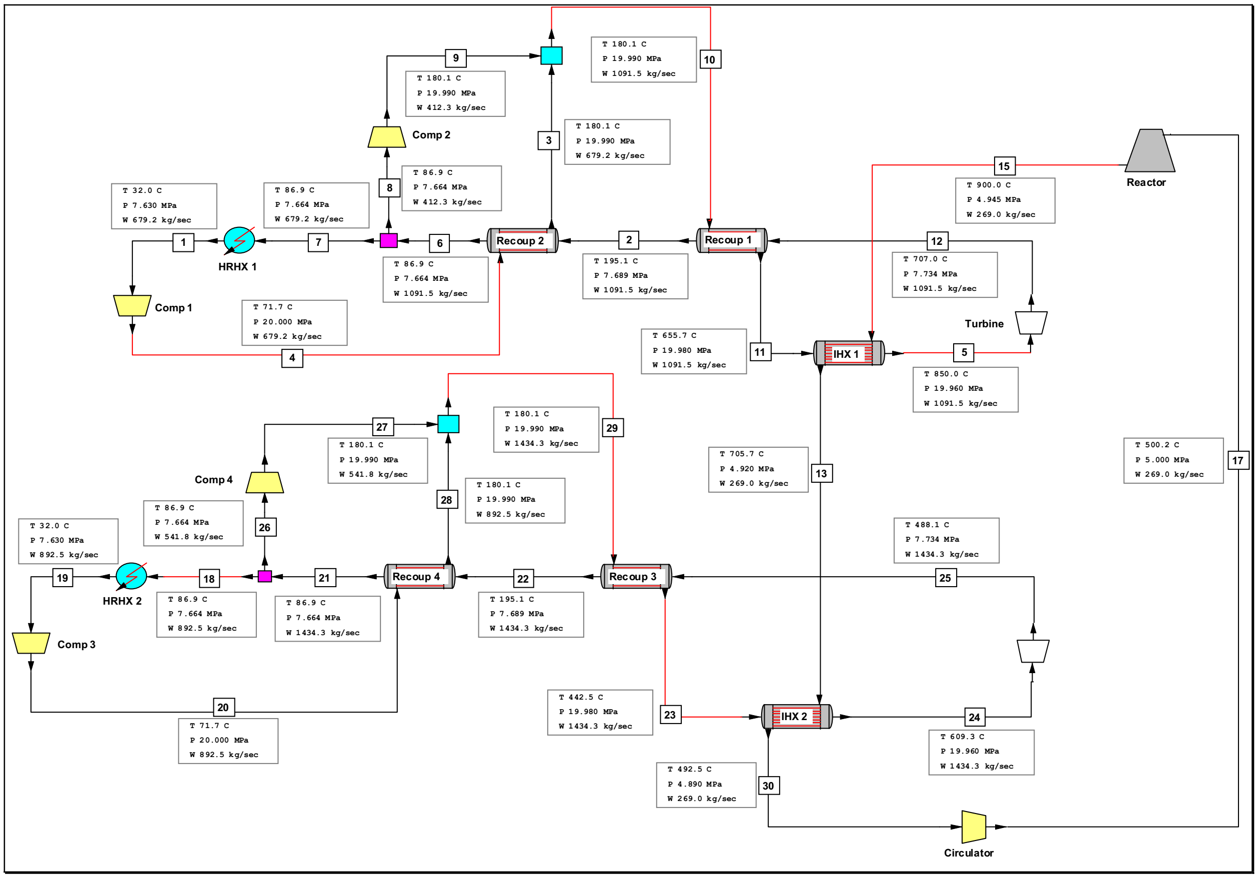


Figure 12. Cascaded Supercritical CO_2 Cycle.

2.4.2 Power Balance

The power balance for the cascaded supercritical CO_2 cycle is shown in **Table 8**. This cycle produces a net plant power output of 281.6 MWe from 565 MWt of reactor power for a net plant efficiency of 49.8%.

Table 8. Cascaded Supercritical CO_2 Cycle Power Balance.

	Power (MW)		Efficiency (%)
Reactor	565.0	Carnot Efficiency	72.8%
Heat Losses	-5.65	Gross Cycle Efficiency	52.9%
Circulator Power	10.96	% of Carnot	72.6%
Net to Cycle	**570.3**	**Net Cycle Efficiency**	**49.8%**
Turbines	406.3		
Comps	-40.6		
Recomps	-64.1		
Gross Cycle Power	**301.7**		
Generator Losses	-6.03		
BOP Losses	-3.02		
Circulator Power	-10.96		
Net Cycle Power	**281.6**		

2.4.3 Component Sizes and Costs

Estimates for the component sizes and costs for the cascaded supercritical CO_2 cycle are given in **Table 9**. Again, the cost drivers are the IHX and the recouperators and not the turbomachinery. Although the number of components approximately doubles, the costs are about the same as the simple SCCO2 because the costs of the each heat exchangers are roughly half of their SCCO2 counterpart. Turbomachinery costs are more but since they are not a cost driver, they do not make a significant impact on overall costs.

Table 9. Cascaded Supercritical CO_2 Component Sizes and Costs.

Component	Heat Transfer Area (m^2)	Size (m^3)	Cost (Rel.)	Floor Space (m^2)
Heat Exchangers	**68,073**	**184**	**29.6**	**143.5**
IHX	3,780	61	2.4	7.2
HP Recouperator	21,500	26	7.5	42.8
LP Recouperator	7,009	8	6.0	13.3
HRHX	2,089	6	1.7	12.3
IHX	2,934	47	1.9	7.2
HP Recouperator	16,942	21	5.9	34.2
LP Recouperator	12,153	10	3.0	17.2
HRHX	1,666	5	1.3	9.4
	Stages	**Size (m^3)**	**Cost (Rel.)**	**Floor Space (m^2)**
Turbomachinery		**7.4**	**0.7**	**8.8**
Turbine	6	1.7	0.17	1.4
Compressor	4	0.5	0.05	0.8
Recompressor	8	1.7	0.17	2.3
Turbine	5	1.2	0.12	1.1
Compressor	3	0.5	0.05	0.8
Recompressor	7	1.8	0.19	2.3
Building			**1.3**	
Total		**191.0**	**31.7**	**152.3**

2.4.4 Technology Readiness

The technology readiness for the cascaded supercritical CO_2 cycle is assessed as level 3 for the same reasons and rationale previously discussed for the supercritical CO_2 cycle.

2.4.5 Issues

The issues associated with the cascaded version of the supercritical CO_2 cycle are the same issues previously discussed for the uncascaded version. However, the issue of IHX construction to retain the high pressure at the higher IHX temperature is exacerbated. Further, operability of two cycles operating in series needs to be addressed.

2.5 Combined Cycle Gas Turbine

2.5.1 Description

This cycle is a variant of the mature, fossil-fuel fired, open-cycle combined cycle plants. In this cycle, a mixture of about 80 wt% nitrogen and 20 wt% helium is used as the PCS working fluid in a closed cycle. Referring to **Figure 14**, the temperature of the high pressure N_2/He working fluid is raised to 850 °C by passing it through the IHX. The hot, high pressure gas is expanded through a turbine. The turbine exhaust gas is passed through a heat-recovery-steam-generator (HRSG) producing steam to drive a conventional subcritical steam-Rankine power plant. The HRSG exhaust is then repressurized and returned to the IHX.

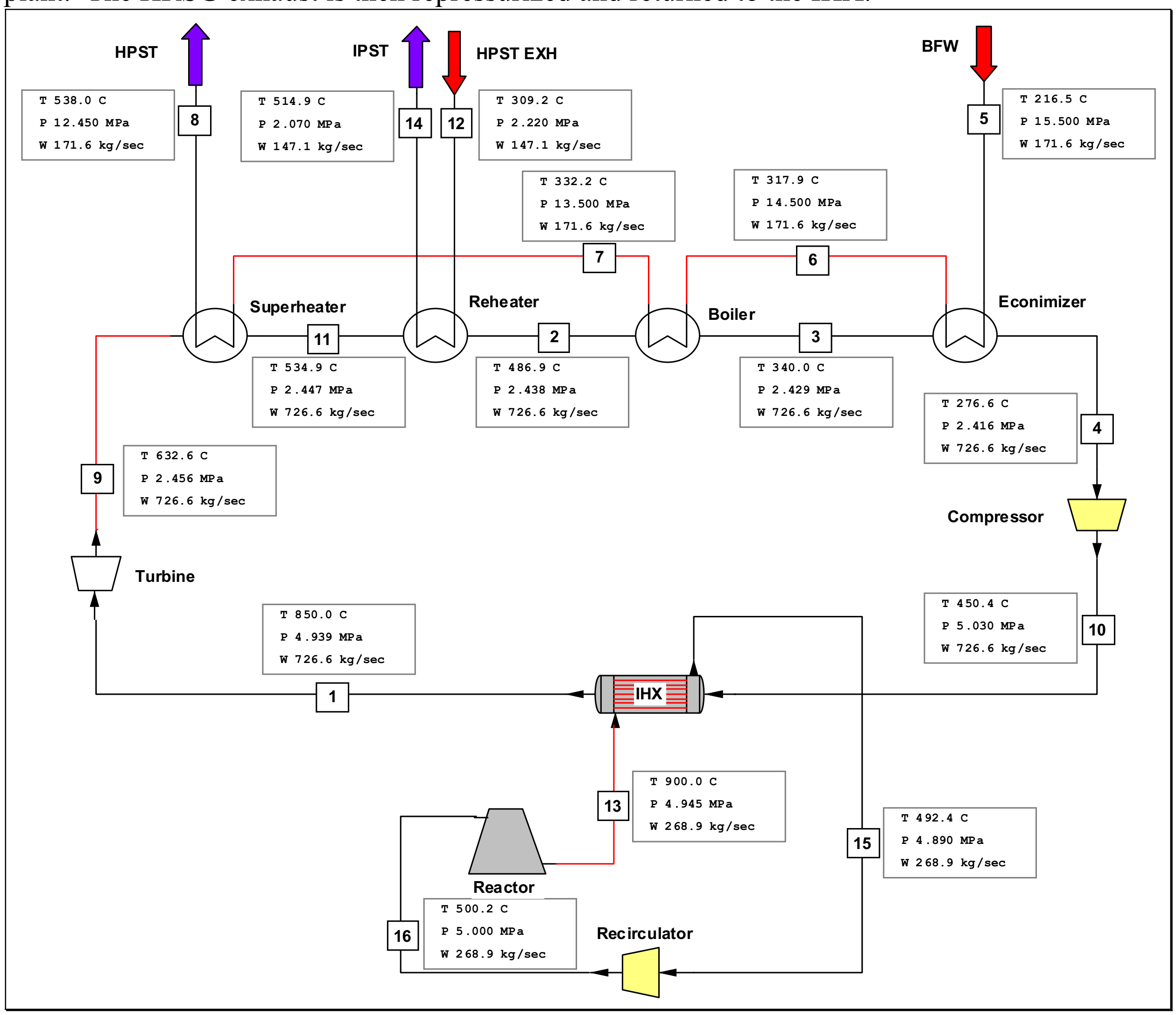


Figure 13. Combined Cycle Gas Turbine Cycle.

2.5.2 Power Balance

The power balance for the CCGT is shown in **Table 10**. This cycle produces a net plant output of 270.9 MWe from 565 MWt of reactor power for a net plant efficiency of 47.9%.

Table 10. CCGT Power Balance.

	Power (MW)		Efficiency (%)
Reactor	565.0	Carnot Efficiency	72.8%
Heat Losses	-5.65	Gross Cycle Efficiency	51.6%
Circulator Power	10.97	% of Carnot	70.9%
Net to Cycle	**570.3**	**Net Cycle Efficiency**	**47.9%**
Turbine	314.5		
Compressor	-243.1		
Steam Turbine	223.1		
Gross Cycle Power	**294.5**		
Generator Losses	-5.89		
BOP Losses	-2.94		
Circulator Power	-10.97		
Feed & Condensate	-3.82		
Net Cycle Power	**270.9**		

2.5.3 Component Sizes and Costs

The component sizes and costs for the CCGT are shown in **Table 11**. Cost drivers are the IHX, condenser, and the turbomachinery. The IHX is large due to the close approach temperature of 50 °C and the lower thermal conductivity (compared to He) of the He/N_2 secondary working fluid. Also, because of the low molecular weight (compared to air) and the high compressor inlet temperature, the compressor is large for a low compression ratio compressor. The extra costs of the gas-turbine components coupled with the costs of the steam turbine drives up the costs of this cycle.

Table 11. CCGT Component Sizes and Costs.

Component	Heat Transfer Area (m^2)	Size (m^3)	Cost (Rel.)	Floor Space (m^2)
Heat Exchangers	**49,336**	**796.3**	**45.9**	**107.5**
IHX	25,657	415	16.1	43.3
Superheater	1,525	25	1.2	3.6
Reheater	1,460	24	1.2	3.6
Evaporator	2,278	35	1.4	3.6
Econimizer	3,601	62	2.8	7.2
HP Feedwater Heaters	2,149	34	3.4	11.8
LP Feedwater Heaters	665	11	1.0	6.5
Condenser	12,000	191	18.9	27.8
	Stages	**Size (m^3)**	**Cost (Rel.)**	**Floor Space (m^2)**
Turbomachinery		**916**	**51.7**	**169.9**
Turbine	4	190.0	19.3	29.9
Compressor	12	85.5	8.7	36.5
Steam Turbine		640.5	22.9	103.6
Feed & Condensate			0.8	
Building			**2.4**	
Total		**1,712**	**100.0**	**277.4**

2.5.4 Technology Readiness

The technology readiness of the CCGT is assessed as a level 6. Air fossil fuel fired CCGTs have seen extensive commercial applications. Further, the steam bottoming cycle is very mature. The closed CCGT has not been demonstrated however, but closed loop Brayton cycles have been demonstrated at a lower power levels. Also, a different working fluid coupled with a different pressure ratio; conventional gas-turbines operate at a pressure ratio of about 16:1, the pressure ratio of the CCGT is just over 2:1; a higher inlet pressure and a higher compressor inlet temperature, will necessitate a modified turbomachinery design.

2.5.5 Issues

Several issues have been identified for this cycle. As with the other high-temperature cycles, a thermal liner will be needed in the hot ducting and hot parts of the IHX to protect the structure from exceeding temperature limits. The PCS system pressure is matched to the primary circuit pressure to keep stresses in the IHX low. Because a helium-nitrogen mix is used as the PCS working fluid, the Potential of nitriding in high temperature components is a concern. Nitriding tends to harden and embrittle metals and is a concern where the loss of ductility would result in a failure. A survey of the Literature indicates that the actual severity of nitriding may be less than the theoretical. Methods for inhibiting nitriding will need to be explored. One possibility is formation of a passive layer either as a coating or as part of the metal surface structure (e.g. the passive oxide layer that passivates stainless steel). Another option is to use a gas mixture that does not contain nitrogen such as a noble gas mixture like helium-argon. As previously mentioned, the non-standard pressure ratio coupled with the higher compressor inlet temperature and the different working fluid will result in a new gas-turbine design.

There are issues associated with the steam bottoming cycle as well. This cycle is very mature and it has well known issues such as corrosion. Further, unlike the open cycle combined-cycle, the steam generators will require thicker pressure vessels to withstand the almost 30 atmospheres of differential pressure it must contain.

2.6 Subcritical Steam Rankine Cycle

2.6.1 Description

The subcritical steam-Rankine cycle is shown in **Figure 15**. Here the primary coolant leaving the reactor is sent through a steam generator to produce steam for a conventional subcritical steam cycle with reheat. Primary and reheat steam temperatures are 565 °C. Condenser pressure is 4.75 kPa.

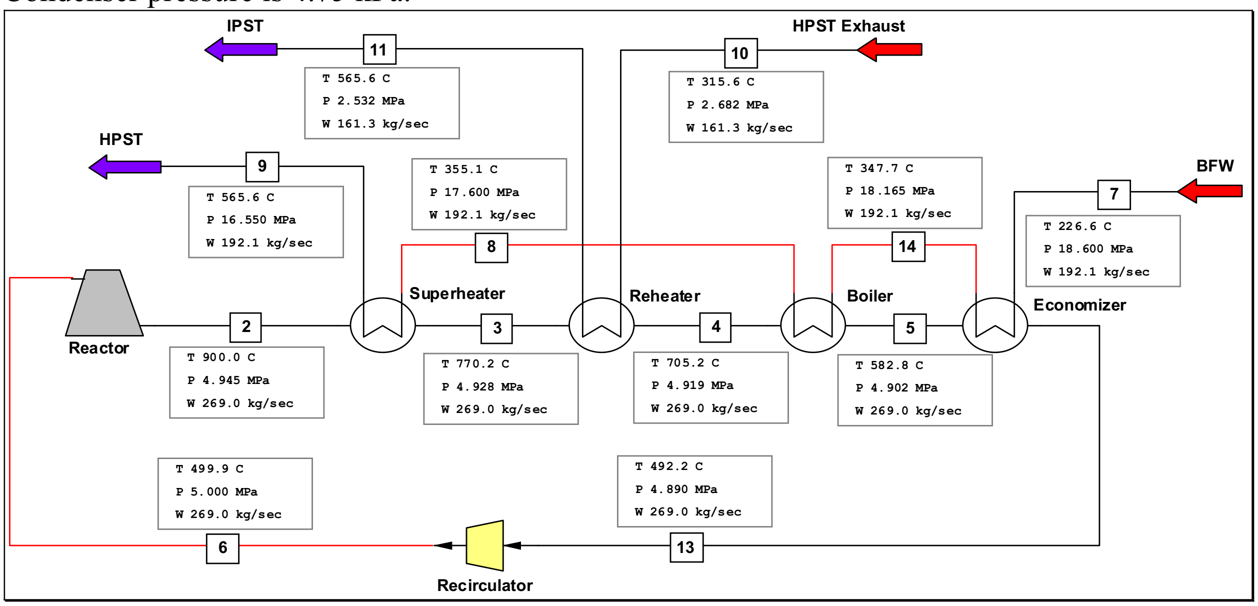


Figure 14. Subcritical Steam Rankine Cycle.

2.6.2 Power Balance

The power balance for the subcritical steam-Rankine system is shown in **Table 12**. This cycle produces a net plant power output of 241.7 MWe from 565 MWt of reactor heat for a net plant efficiency of 42.8%.

Table 12. Subcritical Steam-Rankine Power Balance.

	Power (MW)		Efficiency (%)
Reactor	565.0	Carnot Efficiency	63.6%
Heat Losses	-5.65	Gross Cycle Efficiency	46.6%
Circulator Power	10.97	% of Carnot	73.3%
Net to Cycle	**570.3**	**Net Cycle Efficiency**	**42.8%**
Steam Turbine	265.8		
Gross Cycle Power	**265.8**		
Generator Losses	-5.3		
BOP Losses	-2.7		
Circulator Power	-11.0		
Feed & Condensate	-5.1		
Net Cycle Power	**241.7**		

2.6.3 Component Sizes and Costs

Componet sizes and costs for the subcritical steam-Rankine cycle are shown in **Table 13**. Here, the cost drivers are the condenser and the steam turbine. Steam generating equipment size is low due to the large approach temperature of over 300 °C between the primary loop and the steam generator. A commercial plant using a subcritical steam-Rankine PCS for electricity production would most likely use lower reactor temperatures.

Table 13. Subcritical Steam-Rankine Cycle Component Sizes and Costs.

Component	Heat Transfer Area (m^2)	Size (m^3)	Cost (Rel.)	Floor Space (m^2)
Heat Exchangers	**17,957**	**285.2**	**28.7**	**62.6**
Superheater	760	12	1.2	3.6
Reheater	494	8	1.2	3.6
Boiler	442	7	0.7	3.6
Econimizer	295	5	0.5	3.6
HP Feedwater Heaters	2,439	39	3.8	12.5
LP Feedwater Heaters	731	12	1.2	6.9
Condenser	12,796	203	20.2	28.7
	Stages	**Size (m^3)**	**Cost (Rel.)**	**Floor Space (m^2)**
Turbomachinery		**676**	**25.3**	**107.3**
Steam Turbine		676.0	24.5	107.3
Feed & Condensate			0.8	
Building			**1.4**	
Total		**961**	**55.4**	**169.9**

2.6.4 Technology Readiness

The technology readiness of the subcritical steam-Rankine cycle is rated as level 9. This cycle is very mature and is seeing much service in commercial applications.

2.6.5 Issues

Issues associated with the subcritical steam-Rankine cycle are corrosion and water chemistry control. The potential of water ingress from a steam leak into the primary system should be addressed during system design.

2.7 Supercritical Steam Rankine cycle

2.7.1 Description

The supercritical steam-Rankine cycle is shown in **Figure 15**. It is similar to the subcritical steam-Rankine cycle but the main steam pressure is greater than the critical pressure, a second reheat is included to improve thermodynamic efficiency, and the main and reheat steam temperatures have been increased to 602 °C.

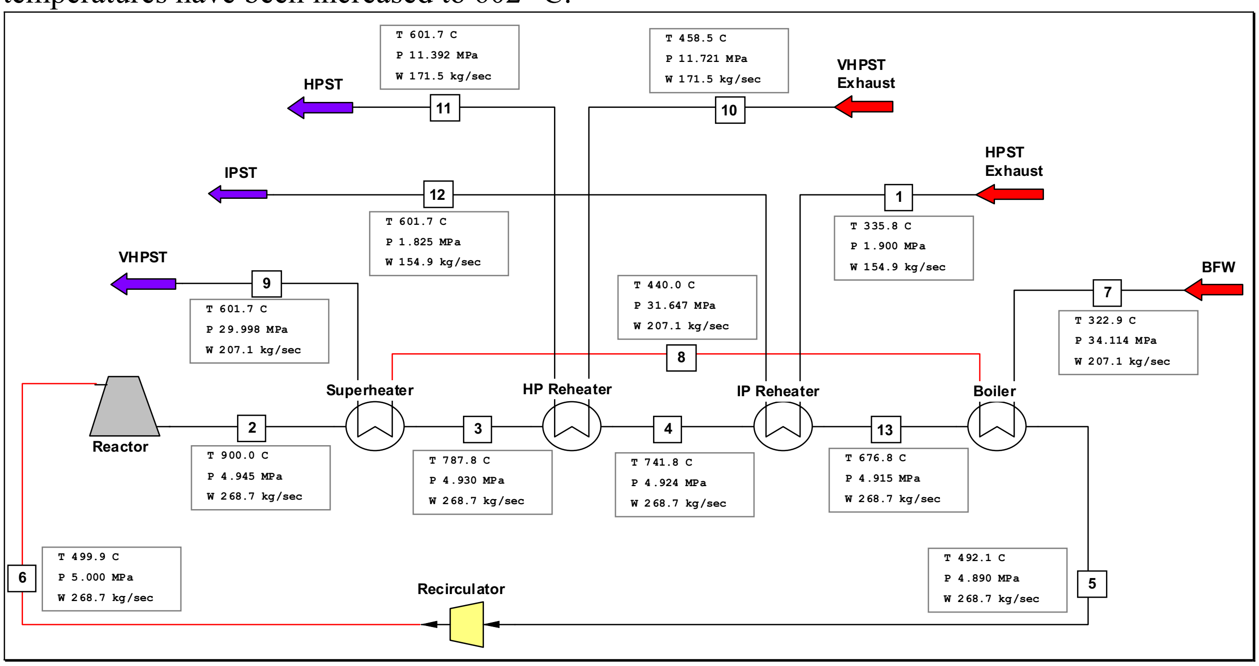


Figure 15. Supercritical Steam Rankine Cycle.

2.7.2 Power Balance

The power balance for the supercritical steam-Rankine cycle is shown in **Table 14**. This cycle provides a net plant power output of 265 MWe from 565 MWt of reactor power for a net plant efficiency of 46.9%.

Table 14. Supercritical Steam-Rankine Cycle Power Balance.

	Power (MW)		Efficiency (%)
Reactor	565.0	Carnot Efficiency	65.1%
Heat Losses	-5.65	Gross Cycle Efficiency	51.8%
Circulator Power	10.97	% of Carnot	79.5%
Net to Cycle	**570.3**	**Net Cycle Efficiency**	**46.9%**
Steam Turbine	295.2		
Gross Cycle Power	**295.2**		
Generator Losses	-5.9		
BOP Losses	-3.0		
Circulator Power	-11.0		
Feed & Condensate	-10.6		
Net Cycle Power	**264.8**		

2.7.3 Component Sizes and Costs

Component sizes and costs for the supercritical steam-Rankine cycle are given in **Table 15**. Cost drivers are the condenser and the steam turbine. The condenser is smaller and less expensive than the condenser for the subcritical steam-Rankine cycle because of the higher efficiency of this cycle, but the steam turbine is larger and more complex due to the higher steam pressures, temperatures, and the added reheat stage.

Table 15. Supercritical Steam-Rankine Component Sizes and Costs.

Component	Heat Transfer Area (m^2)	Size (m^3)	Cost (Rel.)	Floor Space (m^2)
Heat Exchangers	**18,704**	**297.4**	**29.8**	**66.6**
Superheater	760	12	1.2	3.6
HP Reheater	435	7	1.2	3.6
LP Reheater	627	10	1.2	3.6
Boiler	972	16	1.2	3.6
HP Feedwater Heaters	2,386	38	3.8	12.4
LP Feedwater Heaters	2,792	44	4.4	13.4
Condenser	10,733	170	16.9	26.3
	Stages	**Size (m^3)**	**Cost (Rel.)**	**Floor Space (m^2)**
Turbomachinery		**710**	**26.8**	**110.9**
Steam Turbine		710.0	26.0	110.9
Feed & Condensate			0.8	
Building			**1.5**	
Total			**58.1**	

2.7.4 Technology Readiness

The technology readiness for the supercritical steam-Rankine cycle is rated as a level 8. Although there are several fossil-fuel fired supercritical steam-Rankine systems in service, many of these are older, operate at lower steam temperatures, and employ only one reheat. There are two known systems in operation that contain two reheats. These are both located in Denmark; these are Skærbæk unit 3 and Nordjyllandsværket unit 3. There may be others, but further investigation is needed to ascertain this.

2.7.5 Issues

Issues associated with the supercritical steam-Rankine cycle are similar to the subcritical steam-Rankine cycle. These are: corrosion, water chemistry control and water ingress into the primary system due to steam leaks. Further, the increased complexity of the added reheat, the addition of a very-high-pressure-steam-turbine (VHPST) stage, and the higher operating pressure need to be weighed when considering this cycle for power conversion.

3.0 Evaluation

3.1 PCS Cycle Comparison

Key parameters of the power conversion system cycles previously discussed are juxtaposed in **Table 16**. These parameters include: cycle hot let and cold leg temperatures, heat exchanger and turbomachinery sizes, footprint and costs, operation and maintenance assessment, and system maturity (i.e. TRL). The cascaded supercritical CO_2 cycle provides the most power and the greatest thermodynamic efficiency followed closely by the CCGT, the direct Brayton cycle, the supercritical CO_2 cycle, and the supercritical steam-Rankine cycle, the indirect Brayton cycle and the subcritical steam cycle provide the lowest power output and thermodynamic efficiency.

The trend in heat exchanger sizes is that the Brayton and supercritical CO_2 cycles require the greatest amount of heat transfer surface area. This primarily due to the high degree of recouperation required for these cycles to operate efficiently. The direct Brayton cycle area required is lower because an IHX is not required. The CCGT heat transfer surface area requirement is driven mostly by the IHX and the condenser in the steam bottoming cycle. Both the subcritical and the supercritical steam-Rankine cycles have the lowest heat transfer surface area requirements. This is due to the large driving temperature differential between the primary at 900 °C and the steam at 565 and 602 °C, respectively. Most of the required heat transfer area for these two cycles occurs in the condenser and the feedwater heaters.

The trend in turbomachinery sizes is that the supercritical CO_2 cycle and the cascaded supercritical CO_2 cycle turbomachinery is very compact. This due to the nature of the supercritical CO_2 cycle operating near the critical point where the fluid is very dense compared to an ideal gas. The Brayton cycle turbomachinery size is moderate, and the steam turbines in the CCGT bottoming cycle and the two steam-Rankine cycles are the largest due to the large expansion ratios. Further, the CCGT also has the gas-turbine size included.

Capital cost trends are quite interesting. The supercritical CO_2 cycles tend to be the lowest due to their compact turbomachinery; costs for these cycles are driven by heat transfer equipment. The costs of the Brayton cycles is large and is driven by both turbomachinery and heat transfer equipment. The costs of the steam-Rankine cycles are less than the Brayton cycles and are driven by the costs of the steam turbine and the condenser. The costs of the CCGT are the greatest due to the costs of a large IHX, a gas-turbine, plus a steam turbine and condenser.

Operation and maintenance has not been addressed until now. A simple rating system of +, 0, - was used; + being the highest and - the lowest. Because the direct Brayton cycle is coupled directly to the core, where the potential for radioactive contamination to spread through the PCS, and get into inaccessible areas such as bearings, seals, narrow cooling passages, is high, which may require special procedures and equipment to perform maintenance, this system was rated as -. The CCGT and both steam-Rankine cycles were rated as 0 for their corrosion and water chemistry control issues. The indirect Brayton and both the supercritical CO_2 cycles were rated as + because they do not have the radioactive contamination issue and corrosion is judged to be less significant.

For cycle maturity, the steam-Rankine cycles are the most mature; the subcritical steam-Rankine cycle was rated at level 9, and the supercritical steam-Rankine was rated at level 8. The CCGT was rated at a moderate maturity level of 6, while the Brayton cycles, rated at level 4, are somewhat less mature. The supercritical CO_2 cycles are the least mature and were rated at level 3.

Table 16. Comparison of Power Conversion System Cycles.

Parameter	Cycle						
	Direct Brayton	Indirect Brayton	Supercritical CO2	Cascaded Supercritical CO2	CCGT	Subcritical Steam	Supercritical Steam
Hot Temperature (°C)	900	850	642	850	850	565.6	601.7
Cold Temperature (°C)	25	25	32	32	32	32	32
Net Cycle Power (MWe)	268.9	251.6	267.8	281.6	270.9	241.7	264.8
Net Cycle Efficiency (%)	47.6%	44.5%	47.4%	49.8%	47.6%	42.8%	46.9%
Sizes							
Required Heat Transfer Area (m^2)	54,921	63,988	60,976	68,073	49,336	17,957	18,704
Heat Exchangers (m^3)	80	220	144	184	796	285	297
Turbomachinery (m^3)	564	580	5	7	916	676	710
Total Volume (m^3)	645	801	149	191	1,712	961	1,007
Heat Exchangers Floor Space (m^2)	143	160	137	143	108	63	67
Turbomachinery Floor Space (m^2)	131	134	4	9	170	107	111
Total Required Floor Space (m^2)	273	294	141	152	277	170	177
Costs							
Heat Exchanger Costs (Rel.)	21.4	27.1	30.7	29.6	45.9	28.7	29.8
Turbomachinery Costs (Rel.)	57.3	58.9	0.5	0.7	51.7	25.3	26.8
Building Costs (Rel.)	2.3	2.5	1.2	1.3	2.4	1.4	1.5
Total Costs (Rel.)	81.0	88.5	32.4	31.7	100.0	55.4	58.1
O&M	-	+	+	+	0	0	0
TRL (Development Costs)	4	4	3	3	6	9	8

3.2 PCS cycle evaluation

In order to evaluate the various PCS cycles, a simple grading system was used to compare the raw values of key parameters presented in **Table 16** in a concise but clear way. A **+**, **0**, **-**, **-** grading system was used with **+** being very good or favorable, **0** being acceptable, **-** being marginal or a moderate concern, and **-** meaning that this is a major concern or disadvantage. The parameters used for the evaluation were judged to be the most salient for evaluating a PCS option. These are cycle power/efficiency, total costs, operation and maintenance, and system maturity. This evaluation is presented in **Table 17**.

For cycle power, the cascaded supercritical CO_2 cycle delivered the most power and was given the highest ranking. The CCGT, supercritical CO_2 cycle, direct Brayton cycle, and supercritical steam-Rankine cycle gave comparable, but lower than the cascaded supercritical CO_2 cycle, powers and were given a good rating. The other two cycles, the indirect Brayton and the subcritical steam-Rankine, produced the least amount of power and were given a moderate rating.

For total costs, the two supercritical CO_2 cycles had the lowest costs and the highest rating. The two Brayton cycles and the two steam-Rankine cycles were rated as good, and the CCGT was rated as moderate.

For operation and maintenance, the indirect Brayton and the two supercritical CO_2 cycles were given the highest rating. The CCGT and the two steam-Rankine cycles were rated lower because of corrosion and water chemistry control issues, and the direct Brayton cycle was given a moderate rating because of the radioactive contamination issue.

For system maturity, the two steam-Rankine cycles were rated the highest. The CCGT was rated as acceptable due to its moderate maturity, and the other cycles were rated as marginal due to their low maturity.

In addition to the above categories, each PCS cycle was assessed whether it would be sufficiently mature to meet a start-up date of 2018. The two steam-Rankine cycles were rated as favorable due to their maturity. The two Brayton cycles and the two supercritical CO_2 cycles were rated as marginal due to their lack of sufficient maturity to provide confidence that they would be available in time. The CCGT was rated as acceptable even though there are risk issues which need to be addressed such as closed loop operation, high compressor inlet temperature, and materials issues.

Table 17. Power Conversion System Evaluation.

Parameter	Cycle						
	Direct Brayton	Indirect Brayton	Supercritical CO2	Cascaded Supercritical CO2	CCGT	Subcritical Steam	Supercritical Steam
Cycle Power / Efficiency	0	–	0	+	0	–	0
Total Costs (Rel.)	0	0	+	+	–	0	0
O&M	–	+	+	+	0	0	0
TRL (Development Costs)	–	–	–	–	0	+	+
Schedule	–	–	–	–	0	+	+

3.3 Commercial Plant Applicability

In addition to the PCS evaluation categories discussed above, the various PCS cycles were assessed for their impact upon commercial plant considerations. Specifically, three categories were examined. These are: reactor impact which examines how reactor materials, schedule, fuel cycle length, and power level are affected by selection of a given PCS cycle; process heat flexibility which examines the adaptability of the PCS to provide process heat to a broad, diverse, and most likely near term markets for medium and high temperature steam (i.e < 550 °C); and very high-temperature process heat (VHTPH) which examines the ability to service the thermo-chemical hydrogen production processes being considered.

The results of this evaluation are shown in **Table 18**. The supercritical CO_2 cycle and the two steam-Rankine cycles had the most positive impact on the reactor. This is because these cycles operate at lower turbine inlet temperatures implying that the reactor temperatures can be lowered, thus permitting higher reactor power, increased fuel cycle length, and a positive impact on reactor material selection and schedule. The CCGT, direct Brayton cycle, and the cascaded supercritical cycle had a smaller positive impact on the reactor because these cycles could be operated at 850 °C reactor outlet temperature. The indirect Brayton cycle did not have a favorable impact on the reactor because reactor outlet temperature could not feasibly be brought lower than 900 °C without having a major impact on performance, and the high reactor inlet temperature of 550 °C significantly affects vessel cooling. For process heat flexibility, the two steam-Rankine cycles and the CCGT were given the highest rating, because the ability to extract steam at the desired conditions from a steam cycle is well established. The two Brayton cycles were given negative ratings because the ability to provide steam for process heat is impractical. The two supercritical CO_2 cycles have not been rated because the cycle has not been extensively studied to determine the practicality of providing process heat. For applicability of providing VHTPH, the indirect Brayton and the CCGT were given the highest ratings because of the high reactor outlet temperature coupled with the use of an IHX which can incorporate a VHT loop for thermo-chemical processes. The two steam-Rankine cycles were given poor ratings because electric supplemental heating would most likely be needed to achieve the desired high temperatures. The direct Brayton cycle was given a negative rating due to the difficulty of incorporating a parallel VHT loop. The two supercritical CO_2 cycles were not rated for the same reasons given for not rating them for process heat flexibility.

Table 18. Commercial Plant Applicability Evaluation.

Parameter	Cycle						
	Direct Brayton	Indirect Brayton	Supercritical CO2	Cascaded Supercritical CO2	CCGT	Subcritical Steam	Supercritical Steam
Reactor Impact	0	–	+	0	0	+	+
Process Heat (PH) Flexability	–	–	?	?	+	+	+
VHTPH	–	+	?	?	+	–	–

4.0 Conclusions & Recommendations

4.1 Conclusions

The best PCS cycle for the NGNP is dependent upon the worth of cycle efficiency and the importance of cycle maturity. The supercritical CO_2 cycles are very promising for longer term applications. The need for development is a disadvantage for near term applications. The ability to arrange them in a cascaded configuration for large ΔT applications is a plus. Steam-Rankine cycles are the most mature, but the cost of steam turbines and supporting equipment reduces their attractiveness. The supercritical steam cycle with two reheats is the best steam cycle option. The Brayton cycles are marginal in cost and performance. Operation and maintenance difficulties from radioactive contamination of the PCS are a negative for the direct Brayton cycle. On the other hand, the loss of efficiency as a result of the temperature drop across the IHX reduces the attractiveness of the indirect Brayton cycle. Further, because of the relative unattractiveness of the Brayton cycles when compared to the supercritical CO_2 cycles brings further pursuit of Brayton cycle development into question. The CCGT performance is good but the costs, added complexity and lower maturity when compared with steam-Rankine cycles reduces its attractiveness. The potential for long term economic advantage from small efficiency differences when compared to the supercritical steam-Rankine cycle or the indirect Brayton cycle may swing the advantage to the CCGT.

4.2 Recommendations

Steam-Rankine cycle (possibly supercritical) is the best fit for near term applications. It provides high efficiency electricity production and can readily service near term process heat markets. The promising benefits of the supercritical CO_2 cycle warrant continuing development for long term electricity production applications. Further, a more detailed evaluation of equipment costs and size would be beneficial for confirmation of these recommendations.

4.3 References

Hamilton Sundstrand
A United Technologies Company

[1] M. J. Driscoll, *Interim Topical Report Supercritical CO2 Plant Cost Assessment*, Report No. MIT-GFR-019, MIT (2004).

[2] J. C. Mankins, *Technology Readiness Levels; A White Paper*, NASA Office of Space Access and Technology, NASA (1995).

[3] J. W. Bilbro and R. L. Sackheim, *Managing a Tchnology Development Program*, NASA Marshal Space Flight Center, NASA (2002).

[4] H. C. No, J. H. Kim and H. M. Kim, A Review of Helium Gas Turbine Technology for High-Temperature Gas-Cooled Reactors, J. Nucl. Eng. And Tech., 39, 21 (2007).

PRECONCEPTUAL DESIGN STUDIES REPORT

APPENDIX B4

(Previously Issued as 12-9045705-001)

NGNP with Hydrogen Production

Primary and Secondary Cycle Concept Study

April 2007

BEA Contract No. 000 60209

AREVA NP Inc.

Record of Revisions

Revision	Date	Pages/Sections Changed	Brief Description
000	3/14/2007	All	Initial Issue
001	4/18/2007	3.4.3, 3.4.4, 5.0	Add Figure 3-16 to provide schematic of four loop configuration. Add discussion of Option B and add to discussion of Option C in Section 3.4.4. Updated citations for References 1 and 4.

Table of Contents

Table of Contents
(continued)

Page

List of Tables

Page

List of Figures

1.0 INTRODUCTION

The Primary and Secondary Cycle Concept Study is one of four Next Generation Nuclear Plant (NGNP) preconceptual design special studies performed by the AREVA NGNP team:

- Reactor Type Comparison Study
- Prototype Power Level Study
- Power Conversion System Study
- Primary and Secondary Cycle Concept Study

1.1 Background

The NGNP project is intended to demonstrate the applicability of the high temperature reactor (HTR) to high efficiency electricity production and to nuclear hydrogen production. The Idaho National Laboratory (INL) is facilitating the NGNP project for the U.S. Department of Energy (DOE). A goal of the project is to perform the concept development, technology development, and prototype demonstration in cooperation with industry to lead to the future commercialization of this technology.

The NGNP project is currently in the Preconceptual Design phase. INL and laboratory operator Battelle Energy Alliance (BEA) contracted with the AREVA NGNP team and other design teams to perform preconceptual design engineering studies. The resulting studies and design recommendations will be used to determine and refine the NGNP technical and functional requirements that will form the basis for future Conceptual Design phase work and they will be used to focus and prioritize NGNP R&D activities.

The AREVA NGNP team agreed scope of work includes four special studies, as noted above, and development of an NGNP preconceptual design concept adapted from AREVA's ANTARES commercial HTR concept.

Each of the studies being performed will provide information to INL to be used in the selection of one or more design concepts to be further evaluated and developed during the Conceptual Design phase.

1.2 Primary and Secondary Cycle Concept Study Objectives

The main objective of the Primary and Secondary Cycle Concept Study is to establish the basic NGNP operating parameters. The Prototype Power Level Study will establish the design reactor power level for the NGNP and the power that must delivered to the Hydrogen production process. The primary and secondary study determines the remaining key parameters. Specifically, the parameters to be established are the primary and secondary temperatures and the system pressure.

In addition, the basic configuration of the NGNP nuclear heat supply system (NHS) will be established. Two distinct aspects of the system configuration are of interest. The first deals with whether the heat loads from the hydrogen production plant and the electricity generating system are placed in series with one another or in parallel. The second deals with the number of primary loops to be used to support these heat loads.

In establishing these parameters, the study will confirm or modify key parameters in the NGNP Preconceptual Design Baseline document [1], and it will enhance the basis for the Design Baseline.

A secondary purpose of all the special studies is to provide a basis for the selection of the NGNP concept to be developed during the Conceptual Design phase.

1.3 Scope and Assumptions

The purpose of this study is to recommend operating parameters for the NGNP NHS and associated systems. The detailed design of those systems and the implementation of the selected parameters is not part of the study scope. These parameters will be used in separate design activities where plant systems and components will be developed as defined in AREVA's NGNP Preconceptual Design Work Plan [2] as approved by INL.

In addition, while this study provides an enhanced basis for the NGNP Design Baseline [1], it supports only a limited number of design baseline parameters. Only the primary and secondary hot and cold leg temperatures and the system pressures are developed within the study. As part of the Preconceptual Design Study phase, other lower level parameters will be established based on direct design adaptation from the ANTARES reference HTR concept. Other lower level parameters will be the subject of more detailed evaluation during the Conceptual Design phase.

Several assumptions govern this study and the application of its results.

Implicit in this study is the fact that the AREVA NGNP team's design activity is limited to an adaptation of the ANTARES which is AREVA's commercial HTR design concept. Therefore, for the Primary and Secondary Cycle Concept Study, it is predetermined that the NGNP concept is based on an indirect cycle prismatic block design coupled to a combined cycle gas turbine (CCGT) generating system. Heat from the HTR is supplied to the Power Conversion System (PCS) through an intermediate heat exchanger (IHX) for electricity production. The heat supplied to the PCS first drives a Brayton topping cycle and then a Rankine bottoming cycle. Similar to ANTARES, the NGNP Brayton topping cycle is assumed to use a nitrogen-based fluid to allow the use of air-breathing gas turbine technology. Thus, when the characteristics of the PCS must be considered in evaluating the issues addressed by this study, those of the ANTARES concept are assumed.

Since the design of the Hydrogen Production Process Plant and the High Temperature Heat Transport Loop which carries heat between the NHS and the hydrogen plant are outside of the AREVA NGNP team's assigned scope, assumptions must be made about these systems for this study.

The Hydrogen Production Process Plant is assumed to demonstrate two high temperature hydrogen production processes consistent with the goals of the NGNP program high level technical requirements [3] and the AREVA NGNP Preconceptual Design System Requirements Manual [4]. Based on these, the NGNP is required to demonstrate the direct Sulfur-Iodine thermochemical process and the high temperature electrolysis (HTE) process. These processes are assumed in the determination of the required NHS operating parameters and configuration.

The High Temperature Heat Transport Loop is assumed to be a closed helium system. A single intermediate heat transport loop between nuclear heat source and hydrogen process is assumed. This implies two intermediate heat exchangers. One is the IHX between the primary and heat transport loop. This IHX is taken as part of the NHS. The second intermediate heat exchanger is the process heat exchanger between the heat transport loop and the chemical process reactants. Thus, if the reactor outlet temperature is "r", the NHS IHX approach temperature is "d1", and the process heat exchanger approach temperature is "d2", the temperature in the process itself would be $p = r - d1 - d2$.

The evaluations and results obtained within this study are obviously tentative in nature. As preconceptual design studies, they are limited due to the incomplete and speculative nature of all the information on which they are based. Therefore, all of the results will have to be reconfirmed during the Conceptual Design phase. This includes any detailed analyses or evaluations performed as a part of this study. None of these calculations represent formal design calculations.

1.4 Document Structure

Following this introduction, the main body of this report is divided into three main sections.

Section 2.0 identifies each of the questions to be answered. The current baseline is recalled, and the potential alternatives are identified for each decision. The decision process to be used to answer all of the identified questions is outlined.

Section 3.0 provides the evaluation of each question including the tradeoffs between the key considerations affecting each of the questions. Each question is addressed in a specific subsection:

Section 3.1 Reactor Outlet Temperature

Section 3.2 Reactor Inlet Temperature

Section 3.3 Configuration – Parallel or Series

Section 3.4 Configuration – Number of Loops

Section 3.5 Secondary Temperatures

Section 3.6 System Pressures

Section 3.7 provides a brief review of the decision results to make sure that they are compatible.

Section 4.0 summarizes the overall conclusions and results.

References are listed in Section 5.0.

Appendix A discusses high temperature heat transport options. This was not part of the study scope, but it is relevant to many of the issues discussed.

2.0 PRIMARY AND SECONDARY CYCLE CONCEPT STUDY DEFINITION

The problem statement to be addressed by the study is defined in the following subsections. This consists of the questions that must be answered, the considerations and criteria that must be satisfied in selecting the best solution, and the approach to the study that is to be employed. The range of alternatives is also discussed.

2.1 Questions Addressed by Primary and Secondary Concept Special Study

The questions that must be resolved within this study are divided into five main issues:

Issue 1 - What is the recommended reactor Tout?

Issue 2 - What is the recommended reactor Tin?

Issue 3 - What should the system configuration be?

Issue 3.1 - Should heat supply to the hydrogen process be parallel or in series with power generating system?

Issue 3.2 - How many loops should the system have?

Issue 4 - What is the secondary side T_{hot} and T_{cold}?

Issue 5 - What are the primary and secondary system pressures?

Considering that there are two distinct aspects of the system configuration issue, this results in six main questions that must be addressed. Each of these questions is evaluated individually in Section 3.0.

2.2 Overall Considerations

Having established the questions to be evaluated, the next step is to define the criteria used to evaluate questions. There are three main requirements which drive this special study:

- Demonstrate scalability to commercial electricity and hydrogen production
- Demonstrate advanced hydrogen production processes
  - Sulfur-Iodine
  - High Temperature Electrolysis
- Initial NGNP operation by 2018

When selecting the NGNP operating parameters and system configuration, it is important to consider whether the resulting NGNP concept will satisfy these requirements, since the selected parameters will implicitly define the NGNP concept characteristics.

The first driving requirement is intended to ensure that the NGNP will provide a near term step to commercial electricity and hydrogen production using an HTR. The NGNP must demonstrate technologies and approaches applicable to a commercial plant, and the demonstration must be such that the NGNP concepts can be scaled directly to a commercial plant. The second driving requirement is intended to ensure that the NGNP will demonstrate advanced hydrogen production processes in order to maximize the performance of the resulting commercial system. The Sulfur-Iodine (SI) and High Temperature Electrolysis (HTE) processes are widely considered to be the most likely candidate processes. The third main driving requirement is intended to ensure that the NGNP demonstration will provide the required technology for commercial deployment on a timescale compatible with end user considerations and the need for alternate energy sources.

Together these driving requirements dictate the performance requirements that will be imposed on the NGNP and the level of feasibility and technical maturity required of potential NGNP systems.

Table 2-1 provides a more complete list of all the considerations that affect the selection of the NGNP system parameters and configuration. They are divided into the areas of Feasibility and Risk, Safety, Performance, Flexibility, Cost, and Schedule. All of these considerations are important in the design of the NGNP and in addressing the full set of NGNP design requirements, but not all are distinguishing factors in determining the NGNP parameters.

The feasibility and technical risk considerations are very important in the evaluations performed in this study, particularly in light of the planned 2018 NGNP startup. The study examines the feasibility of the NHS and to a lesser extent the PCS and hydrogen process plant, since the selected operating parameters have a strong impact on these feasibilities.

Safety is undoubtedly an important parameter in the development of the NGNP concept. However, it is assumed that the plant will be designed to the same safety guidelines regardless of the system operating parameters selected. Therefore, it is not a direct discriminator in selecting the operating parameters and heat delivery system configuration. Safety considerations are reflected indirectly in the decisions to the extent that other parameters are constrained in order to maintain consistent safety margins. For example, if a design option requires the power level to be reduced in order to maintain consistent safety margin, then that is reflected as a tradeoff against plant performance (i.e., power level), since that is what is impacted. The safety margin is kept constant.

Several aspects of the NGNP plant performance are important. Foremost in selecting the operating parameters is the plant efficiency, since both the electricity generating efficiency and the hydrogen production efficiency are expected to be affected by temperature. In addition, the power level that the plant can achieve will be affected by the operating parameters. The operating parameters and system configuration also influence the other performance characteristics including maneuverability, maintainability, etc.

Three distinct aspects of NGNP flexibility are important. The first is basic operational flexibility which deals with the plants ability to operate in an all electric mode, an all hydrogen production mode, a cogeneration mode, and the ability to switch back and forth between these modes. The second deals with the adaptability of the plant to demonstrate other technologies including alternate hydrogen production processes and alternate power conversion system concepts. This flexibility is supported both by a system configuration that is adaptable to alternate equipment and a plant that can accommodate a range of operating parameters. The third aspect of flexibility deals with market flexibility. This consideration values NGNP concepts which have the flexibility to economically support a greater number of applications in the marketplace.

Cost is an important factor in determining the commercial applicability of an NGNP concept, including the development cost, the plant capital cost, and the operating and maintenance costs.

Schedule considerations have direct bearing on the ability to meet the requirement for NGNP startup by 2018. This includes the design schedule, the fabrication and construction of the plant, and the time required for technology development. The design selections made within the special study may impact each of these schedules depending on the resulting design complexity, the need for long-lead component procurement, and the time required to develop supporting technologies.

Given this broad set of general considerations, many of the decisions in the Primary and Secondary Cycle Concept Study come down to the competition between two main factors: hydrogen process requirements on one hand, and NHS design impact on the other. A major factor for the hydrogen process is the range over which heat must be supplied (both the peak temperature and the energy use temperature spectrum). For the NHS, the main factor is the decreasing feasibility that comes at higher heat delivery temperatures and the resulting increase in design difficulty.

The specific considerations that are dominant for each issue to be evaluated are emphasized at the beginning of each evaluation in Section 3.0.

Table 2-1: Overall Considerations for Primary and Secondary Cycle Concept Study

- **Feasibility and risk**
    - **NHS feasibility**
    - **PCS feasibility**
    - **Hydrogen process impact**
- **Safety**
    - **Accident scenarios**
    - **Accident frequency**
    - **Accident severity**
    - **Safety margins**
    - **Inherent safety**
- **Performance**
    - **Plant efficiency**
    - **Power level**
    - **Maneuverability**
    - **Availability**
    - **Maintainability**
- **Flexibility**
    - **Operational flexibility**
    - **Demonstration flexibility (alternate hydrogen, electricity, or other)**
    - **Market flexibility of commercial concept**
- **Cost**
    - **Development**
    - **Capital**
    - **O&M**
- **Schedule**
    - **Design**
    - **Procurement**
    - **R&D**

2.3 Decision Process

As indicated in Section 2.1, there are six major decisions to be made within the Primary and Secondary Cycle Concept Study. These decisions are not independent, since each decision is influenced by the results of one or more of the other decisions. Table 2-2 shows the general interdependency of each of the decisions. The decision process used to evaluate each of the six questions must take this interdependence into account. The basic approach is to answer the questions in sequence. The intent is to first answer the study questions most narrowly constrained by the study considerations and least influenced by the other study questions. Then the questions most strongly dependent on the preceding questions will be evaluated.

Table 2-2: Multiple decisions and interdependencies

	Other Questions						Main Driver		
For each question below, table indicates major dependency on other questions shown at right	Reactor Tout	Reactor Tin	Parallel or Series	No. Loops	Secondary Temps	Pressure	NHS Feasibility	Performance	Cost
Reactor Tout?					x		X	X	
Reactor Tin?	X						X		
Parallel or Series?	X	X					X	X	
No. Loops?	X	X	X			x	X		X
Secondary Temps?	X						X	X	X
System Pressure?		X		X			X		X

The reactor outlet temperature is perhaps the most independent of the questions. It is driven by the NHS feasibility and the hydrogen process performance characteristics. It has a strong impact on the remaining decisions.

The reactor inlet temperature has a significant impact on the NHS feasibility. A key consideration is that it can be used to compensate for the effects of higher reactor outlet temperature, if that has been selected. As will be discussed later, selecting a higher core inlet temperature can reduce peak operating fuel temperatures, although other NHS factors are also affected.

Whether the NHS configuration should supply heat to the hydrogen process and the PCS in series or in parallel is driven by the hydrogen process energy use spectrum. If the energy is needed only over a small fraction of the range from reactor outlet temperature down to reactor inlet temperature, then a series option may be preferred. On the other hand, if the energy is needed over the full core temperature range, then a parallel configuration may be best.

The decision regarding how many loops the plant should have is driven by the feasibility of the NHS components at the selected temperatures. It is also obviously affected by the decision whether the system should be in series or parallel, since the parallel configuration is inherently more adaptable to multi-loop arrangements.

The secondary side hot and cold leg temperatures are driven by NHS feasibility and hydrogen process performance. The decision is also influenced by plant cost. In essence, the question is how high of an effectiveness should be specified for the IHX. The reactor outlet temperature decision is somewhat dependent on this decision, since a lower effectiveness IHX can be compensated for by a higher reactor outlet temperature. However, the reactor outlet temperature decision is given higher priority, since it is more severely constrained by the NHS feasibility.

The primary and secondary system pressures are driven by NHS component feasibility and cost. The main tradeoff is between circulator pumping power requirements and vessel system cost. The decision is strongly impacted by the number of loops, since multiple loops significantly reduce the performance requirements for individual circulators.

The decisions to be made within the Primary and Secondary Cycle Concept Study are also closely connected with the decisions in other special studies. For example, the system operating temperatures directly affect allowable power level that can be achieved while maintaining passive cooling capability. The power level affects system design constraints, including operating fuel temperature, component sizing and feasibility, and component service conditions and material requirements, all of which influence the selection of the operating parameters.

There is a similar connection with the PCS Study. The PCS design and performance depends on the system temperature. And for indirect cycle systems the IHX design must accommodate the reactor outlet temperature. (Of course, the turbine must accommodate this temperature for direct Brayton cycles.)

Finally, the design adaptation of the Nuclear Island and the PCS are certainly impacted by the system parameters.

Considering all these factors the decision sequence is as follows:

1. Establish reactor outlet temperature

   This is primarily a tradeoff between hydrogen process performance and nuclear heat source (and hydrogen plant) feasibility.

2. Establish reactor inlet temperature

   Given the tentative reactor outlet temperature, the reactor inlet temperature decision is then a tradeoff between core design, vessel and internals materials considerations.

3. Establish plant schematic

   The optimum plant schematic (series or parallel heat loads) is determined by the hydrogen plant energy requirement spectrum and operational flexibility issues. The key question is whether the hydrogen plant needs all of its energy input at the reactor outlet temperature or over a broad range more consistent with the reactor temperature range.

4. Establish secondary temperatures

   The decision of how close the secondary coolant temperatures should be to the primary coolant temperatures is determined by economics. It is a tradeoff between system performance and component (primarily IHX) cost.

5. Establish system pressure

   This is a tradeoff between vessel cost and pumping power.

Once all the decisions have been evaluated, the initial decisions are revisited in light of the subsequent decisions. This is done to ensure that the initial decisions (e.g., reactor outlet temperature) did not unintentionally constrain one of the later decisions to an unanticipated and undesirable outcome.

2.4 Alternatives Considered

The key parameters in the initial AREVA NGNP Preconceptual Design Baseline are summarized in Table 2-3. These parameters provide the starting point for the current study. The study results either confirm or replace these values.

The range of alternatives considered within the study to replace these initial values is noted in Table 2-4. It is noted that the question of how many loops is somewhat more complicated than simply saying one loop or three loops, etc., since the number of circulators and heat exchangers can be different. This will be addressed more specifically in the detailed evaluation.

Table 2-3: Original NGNP Preconceptual Design Baseline

Initial NGNP Design Baseline (Selected parameters) Starting point for Primary and Secondary Cycle Concept Study	
Reactor outlet temperature	900°C
Reactor inlet temperature	500°C
System configuration	H2 and PCS in parallel
Number of loops	4 loops
	3 tubular IHX for PCS
	1 compact IHX for hydrogen plant
Secondary temperatures	450-850°C for PCS (50°C approach)
	475-875°C for hydrogen plant (25°C approach)
System pressure	5.0 MPa

Table 2-4: Range of Options Considered in Study

Reactor outlet temperature	Overall range 850-950°C Emphasis on 875°C, 900°C, and 925°C
Reactor outlet temperature	Overall range 400-600°C Emphasis on 500°C, 525°C, and 550°C
System arrangement	Parallel H2 IHX and PCS Series (H2 IHX ahead of PCS) Single loop vs. multiple loop (X PCS + 1 H2)
Secondary temperatures	IHX approach temperature between 25°C – 50°C
System pressure	Overall range 4.0-8.0 MPa Emphasis on 5.0 MPa, 5.5 MPa, 6.0 MPa, and 6.5 MPa

3.0 DECISION EVALUATION

3.1 Issue 1 – Evaluation of Reactor Outlet Temperature

The first decision to be addressed is the reactor outlet temperature. A summary of the issue is reviewed in the following subsection. The next subsection provides the evaluation of the discriminating considerations. Then the conclusion for this decision is summarized in the final subsection on this decision.

3.1.1 Summary of Reactor Outlet Temperature Issue

Key question:

"What is the recommended reactor T_{out}?"

Range of options:

Overall range 850-950°C with emphasis on 875°C, 900°C, and 925°C

Major considerations:

Hydrogen plant performance

Nuclear heat source feasibility

Other discriminators:

PCS performance

R&D Schedule

NHS and PCS cost

3.1.2 Assessment of Reactor Outlet Temperature Considerations

3.1.2.1 Hydrogen Plant Performance Considerations for Reactor T_{out}

An important NGNP mission is to demonstrate nuclear hydrogen production, and the temperature at which heat is supplied to the hydrogen process is a key factor in determining the success of the process demonstration. The NGNP is expected to demonstrate both the sulfur-iodine (S-I) thermochemical process and the high temperature electrolysis (HTE) process. It is likely that other processes will also eventually be demonstrated in the NGNP facility, but the S-I and HTE processes provide reasonably bounding surrogates for many other processes in terms of the required thermal and electrical loads which might be placed on a commercial nuclear hydrogen facility.

The S-I process puts a greater burden on the high temperature process heat delivery capability of the NGNP, so it will be the main process considered in evaluating the required heat delivery temperature and indirectly the required reactor outlet temperature. The S-I process temperature governs both the process efficiency and the process equipment sizing for the thermochemical hydrogen production process. The process efficiency is essentially the thermodynamic efficiency of the process. It provides a measure of the net energy content of the hydrogen produced for a given thermal energy input to the process. The temperature also affects the reaction yields within the process. While this does not directly affect the process efficiency, it has a significant impact on the size of the piping, pumps, heat exchangers and other equipment used in the process plant. Therefore, this also affects the plant economics.

HTE requires a relatively small amount of thermal energy and a large amount of electrical energy. Therefore, the plant's electricity generating efficiency is very important for HTE. The HTE process does require process temperatures in the neighborhood of approximately 800°C. However, the process performance does not vary strongly above this temperature. So there is little incentive to apply significantly higher temperatures for HTE. This implies a minimum NGNP outlet temperature of 850°C.

As suggested in the previous paragraph, the temperature drop between the reactor and the hydrogen process must be considered in assessing the impact of the hydrogen process on the required reactor outlet temperature. This temperature drop must take into account all temperature drops between the NHS core outlet and the hydrogen process peak chemical reactor temperature. A single intermediate heat transport loop is assumed. Therefore, the resulting temperature drop must include the main IHX between the NHS primary circuit and the intermediate heat transport loop and the process heat exchanger (chemical reactor) which separates the intermediate heat transport loop fluid and the chemical reactants.

In addition, any temperature drop along the intermediate heat transport loop will also have to be accounted for. For a well designed commercial scale intermediate heat transport loop, this temperature drop is expected to be minimal. The actual drop for the NGNP will have to be evaluated considering the smaller heat transport lines and the actual distance to the hydrogen production plant. This evaluation is beyond the scope of this study, but it is reasonable to assume that this temperature drop will be small compared to the heat exchanger temperature drops for a well designed NGNP system.

For this study, an NGNP hydrogen production process temperature 50°C below the reactor outlet temperature is assumed. This is based on a 25°C approach temperature in the IHX and a 25°C approach temperature in the process heat exchanger. This represents reasonable performance based on the anticipated heat exchanger technology. This assumption will be revisited in Section 3.5.

As indicated above, the first temperature consideration for the S-I process is the effect of temperature on the theoretical hydrogen process efficiency and the resulting energy requirement. Application of Knoche and Funk's analysis [5] shows that the ideal efficiency limit for a coupled water-splitting process will be only a mild function of the reactor coolant outlet temperature for operations above 800°C. The return temperature of the coolant, however, has a significant effect on the ideal efficiency limit. This is illustrated in Figure 3-1 below. Knoche and Funk's analysis shows that the relative increase in net thermal efficiency achieved by increasing the reactor outlet temperature from 850°C to 950°C is only about 2%, and decreases as the return temperature is increased.

Efficiencies of real water-splitting processes can be expected to behave in proportion to ideal efficiencies. Evaluations of different detailed hydrogen processes coupled to high temperature reactor heat source have been performed by various investigators. The relative efficiency and temperature regime are shown for some of these in Figure 3-2. The 2002 S-I flowsheet had a suboptimal S-I Section 2 (H_2SO_4 decomposition), an infeasible S-I Section 3 (HI decomposition), and achieved 42% net thermal efficiency (Higher Heating Value (HHV) basis) at an 827°C peak process temperature (875°C reactor temperature). The 2005 S-I flowsheet had an optimized Section 2, a redesigned Section 3 (reactive distillation), and achieved 46% net thermal efficiency (HHV) at a 900°C peak process temperature (950°C reactor temperature). The 2004, 2005, and 2006 Hybrid-Sulfur flowsheets achieved successively higher net thermal efficiencies (46, 48.5, and 52%, HHV basis) at peak process temperatures between 850°C and 900°C due primarily to improved energy recovery. As can be seen in Figure 3-2, increasing the peak process temperature does not have a significant effect on net thermal efficiency. The blue line extrapolates the 2002 S-I flowsheet performance at different temperatures in proportion to its ideal efficiency limit, while the orange line describes 70% of ideal efficiency. Real process efficiencies can be expected to vary with reactor temperature in similar fashion.

The magenta-shaded region between 850°C and 950°C, and between 50 and 60% net thermal efficiency (HHV) represents a reasonable performance target for sulfur-based thermochemical cycles.

The conclusion that can be drawn from this analysis is that as long as the reactor outlet temperature is above 800°C, the effect of higher temperatures on net thermal efficiency will not be very significant.

With regard to capital cost, temperature does affect conversion in the sulfuric acid decomposition reactor. As is typical of endothermic reactions, higher temperatures favor increased conversion. That means higher temperatures will result in smaller process volumes due to a reduction in the amount of unconverted materials that need to be recycled. Smaller process equipment means a reduction in capital cost. The effect of temperature on the conversion yield of the sulfuric acid decomposition reaction is illustrated in Figure 3-3 [6]. The figure also shows the affect of pressure on conversion. In the range of likely operating temperatures, yield benefits significantly from increased temperature, particularly at higher pressures.

Table 3-1 summarizes the evaluation of the temperature requirement for the hydrogen production process. Process efficiency requires a minimum of 800°C (which translates to 850°C at the reactor outlet). However, process yield considerations suggest a higher temperature in order to provide reasonable capital equipment costs for the process recycle loops, etc. Therefore, a process temperature near 850°C is preferred to provide margin. At very high temperatures, the feasibility of process equipment may decrease unacceptably unless ceramic components are assumed. Thus, from the hydrogen process perspective, a peak process temperature in the range of 850-875°C is recommended.

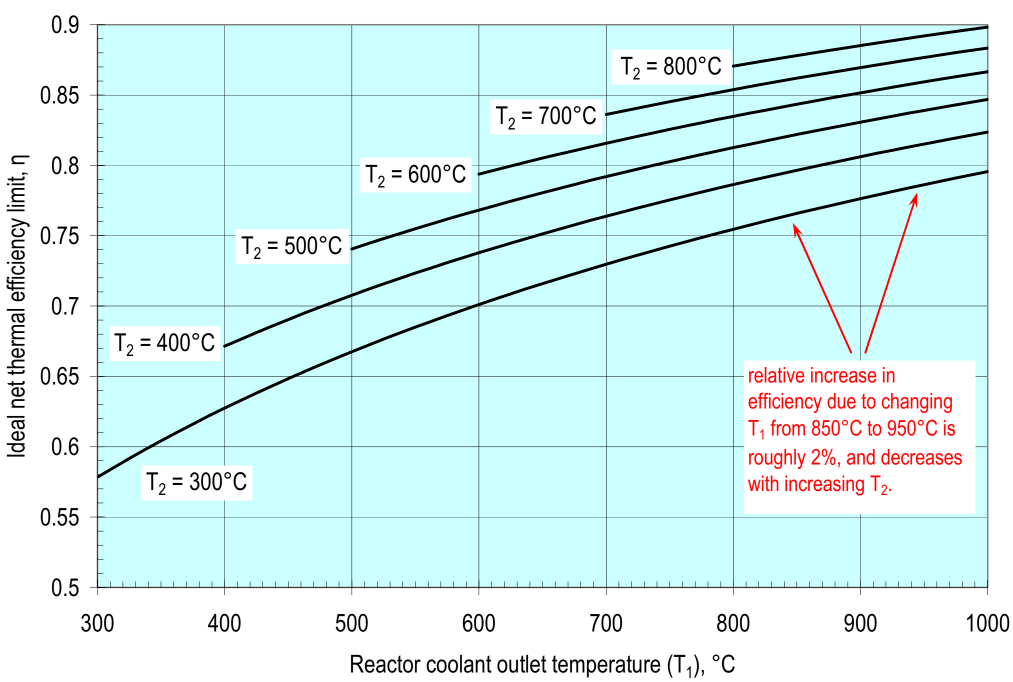


Figure 3-1: Ideal Thermochemical Hydrogen Production Efficiency Temperature Sensitivity

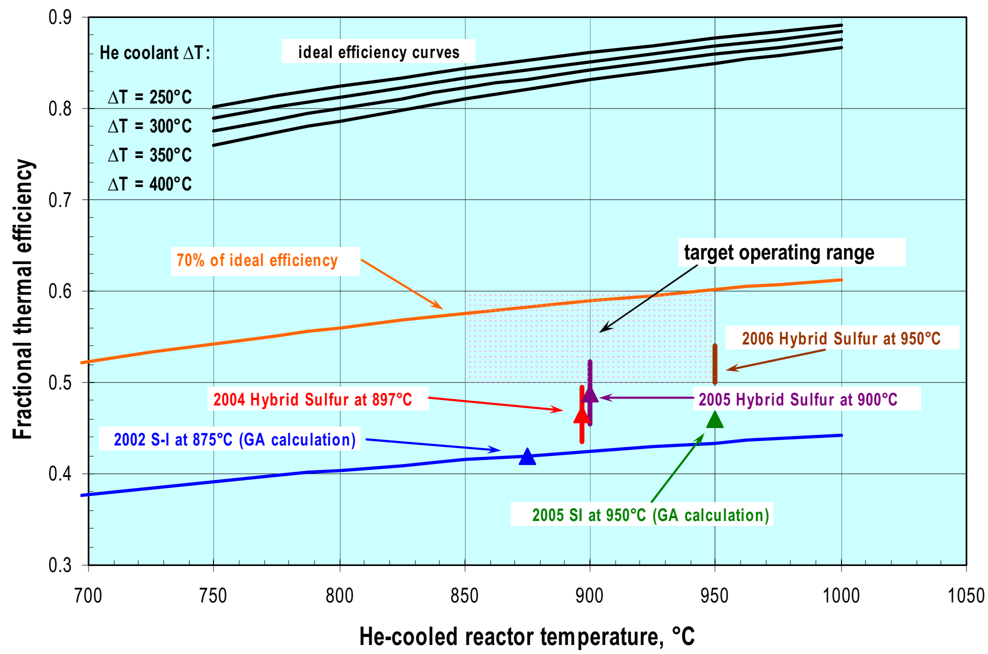


Figure 3-2: Efficiency of "Real" Sulfur Cycles As Function of Temperature

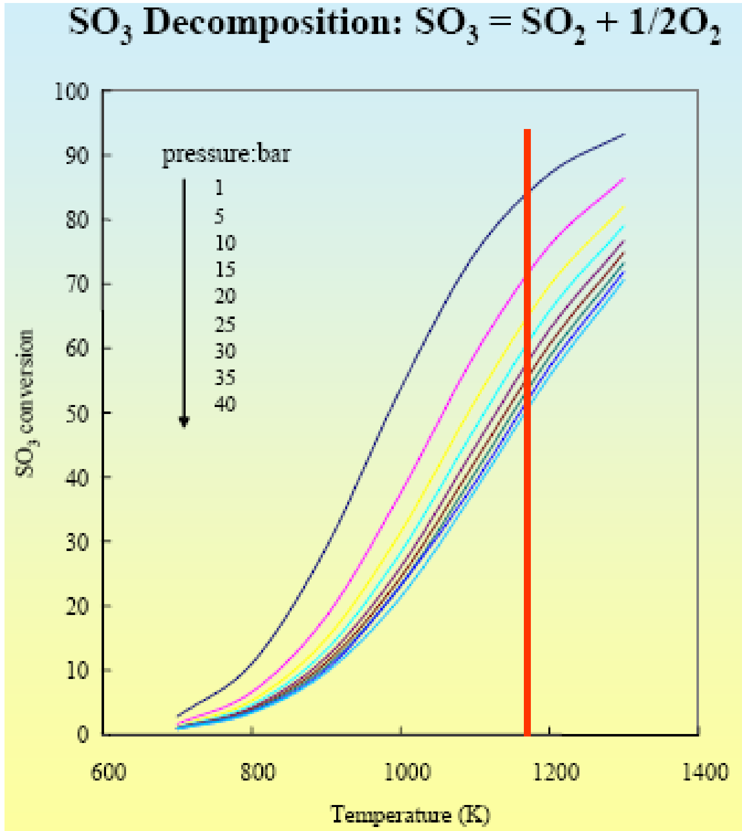


Figure 3-3: SO_3 Decomposition Equilibrium as a Function of Temperature and Pressure

Table 3-1: Summary of S-I Process Performance Temperature Needs

Reactor Outlet Temperature	Assumed H_2 Process Temp	H_2 Process Performance
950°C	900°C	Very Good
925°C	875°C	Good
900°C	850°C	Good
875°C	825°C	OK
850°C	800°C	Marginal

3.1.2.2 Nuclear Heat Source Feasibility Considerations for Reactor T_{out}

The NHS feasibility concerns due to increased reactor outlet temperature are associated with three main parts of the system:

- Core design
- Hot duct
- IHX

The ceramic support structures are not sensitive to high temperatures. Therefore, changes to the reactor outlet temperature would not change affect these structures except for the possible thickening of the graphite between the reactor outlet plenum and the metallic core support structure, which is only a design detail.

3.1.2.2.1 Core Design Considerations for Reactor T_{out}

The main effect of modified reactor outlet temperature on the core design is the need to control operating fuel temperatures. A higher core outlet temperature will result in higher operating fuel temperatures, if the core design is not modified to compensate for the higher temperature. Changes in operating fuel temperature may affect fuel performance including normal operating releases as well as subsequent accident performance or the fuel qualification. Greater operational release could result in higher circulating activity and increased primary circuit component plateout unless the compensating helium purification improvements are implemented.

If operating releases increase, maintenance could be more challenging due to increased contamination of some components. This is a greater concern for direct cycle systems, since the higher maintenance gas turbine would be contaminated. It is less important for indirect cycles, but it is still an issue. Thus, increased operating fuel temperatures could require compensating adjustments in other systems, possibly including greater coolant purification, more complex maintenance procedures, and increased safety margins. Obviously it is desirable to minimize the need for such adjustments to the reference ANTARES concept.

The goal for AREVA's NGNP design is to avoid a significant increase in peak operating temperature from the reference ANTARES core design.

The key to controlling operating fuel temperatures is appropriate core design optimization. Modern light water reactor (LWR) cores are highly optimized to control local power generation, burnup, and fuel temperature margins. Similar techniques and design tools can be applied to modern prismatic HTR cores. Prismatic core layouts allow optimization of the fuel loading to control peak operating temperatures through variations in burnable poison, fuel particle packing fraction, and fuel enrichment.

While sophisticated core designs offer the potential to reduce peak fuel temperatures substantially, they may also increase fuel design and fabrication costs, just as they do for LWRs. However, as is the case in commercial LWRs, the cost is justified to achieve both increased fuel operating margins and increased fuel cycle flexibility.

In addition, the reactor inlet temperature would also be increased to compensate for the increased outlet temperature. Due to the thermal hydraulics of prismatic core design, this will minimize any increases in peak fuel temperature, although average temperatures will increase. This tradeoff is beneficial, since particle fuel performance variability with temperature is very non-linear. Generally, the overall performance is governed by only a small fraction of the fuel which is at the highest operating temperatures. This area will be addressed further as part of the reactor inlet temperature evaluation in Section 3.2.

In the future, advanced fuel forms such as ZrC coated particles may be beneficial in providing improved performance at substantially higher temperatures. This would require less optimization for temperature control and greater design flexibility. However, these fuel forms are not expected to be ready for deployment in the 2018 timeframe. Therefore, the current design approach does not rely on them.

3.1.2.2.2 Hot Duct Considerations for Reactor T_{out}

The hot duct channels the hot reactor outlet coolant from the reactor outlet plenum to the IHX. The hot duct generally consists of a metallic pressure boundary which is protected by an internal thermal liner. It is the thermal liner that is actually exposed to the hot gas. Similar structures are used to transport the out secondary gas from the IHX outlet. An example of a hot duct and liner concept developed for a previous HTR project is shown in Figure 3-4.

Since the hot duct is protected by an internal thermal liner, the integrity of the thermal liner is the main question associated with hot duct feasibility. The metallic pressure boundary or support tube carries the main loads while operating at only moderate temperature. The cover plates in the thermal liner are normally under very low stress, but they are directly exposed to the reactor outlet temperature. The effects of hot streaks and transient variation must also be taken into account.

The main impact of the reactor outlet temperature on the hot duct is in the selection of the thermal liner cover plate material. Metallic plates can be considered for a nominal reactor outlet temperature up to about 900°C. Ceramic plates would be used for higher temperatures.

Historically, cover plates of Alloy 800H were specified for HTR concepts with reactor outlet temperatures in the range of 750-850°C. More advanced alloys might be considered for somewhat higher temperatures. High temperature designs use ceramic cover plates for insulation. Both graphite and composite liner systems have been developed as illustrated in Figure 3-5. In the 1980s, ceramic hot duct concepts were tested in high temperature service conditions (Figure 3-6).

The result of this experience is that hot duct feasibility is not an issue, even at reactor outlet temperatures of 950°C. The selection of the appropriate hot duct concept and liner material is a design optimization issue.

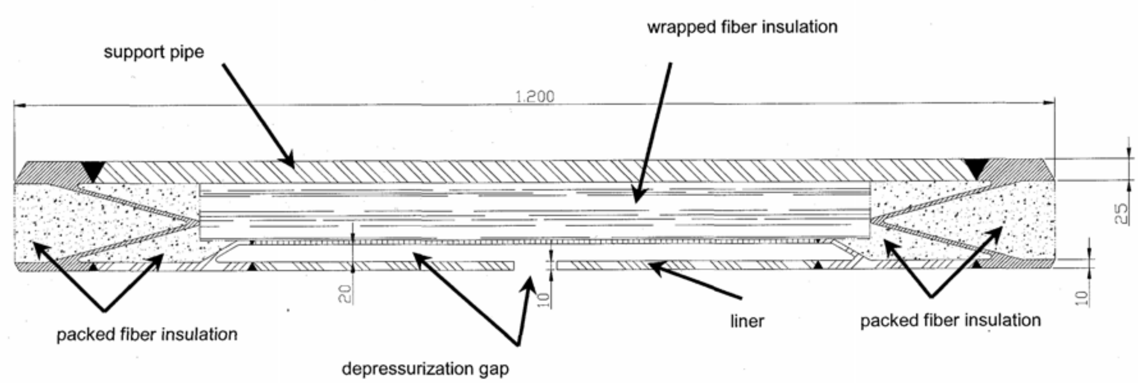


Figure 3-4: Hot Duct with Metallic Liner Concept

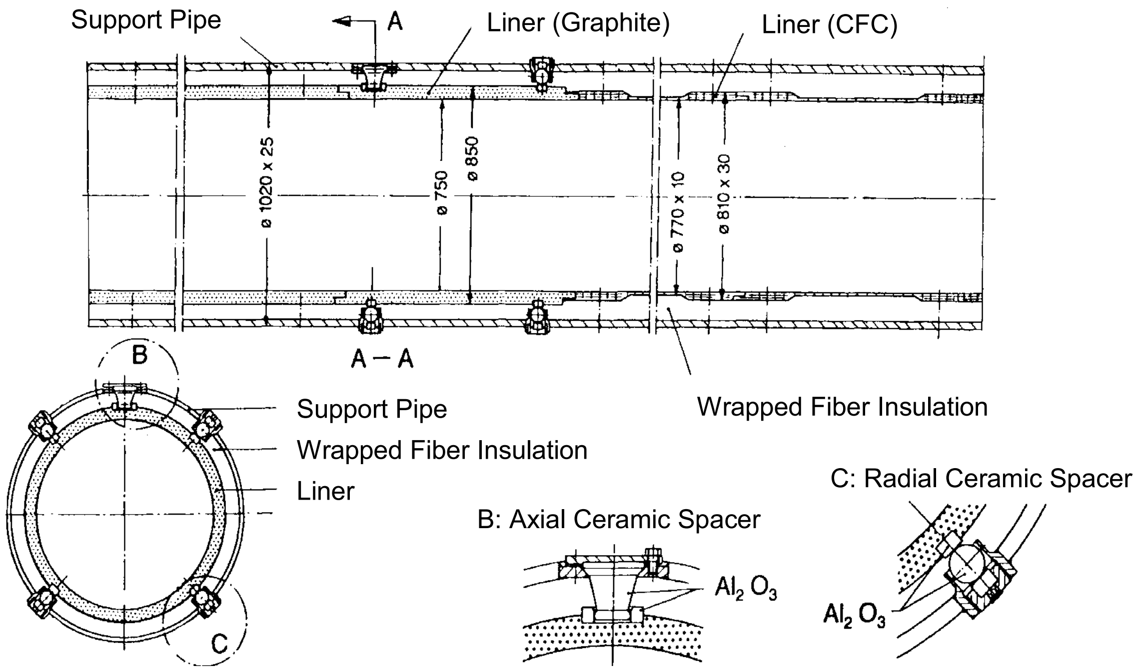


Figure 3-5: Graphite and Composite Ceramic Hot Duct Liner Concepts

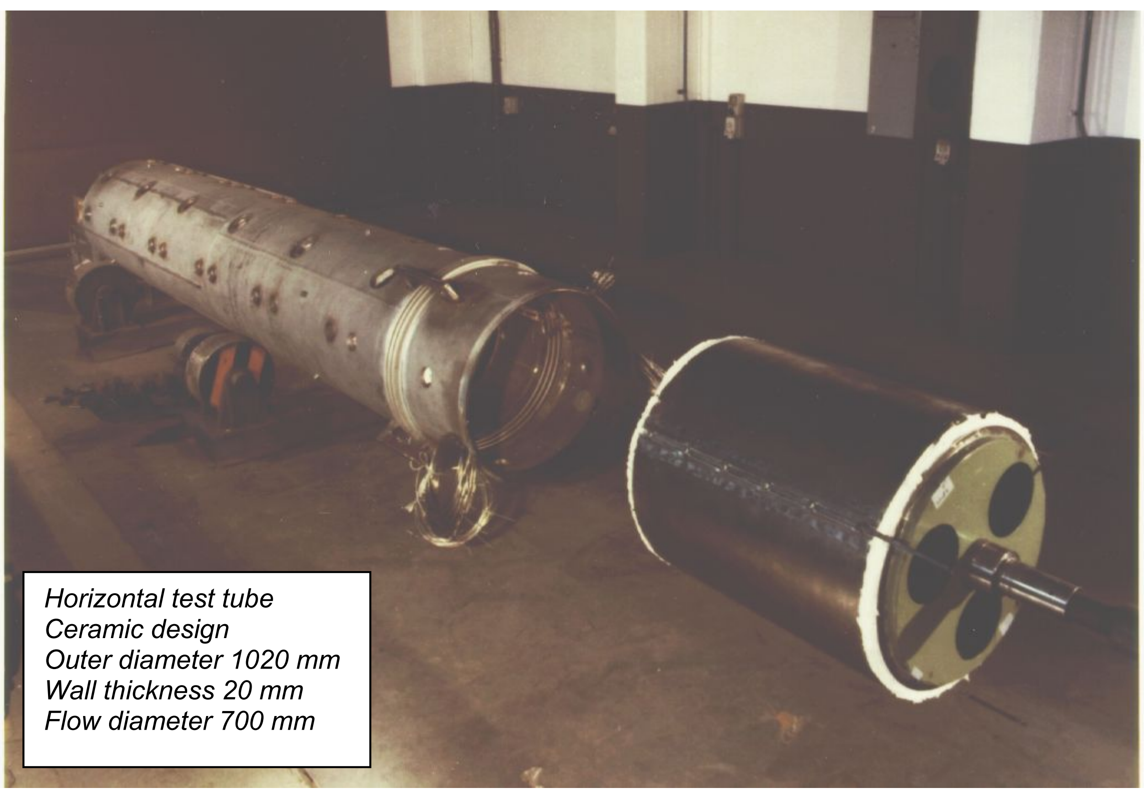


Figure 3-6: Ceramic Hot Duct and Liner Demonstration Element

3.1.2.2.3 IHX Considerations for Reactor T_{out}

The IHX is a key component. The NGNP IHX transfers heat from the primary coolant to the intermediate heat transport loop for use in the hydrogen process. With an indirect cycle PCS, the IHX also transfers the remaining reactor heat to the electricity generating system. In a commercial VHTR, the IHX must transfer all of the reactor output to the heat user.

Both compact heat exchanger and tubular IHX concepts are being considered for VHTR applications. The tubular heat exchangers are related to conventional shell and tube heat exchangers. They commonly use a helical coil tube bundle with the secondary coolant flowing in the tubes and the primary coolant flowing over the bundle in a counterflow arrangement. Tubular IHX concepts are relatively robust and they have been demonstrated at high temperature, but they require a large heat transfer volume. A variety of compact heat exchanger concepts such as plate-fin are also being considered. They require much less heat exchanger volume than tubular concepts, although internal ducting connecting individual modules uses up some of the space savings. Compact heat exchangers offer significant potential, but they have not been demonstrated in the required high temperature service regime.

The IHX feasibility depends directly on the reactor outlet temperature. The IHX heat exchange surface is a primary coolant boundary surface that must accommodate the reactor outlet gas without any thermal protection. The IHX is the only metallic component which must do this. In the future, ceramic heat exchangers may be developed which are optimized for these conditions, but they are not expected to be available in the time frame of the NGNP or early commercial VHTR plants.

Advanced high temperature alloys such as Inconel 617 and Haynes 230 are being considered for IHX fabrication. However even these alloys are approaching their practical limits at the anticipated IHX operating temperature. The strength of the base material decreases rapidly as the temperature goes to 900°C and above. Figure 3-7 illustrates allowable stresses for Inconel 617 as a function of temperature based on a previous ASME code case that was never pursued. The allowable stresses for reasonable component lifetimes are only a small fraction of their value at lower temperatures.

Bonding techniques are also limited at these temperatures. Welding is possible but it is primarily applicable to tubular IHX designs. Compact heat exchanger IHX designs typically rely on diffusion bonding or brazing to fabricate the heat exchanger core. Unfortunately, diffusion bonding or welding can have an adverse effect on the base material, due to the high temperature thermal soak required for the diffusion process. Brazing capability for high temperature service has to be demonstrated, and it is difficult to ensure consistent brazed joints through the heat exchanger core.

IHX feasibility is also strongly linked to the required component lifetime. Increasing temperature adversely affects both IHX design feasibility and design lifetime. Figure 3-8 provides a better view of the strong dependence of component lifetime on temperature for In-617 in the high temperature regime based on the KTA (German design code). A temperature increase of from 850°C to 950°C results in either a reduction in lifetime by more than a factor of 10 or a reduction in design stress by more than a factor of 3. Considering that designing for 850°C is already not easy, this poses a significant challenge.

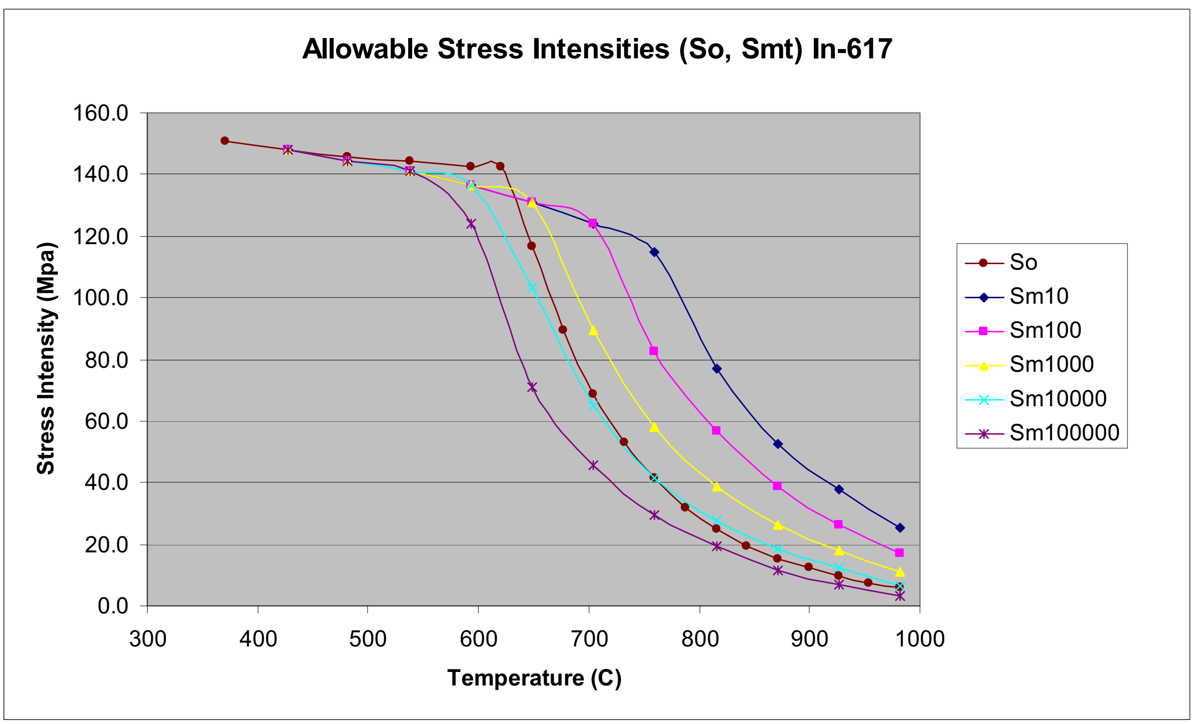
Corrosion concerns also increase at higher temperatures. Primary side metallic corrosion concerns include oxidation, carburization, and decarburization. The specific behavior is governed by very small quantities of impurities in the primary coolant. On the secondary side the same issues exist, although the impurity concentrations may differ due to the absence of the large carbon inventory present on the primary side. For the CCGT secondary system, which will likely use a mixture of nitrogen and helium to simulate air, nitriding must also be considered.

Corrosion is normally more significant at high temperatures. Strategies to control and manage corrosion in HTR systems are being evaluated by AREVA and others. In general, corrosion is viewed to be a manageable problem in the temperature regime being considered within this study. However, the corrosion problem may be more serious for very thin sections such as those in compact heat exchangers. Therefore, tubular IHX designs may offer an advantage in greater corrosion resistance. Anticipated corrosion management strategies may also be easier to implement in tubular IHX geometries. It is AREVA's view that the use of thin section compact heat exchangers at temperatures of 900°C and above may present greater corrosion management difficulties.

IHX feasibility as a function of temperature is summarized in Table 3-2. The feasibility depends on the type of IHX. Tubular IHXs have been demonstrated at high temperature, while compact IHXs have not been demonstrated in VHTR conditions. Current design efforts have faced greater challenges adapting compact heat exchangers to high temperature conditions. Compact heat exchangers have been applied successfully in the recuperator market, but the IHX application requires significantly higher temperatures and it is much less leak tolerant. Tubular heat exchangers provide greater design margin and flexibility in adapting to the required IHX conditions. The feasibility also depends on the intended environment due to corrosion considerations. In general, helium environments are expected to be more benign.

In helium-to-helium service, the tubular IHX is considered feasible over the full temperature range considered in this study. The compact heat exchanger is considered feasible at 850°C and moderately feasible at 900°C. In helium-to-nitrogen/helium service, the tubular IHX is considered feasible up to 900°C, but feasibility declines as temperature increases beyond that point. Compact heat exchangers are not believed to be very feasible at 900°C or above in helium-to-nitrogen/helium service.

Of course heat exchanger performance in terms of size and effectiveness is also a consideration. Tubular IHXs are expected to have longer lifetime but higher initial cost. The recommendation of tubular or compact IHX will be further discussed in Section 3.4.



(based on previous draft In-617 code case)

Figure 3-7: Candidate Inconel 617 Allowable Stresses

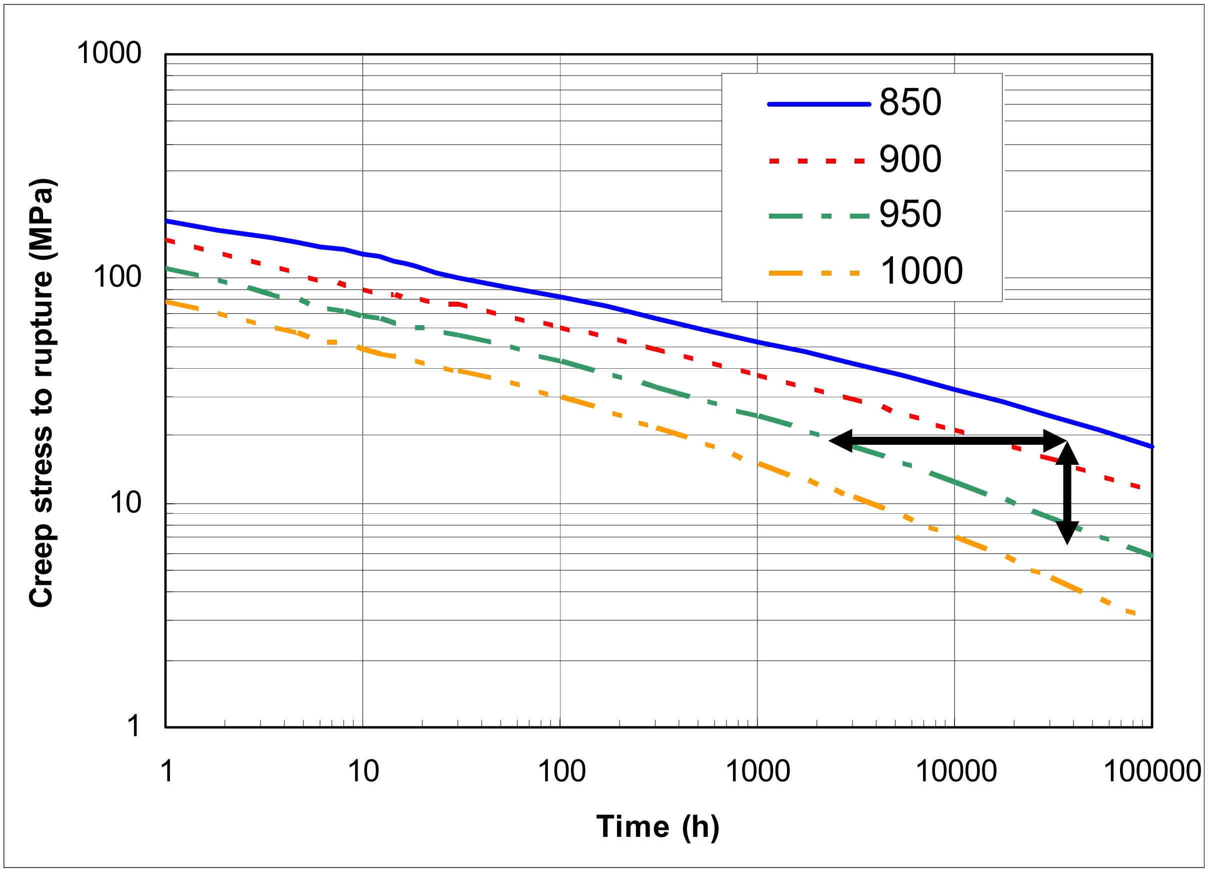


Figure 3-8: Creep Rupture Stress for In-617 (S_r curve KTA base metal)

Table 3-2: Summary of IHX Feasibility

Reactor Outlet Temp (°C)	Compact HX Feasibility		Tubular HX Feasibility	
	He/He	He/N-He	He/He	He/N-He
950	Low	Low	Good	Low
925	Low	Low	Good	Moderate
900	Moderate	Low	Good	Good
875	Good	Moderate	Good	Good
850	Good	Moderate	Good	Good

3.1.2.3 Other Considerations for Reactor T_{out}

The feasibility of the PCS is not impacted by reactor outlet temperature over the range of interest in this study. Air breathing turbines routinely operate at much higher temperatures under more adverse environmental conditions.

The performance of Brayton topping cycles is temperature dependent. Increasing the reactor outlet temperature from 850°C to 900°C translates into an increase in turbine inlet temperature from 800°C to 850°C. This change alone would be expected to give a 1.5-2.0% increase in cycle efficiency (e.g., 47% to 49%). This would suggest a preference for higher operating temperatures. However, around this temperature range, there is a threshold where turbine cooling will be required. Thus, the need for turbine cooling might reduce the efficiency increase to about 1%.

Overall, PCS considerations do not result in a strong preference for reactor outlet temperature. There is a slight preference for higher temperatures, but this is overshadowed by the NHS feasibility concerns at high temperature.

Cost factors are evaluated to see if they have a strong influence on the reactor outlet temperature decision. For the NHS cost, the main factors affected by increased reactor outlet temperature are IHX design and fabrication costs and any indirect impact on the system resulting from corresponding increases in reactor inlet temperature. While IHX feasibility varies significantly over the temperature range of interest, the actual variation in cost over that range is not expected to be important in terms of the overall plant cost when compared to the corresponding impact on system performance.

The PCS costs are not impacted significantly.

The impact of higher temperatures on the heat transport loop cost was not assessed. This impact would be expected to be comparable to the impact on the hot duct, but at a slightly reduced temperature. Cost of the heat transport loop might increase slightly due to the potential need for larger piping or higher temperature. Higher temperature materials might also be required in the hydrogen plant, but these costs would be expected to be compensated for by reduced equipment sizing due to increased conversion efficiency at higher temperature.

3.1.3 Reactor Outlet Temperature Issue Conclusion

The impact of the reactor outlet temperature on the evaluated considerations is summarized in Table 3-3. Two main factors, the hydrogen plant performance and the NHS feasibility, clearly dominate this decision. The hydrogen plant performance is marginal at a process temperature of 800°C. It is somewhat better at 825°C, but a process temperature of 850°C or above is really preferred to ensure acceptable performance. By comparison the NHS is clearly feasible up to a reactor outlet temperature of 900°C, but feasibility is less certain above that temperature. The resulting overlap in these ranges suggests 900°C as the obvious recommendation.

The other considerations identified do not provide a significant basis to modify this recommendation. The PCS concept is compatible with any of the temperatures considered, and the variation in net PCS performance is not strong. In terms of required R&D and schedule risk, most of the temperatures considered would require a very aggressive schedule in order to achieve NGNP startup in 2018, but success should be possible given adequate resources. However, if a reactor outlet temperature of 950°C were selected, it could prove to be extremely difficult to meet the desired startup date. Commercial applicability of any of the proposed temperatures would be good, although an outlet temperature of 850°C might have slightly less market flexibility (albeit with likely earlier market penetration).

Therefore, a reactor outlet temperature of 900°C is recommended. This confirms the initial AREVA NGNP preconceptual design baseline.

Table 3-3: Summary of Reactor Outlet Temperature Considerations

Reactor Outlet Temperature	Importance as Discriminator	850°C	875°C	900°C	925°C	950°C
Assumed H_2 Process Temperature		800°C	825°C	850°C	875°C	900°C
H_2 Performance	HIGH	Marginal	OK	Good	Good	Very Good
NHS Feasibility	HIGH	Very Good	Good	Good	Low	Very Low
PCS Performance	low					
R&D/Schedule/Risk	low					
Plant Cost	low					
Commercial applicability	low					

3.2 Issue 2 – Evaluation of Reactor Inlet Temperature

The second decision to be addressed is the reactor inlet temperature. A summary of the issue is reviewed in the following subsection. The next subsection provides the evaluation of the discriminating considerations. Then the conclusion for this decision is summarized in the final subsection on this decision.

3.2.1 Summary of Reactor Inlet Temperature Issue

Key question:

> "What is the recommended reactor T_{in}?"

Range of options:

> Overall range 400-600°C with emphasis on 500°C, 525°C, and 550°C.

Major considerations:

Nuclear heat source feasibility

Other discriminators:

PCS performance

Hydrogen plant performance

3.2.2 Assessment of Reactor Inlet Temperature Considerations

The reactor inlet temperature impacts the reactor design in several ways. The core design is affected in terms of the temperature rise across core and the normal operating fuel temperatures. The system response during conduction cooldown accidents is strongly influenced by the operating reactor inlet temperature, since it determines the initial temperature of the reflectors and the active core. Circulator performance is also affected by the IHX outlet/reactor inlet temperature.

Metallic internals such as the core barrel are not strongly affected by the reactor inlet temperature, since they are typically governed by conduction cooldown temperatures. Normal operating temperatures are generally less challenging than the accident temperature. Therefore, the metallic internals temperatures should be acceptable, assuming fuel and vessel accident temperatures do not increase significantly.

3.2.2.1 Core Design Considerations for Reactor T_{in}

For a given reactor outlet temperature and power and flow distribution, the reactor inlet temperature determines both the average and the peak operating fuel temperature. HTR core design is somewhat counter-intuitive in that, while reducing the core inlet temperature decreases the average core temperature, it actually raises the peak core temperature. This is due to the inevitable non-uniform radial power distribution.

Figure 3-9 illustrates this principle for a hypothetical VHTR core with no lateral mixing. In this example, the core outlet temperature is 1000°C, and that is the exit temperature for the average coolant channel, regardless of the inlet temperature. However with a Radial Peaking Factor (RPF) of 1.3, the temperature rise in the theoretical "hot" channel is 1.3 times the average. Therefore, in the case with the lower inlet temperature, if the RPF is 1.3, the 1.3 multiplier applies to a larger nominal temperature rise, and the outlet temperature of the hot channel actually increases.

A broader comparison of peak fuel temperature as a function of inlet temperature is provided in Figure 3-10 for different RPF and bypass flow values.

Because of this characteristic, when the core outlet temperature is increased, a corresponding increase in the core inlet temperature is beneficial to minimize any increase in peak fuel temperatures. For the AREVA NGNP design, the recommended reactor outlet temperature of 900°C corresponds to a 50°C increase in the reactor outlet temperature compared to the reference ANTARES design. In order to minimize the increase in peak fuel temperatures, a larger increase in inlet temperature is suggested. A minimum 500°C reactor inlet temperature is recommended from the core design perspective. This represents a 100°C increase from the current reference ANTARES inlet temperature, and a 50°C reduction in the core temperature rise.

In order to assess the adequacy of this change, a scoping fuel temperature calculation was performed using a simple model to evaluate the normal operating fuel temperature fractional distribution for operation 500°C inlet and 900°C outlet. The results in Figure 3-11 indicate that the estimated peak fuel temperature is in the vicinity of 1250°C. During conceptual design more detailed calculations will be required which will include the effects of cross flow and lateral conduction as well as uncertainty analysis. These factors are somewhat offsetting, since cross flow and lateral conduction will each decrease the calculated peak fuel temperatures while uncertainties will likely increase them. It is anticipated that acceptable fuel temperatures will result.

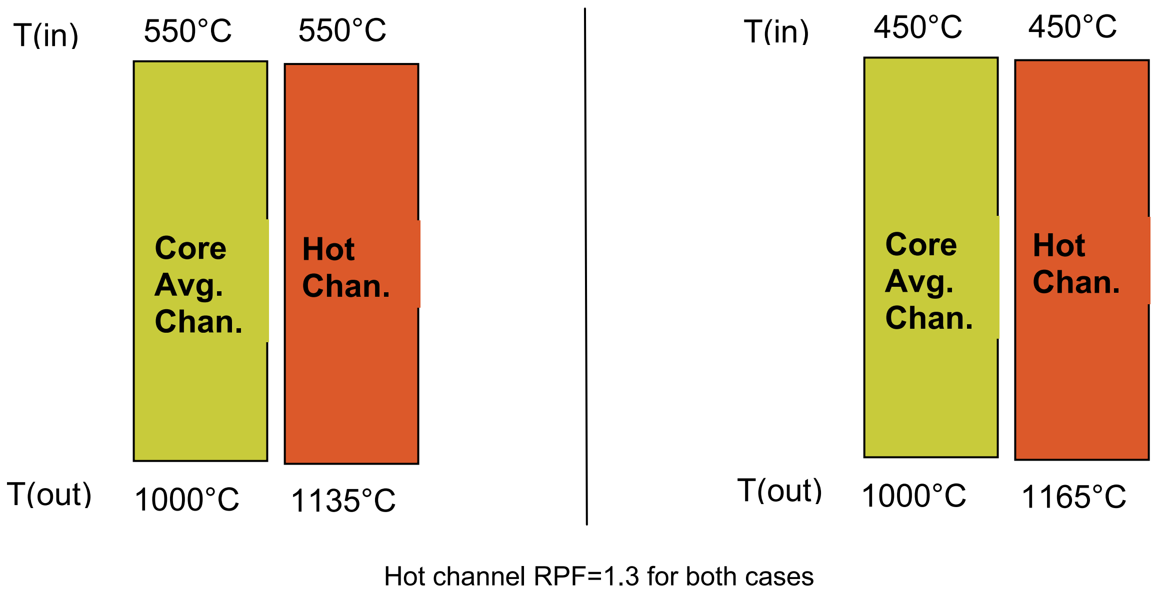


Figure 3-9: Sensitivity of Core Outlet Temperature for Average and Hot Channels In Hypothetical VHTR

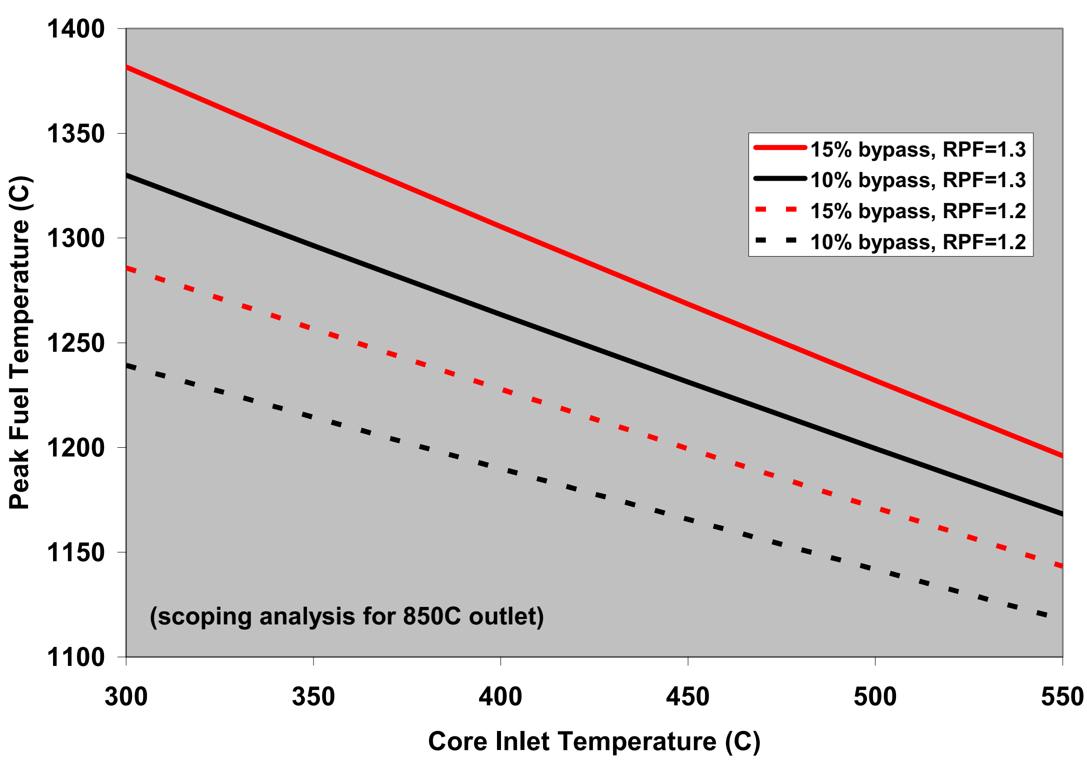


Figure 3-10: Sensitivity of HTR Peak Fuel Temperature to Core Conditions

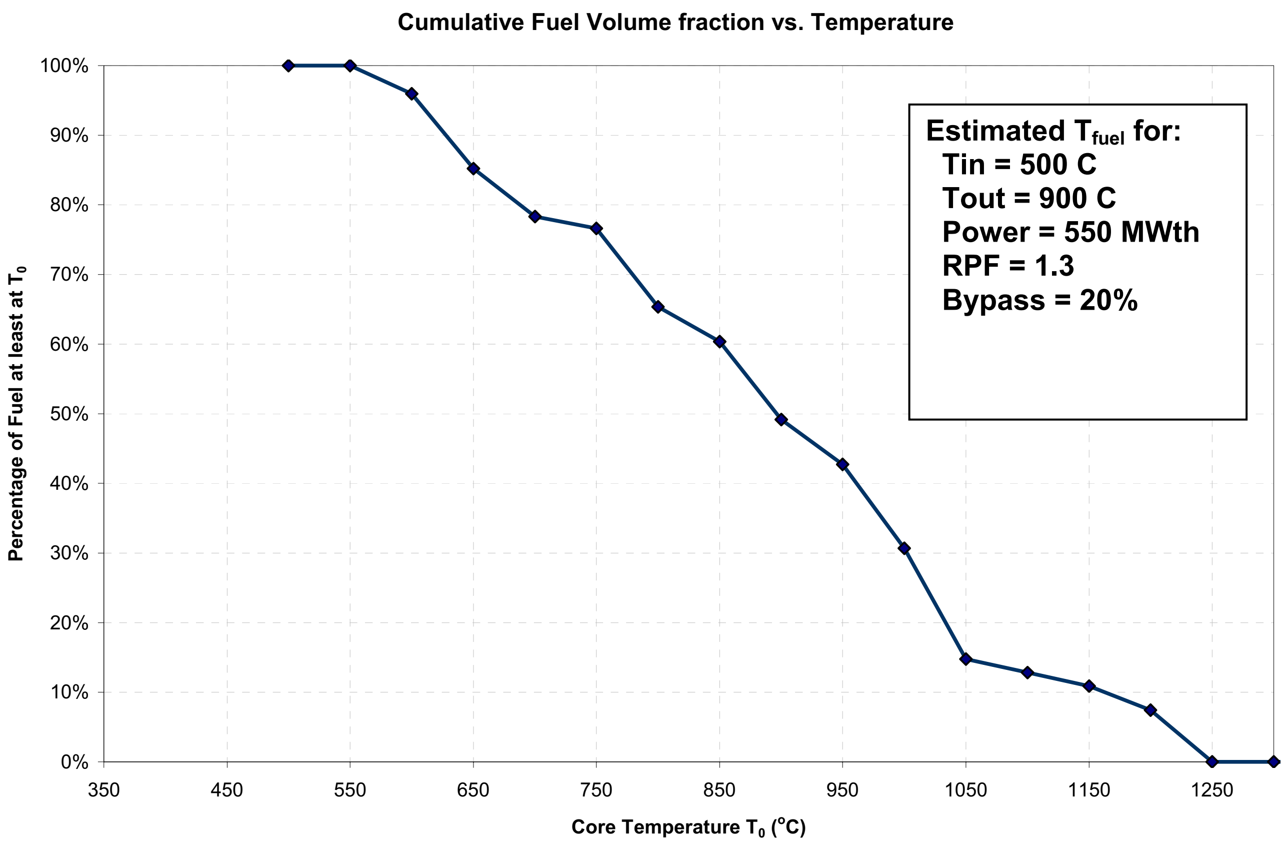


Figure 3-11: Estimated Core Fractional Temperature Distribution

3.2.2.2 Vessel Considerations for Reactor T_{in}

The effect on the vessel operating temperature is an important factor in assessing the impact of increased reactor inlet temperature. The vessel operating temperature is determined by a balance between internal heat flux and external heat flux. The internal heat flux comes from convection due to primary coolant on the inner surface of the vessel and from thermal radiation from the core barrel. The external heat flux comes from the convection and thermal radiation to the Reactor Cavity Cooling System (RCCS). Under equilibrium conditions, the reactor vessel temperature will go to the value that exactly balances these heat fluxes. If the heat flux on the inside surface of the vessel increases or decreases, then the vessel temperature will have to increase or decrease to drive more or less heat to the RCCS.

The main factor in the relationship between vessel temperature and core inlet temperature is whether or not the main coolant flow sweeps the inner surface of vessel. If so, then the core inlet temperature will dominate the vessel temperature. On the other hand, if the coolant in contact with the vessel is stagnant, then the vessel temperature will be largely controlled by the thermal radiation coming from the metallic internals. The latter is the case for the adapted ANTARES design, since the reactor inlet flow is contained within a shroud integral to the core barrel. Stagnant helium separates the internals and the inner surface of the vessel.

AREVA's goal is to keep the reactor vessel within negligible creep regime. For modified 9Cr-1Mo, this is somewhere in the range of 400 to 450°C. This range is relatively conservative, and operation at a through wall temperature in the vicinity of 425°C would likely be acceptable. For SA508, the current code limit is 371°C. This is a hard limit so additional margins to cover transients and operational uncertainties would have to be provided, resulting in a more likely upper bound for the nominal operating point of 350°C.

There are several approaches to controlling the vessel temperature during normal operation:

- Maintain low reactor inlet temperature

  If the reactor inlet coolant temperature is maintained below the creep regime temperature limit, then the issue is eliminated. However, this imposes unrealistic constraints on the core designer and on the hydrogen production process interface.

- More detailed thermal analysis and material characterization

  Due to the convective heat transfer film temperature drop and the temperature gradient within the vessel, the vessel will always be at a lower temperature than the coolant, even with direct contact. Detailed heat transfer calculations can be performed to precisely quantify this. In addition, better characterization of the boundary of the negligible creep regime might allow reductions in current design margins.

- Provide limited thermal protection

  Modest internal protection (e.g., baffles, radiation shields, thermal insulation) may provide significant vessel temperature reduction under operating conditions. Such protection would have a moderate impact on conduction cooldown accident performance. However the impact would mostly be on the vessel temperature (beneficial decrease) and core barrel (slight increase within limits). The peak accident fuel temperature is not strongly coupled to the vessel and only very slight increases would be expected.

Active cooling of the vessel provides another alternative. However, this option is not preferred. It undermines the HTR passive cooling strategy and it increases parasitic heat loss during normal operation. It also increases complexity and it has the potential to reduce plant availability in the event of cooling system failure.

3.2.2.3 Conduction Cooldown Considerations for Reactor T_{in}

Normal reactor inlet temperature affects conduction cooldown peak fuel temperature. The large heat sink formed by the inner and outer reflectors plays a major role in limiting peak fuel temperatures during conduction cooldown. Since the reactor inlet temperature sets the initial temperature of these reflectors at the start of the transient, an increase in inlet temperature has a direct bearing on the amount of heat the reflectors can absorb. This results in higher peak fuel temperatures during conduction cooldown unless compensating changes are made to the design.

The peak fuel temperature during depressurized conduction cooldown (DCC) is a function of several parameters including core geometry, initial temperature, initial reactor power level and resulting decay heat production, and the irradiation history of the core graphite and the resulting impact on effective thermal conductivity. The NGNP is designed to avoid significant fuel failure during DCC by preventing the peak fuel temperature from significantly exceeding an accident temperature guideline of 1600°C. If the reactor inlet temperature (and hence the initial core temperature) is increased, the most straightforward compensating design is to reduce the normal operating power level slightly to maintain DCC performance. This avoids the need for significant redesign of the core geometry and the need for a larger core annulus. It also maintains the flexibility to provide higher power levels in other applications with lower operating temperatures.

Detailed conduction cooldown conditions have not been performed yet for the NGNP configuration and operating conditions. Nonetheless, DCC performance has been estimated using existing sensitivity analyses. Figure 3-12 estimates the maximum allowed power level as a function of reactor inlet temperature. For a given core inlet temperature, the indicated power level is expected to provide peak DCC temperatures near the 1600°C guideline. The figure is based on anticipated NGNP conditions and it includes an estimate of the required conservatisms for uncertainties and design margins. Based on this analysis, a reactor inlet temperature of 500°C would permit a nominal power level of 565 MWth, and a temperature of 525°C would permit 560 MWth.

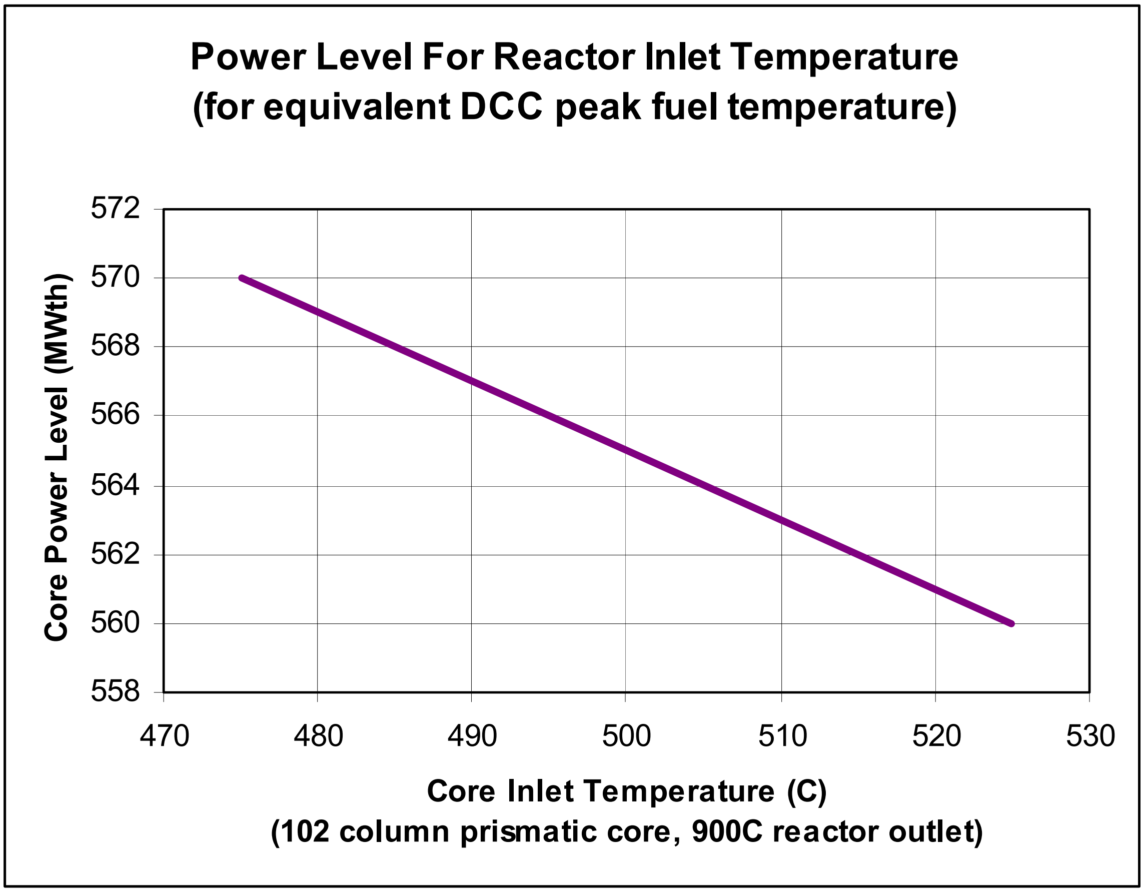


Figure 3-12: Permitted Power Level for Equivalent DCC Performance

3.2.2.4 Circulator Considerations for Reactor T_{in}

As the reactor inlet temperature is increased the required circulator power will increase slightly. For a given power level, reducing the core temperature rise results in increased mass flow. This results in increased pressure drop and a need for higher pumping power. The reference ANTARES configuration uses a single main primary coolant circulator. However, for NGNP conditions, multiple circulators would likely be required due to the increased power requirement.

A higher reactor inlet temperature also reduces the gas density at the circulator. This may adversely affect circulator performance. Higher coolant temperature in the vicinity of the circulator may also require increased thermal protection and cooling for motor cavity.

The basic feasibility of the circulator is not a concern in the anticipated range of operating conditions. Adequate materials are available to allow design for operation to above 500°C. However, the need for multiple machines will have to be considered in light of anticipated technology development. This will be addressed further in Section 3.4.

3.2.3 Reactor Inlet Temperature Issue Conclusion

Based on these considerations, the reactor inlet temperature recommendation is 500°C. This confirms the existing AREVA NGNP design baseline.

The increased reactor outlet temperature necessitates a larger increase of the reactor inlet temperature in order to maintain fuel temperatures within the desired range. Hence, the increase in outlet temperature from 850°C to 900°C suggests a corresponding increase in inlet temperature from 400°C to 500°C.

At temperatures above 450°C, thermal protection for vessel may be required. However, very modest thermal protection is expected to be adequate at 500°C, and any impact on accident fuel temperatures is expected to be negligible.

With an increased core inlet temperature, a slight reduction in nominal reactor power (about 6 % compared to the reference ANTARES) compensates for increased heat sink temperature during accidents.

The circulator feasibility is not challenged by higher reactor inlet temperatures. However, the increase in cold leg temperature does increase the likelihood that multiple circulators could be required to deliver the required pumping power.

The hydrogen production process also exhibits a preference for slightly higher return temperatures in the 550°C range.

The key point is that a reactor inlet temperature of 500°C provides greater core design margin.

3.3 Issue 3.1 – Evaluation of System Configuration – Parallel or Series

The next decision has two parts. This section examines the first of those, which is whether the system configuration for heat supply to the hydrogen process should be in parallel or series with the heat supply to the PCS. A summary of the issue is reviewed in the following subsection. The next subsection provides the evaluation of the discriminating considerations. Then the conclusion for this decision is summarized in the final subsection on this decision.

3.3.1 Summary of Parallel or Series Issue

Key question:

> “Should the H2 and PCS heat loads be in series or parallel?”

Range of options:

> Parallel hydrogen process IHX and PCS, or
>
> Series hydrogen process IHX and PCS

Major considerations:

Hydrogen plant and PCS performance

- Plant efficiency
- Maneuverability
- Availability
- Maintainability

NHS feasibility

Other discriminators:

Operating flexibility

Flexibility for component testing

3.3.2 Assessment of Parallel or Series Considerations

The most important consideration for this study is what the temperature range is over which the hydrogen process requires energy. The two options are shown in Figure 3-13. If the hydrogen process requires energy only within a narrow high temperature band, perhaps from 800 to 850°C, then a series configuration would be advantageous. However, if the hydrogen process requires energy over a broad range, perhaps from 500 to 800°C, then a parallel configuration would be better.

3.3.2.1 Energy User Temperature Requirements

Anticipated high temperature hydrogen production plants are complex facilities requiring heating and cooling of a number of fluid streams at various temperatures. This is true for both S-I and HTE processes. The specific energy management strategy for a particular hydrogen plant depends on the detailed process configuration. Most of the energy needs within the plant can be met through heat recovery where higher temperature streams which must be cooled are used to heat lower temperature streams. The net heat consumption that cannot be met in this way must be supplied by an external heat source such as the NHS for direct nuclear hydrogen production.

Generally, practical hydrogen processes require external heat over a relatively broad range of temperature. A typical composite heating and cooling curve for an S-I plant with a process temperature of 900°C is shown in Figure 3-14. As the composite curves show, high temperature heat is needed to preheat the vaporized sulfuric acid at temperatures from about 550°C to 700°C, and to drive the endothermic decomposition reaction at temperatures above 700°C. This means that the high-temperature heat demand can be satisfied with heat delivered over a range of about 350°C, to maximize efficiency. For example, if the supply temperature is 850°C, the return temperature need not be any less than 500°C.

Thus, range of heat input for the S-I process is of the same order as the reactor operating temperature range. This suggests a parallel arrangement. This is important, since dedicated commercial hydrogen plants will have to use energy over full reactor temperature range for practical system designs. The fact that the hydrogen process is not limited to a narrow range of energy consumption means that such dedicated plants are feasible.

The PCS energy usage temperature spectrum is also important. Maximum PCS performance is achieved using the full reactor temperature range. Conventional gas turbine technology can readily accommodate full reactor (or IHX) outlet temperature, so there is no incentive to place a separate high temperature heat load upstream of the PCS. Again, this suggests a parallel arrangement.

3.3.2.2 Operational Flexibility Considerations and System Configuration

Operational flexibility is greater for parallel loop configurations. This includes both control for normal plant maneuvering and response to upsets and other transients. In a parallel system, the primary coolant flow rate to the IHX supplying each heat load can be controlled independently. This allows the power and temperature range of each load to be matched. This flexibility is available during both normal operation and transients.

In a series configuration, the primary coolant flow rate is the same to both heat loads. Therefore the individual heat loads can only be varied by modulating the flow rate on the secondary side of each IHX, and this is only a weak control. It allows control of the heat load, but it gives little ability to control the secondary temperature. The inlet temperature control to the upstream heat load is primarily the reactor outlet temperature, while the primary temperature control for the downstream heat load is simply the outlet temperature of the upstream process. Neither of these is a particularly effective control.

In a parallel configuration, the heat load to each load can be adjusted independently through the primary coolant flow rate. This, combined with the control of flow rate on the secondary side of each IHX, allows both the secondary heat load and temperature to be varied more independently.

For example, if it is desired to reduce the temperature supplied to the hydrogen process, the only way to accomplish this with the series configuration is to reduce the reactor outlet temperature. However, this will also significantly reduce the performance of the PCS. With the parallel configuration, the primary flow rate to the hydrogen process IHX can be reduced significantly while maintaining a higher secondary flow rate. This will provide a reduced process inlet temperature, without affecting the PCS inlet temperature or performance.

Since the hydrogen process and the PCS are not tightly coupled in the parallel configuration, the need for careful coordination of startup of the individual subsystems is minimized. It is not necessary to start up the NHS, PCS, and hydrogen plant simultaneously. Moreover, load changes in either the PCS or the hydrogen plant need not affect the other system.

More importantly, load rejection for the hydrogen plant does not have a large impact on the PCS in the parallel plant configuration. The circulator in the hydrogen plant branch of the primary circuit would be tripped following the hydrogen plant loss-of-load, and PCS operation would continue unaffected.

However, in the series configuration, loss of hydrogen plant load would result in a step increase in temperature for the PCS. The system has no simple way to mitigate the loss of upstream load, because the core heat capacity is very large and the core outlet temperature can not be decreased rapidly, regardless of any rapid reduction in reactor power. Therefore, in the series configuration, the PCS must be designed for the full reactor (or IHX) outlet temperature regardless of plant configuration to accommodate transients.

Similarly, with the parallel configuration, PCS load changes do not significantly affect the hydrogen process. For either direct or indirect Brayton cycle systems, changes in electricity generation require significant changes in primary coolant flow rate and pressure distribution. In the series system it is very difficult to decouple the hydrogen process from these effects.

3.3.2.3 Testing Flexibility and System Configuration

Parallel configurations also offer significantly more flexibility for future testing of alternate plant equipment with minimal modification of other plant systems. Series configurations result in tight interfaces between the hydrogen process heat supply and the PCS. This includes both hardware interfaces within the primary circuit and process condition and control interfaces.

With a parallel configuration, it is relatively simple to test different process IHX designs and circulators. Since the process IHX loop is not directly integrated into the PCS loop, it can more easily be designed to accommodate alternate components for testing. Again, the flow rates and temperatures can be controlled independently in the parallel configuration in order to accommodate the needs of the alternate equipment in future test programs.

In addition, a parallel configuration may permit special high temperature demonstration operating modes. This is an important potential advantage that would not be possible with a series configuration. The idea would be to operate the reactor at very low power supplying heat only to the hydrogen plant.

The system might be configured to permit low power operation using only hydrogen plant IHX loop. The PCS IHX loop would be shut down and out of service. Reduced power operation (e.g., 10% reactor power) could allow significantly higher outlet temperature without adversely affecting the core. Fuel temperatures would still be within normal operating limits, and safety margins would be maintained due to the low decay heat associated with reduced power operation.

With operation in this mode, temporary demonstration testing with reactor outlet temperatures 1000°C may be achievable.

The PCS IHX loops would not be affected except for possible recirculation of hot gas local hot duct region immediately adjacent to the reactor outlet plenum.

However, operation in this mode would significantly affect the hydrogen plant IHX lifetime. Nonetheless, this would provide valuable hydrogen process performance data, and it would provide an excellent means of testing future ceramic IHX concepts in actual reactor service conditions.

More detailed analysis and design will be required during conceptual design to fully understand the range of testing possible in this mode.

3.3.2.4 Control of Parallel Heat Loads

The flexibility to match the required power to each heat load is achieved using the circulator and shutoff valve in each loop. The flow rate through each loop is controlled accurately using the variable speed circulator for that loop. A variety of control schemes are possible, but a typical scheme would be to monitor the outlet temperature on the secondary side of the IHX and use the deviation from the temperature setpoint as a bias in the circulator speed control.

If heat supply to one load is to be suspended (e.g., for electricity only operation), then the circulator is shut down for the circuit being taken out of service. A reverse pressure differential then exists across the shutdown loop due to the other loop(s) remaining in service. The loop shutoff valve or “flapper” valve in the shutdown loop closes passively due to both gravity and the reverse pressure differential, thus preventing significant backflow through the shutdown loop. Thus, high temperature isolation valves are not required for loop shutdown.

This approach has been used successfully in previous multi-loop HTR plants such as Fort St. Vrain and THTR, and it was the standard design approach for larger multi-loop HTR concepts such as the U.S. DOE 2240 MWth Steam Cycle/Cogeneration design [7].

3.3.2.5 Separate IHXs for Hydrogen Plant and PCS

It has been assumed that heat is not supplied to both the hydrogen production process and the electricity generating system through the same IHX. As an alternative, one could envision a different configuration in which a single IHX is used to supply both the hydrogen process and the PCS with a common secondary fluid carrying heat to both applications. (The alternative of a composite IHX heating two separate fluid streams is really two IHXs that are tightly coupled, but functionally somewhat separable.) In theory the use of a single shared IHX might result in a more compact primary circuit and greater operational flexibility as fluid could be shifted between loads to meet demand. However, there are several difficulties with implementing such a shared IHX concept in a real system, and it is more practical to use separate IHXs for most applications.

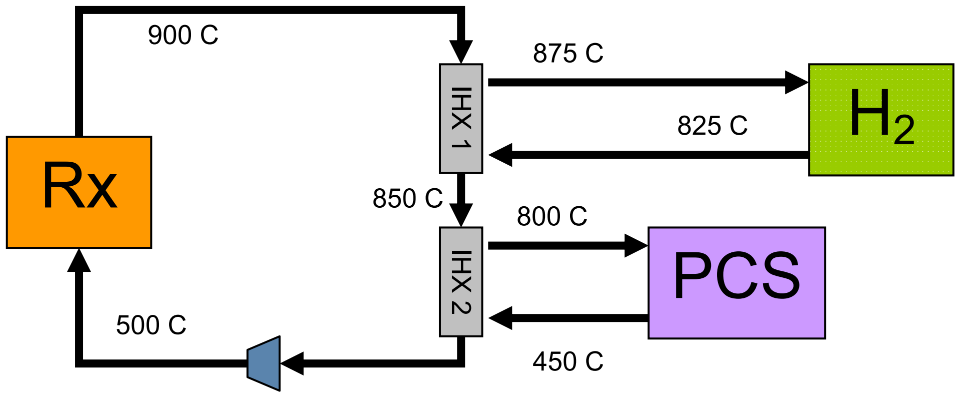
There are several reasons that separate IHXs are preferred.

First, different secondary fluids are preferred for each heat load. In the AREVA NGNP concept, the PCS is based on an indirect CCGT. This system uses a nitrogen-based fluid as a surrogate for air so that air-breathing gas turbine technology can be directly applied. For high temperature heat transport to the hydrogen process, helium is preferred for its better heat transfer properties. Moreover, in future more advanced systems other fluids such as molten salt may be attractive for heat transport.

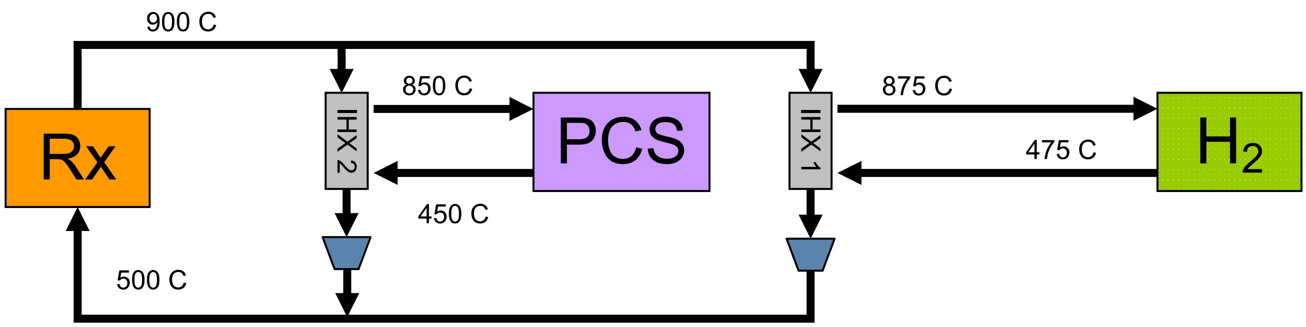
Second, even if both systems used helium as the secondary coolant, it would be desirable to keep the fluid streams separate due to the fundamentally different dynamics of the two systems. The high temperature heat transport loop works best as a constant inventory system. For any PCS involving a Brayton cycle, substantial pressure variations would be expected during load following operations and even just in normal turbomachine load control. It is preferred not the expose the heat transport loop to these variations.

Having separate systems also provides greater flexibility for future missions to demonstrate alternate IHX and heat transport equipment in the IHX supporting the hydrogen process demonstration. These demonstrations might eventually include ceramic IHX concepts and alternate heat transport fluids. The use of separate systems means that the PCS interface with the primary circuit need not be affected by such missions.

Finally, having separate systems helps to isolate contaminants such as process fluids from the hydrogen production process within individual systems.



(a) Series NGNP Configuration



(b) Parallel NGNP Configuration

Figure 3-13: NGNP Configuration Options – Series or Parallel

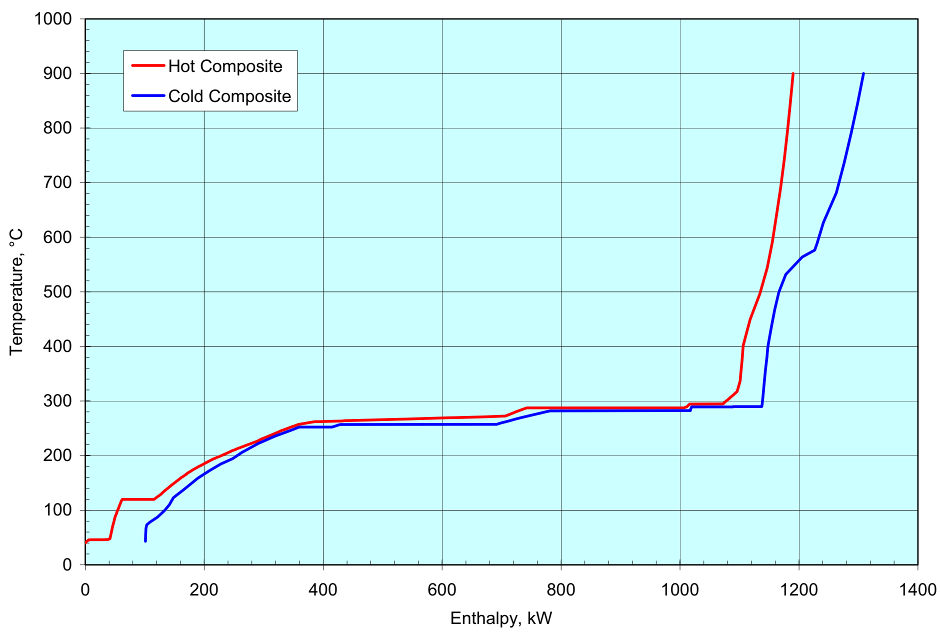


Figure 3-14: Composite Heating and Cooling Curves for the Sulfur-Iodine Process

3.3.3 Parallel or Series Issue Conclusion

The parallel system configuration is selected based on the considerations evaluated.

The parallel configuration is most compatible with the energy needs of both the hydrogen process and the PCS. The parallel system is also the most technically feasible. It has the most manageable operational characteristics. It also provides increased operational flexibility for future testing and demonstration missions.

3.4 Issue 3.2 – Evaluation of System Configuration – Number of Loops

Next, the second part of the system configuration question is examined. This section evaluates how many loops the primary system should have. A summary of the issue is reviewed in the following subsection. The next subsection provides the evaluation of the discriminating considerations. Then the conclusion for this decision is summarized in the final subsection on this decision

3.4.1 Summary of Number of Loops Issue

Key question:

> "How many loops should the system have?"
>
> For the parallel system configuration, main issue is the number of loops required to support the PCS. A separate dedicated loop is assumed for the hydrogen process.

Range of options:

> 1-4 loops

Major considerations:

> NHS feasibility
>
> Plant cost

Other discriminators:

> Plant performance
>
> Operating flexibility

3.4.2 Assessment of Number of Loops Considerations

The number of loops is a complicated question. It really includes four different parts:

- Number of cross vessels,
- Number of IHXs,

- Number of circulators, and
- How they are arranged

A single loop is acceptable, if the design is feasible with one cross vessel, one IHX, and one circulator. Otherwise, some form of multiple loop configuration is required.

A single PCS is used regardless of the number of primary loops. Existing air breathing turbines are adequate to handle the full anticipated output of a single NHS module. If multiple IHX loops are included, the secondary coolant lines are merged into a common header to supply the turbine inlet.

It is assumed that the NGNP will demonstrate multiple hydrogen production processes, with the likelihood that at least two processes might be supplied by the NHS at the same time. Single high temperature heat transport loop will carry heat to the hydrogen production demonstration facility where it will be provided to the hydrogen process(es) being demonstrated in the facility. This is the most logical approach, since the hydrogen production processes would most likely be located at some distance from the NHS and in relatively close proximity to one another. The high temperature heat transport loop would be operated at a temperature adequate to support the current process being demonstrated with the higher process temperature requirement. It would not be economical to build multiple heat transport loops to the hydrogen production area, each with a separate interface with the NHS.

3.4.2.1 Assessment of Number of Cross Vessels

The biggest impact of the number of cross vessels on the NHS design is the change in cross vessel size. When multiple cross vessels are used, the flow per cross vessel is reduced as is the vessel diameter. The cross vessel diameter is determined by the required diameter of the inner hot duct, the thickness of the hot duct thermal liner and support tube, the width of the outer cold duct flow annulus, and the thickness of the cross vessel wall. When the flow per vessel is reduced, both the hot duct diameter and the cold duct annulus thickness are reduced. An initial sizing evaluation was performed to estimate the cross vessel outer diameter for one, two, and three loops:

- 1 loop – 2.5 m
- 2 loop – 1.85 m
- 3 loop – 1.55 m

The cross vessel to the IHX dedicated to hydrogen production might be significantly smaller than this.

The potential reduction in cross vessel diameter has several ramifications for the NHS design. A key impact is on the reactor vessel ring where the nozzles to attach the cross vessels are located. The cross vessel diameter determines the size of the corresponding nozzles on the reactor vessel and hence the size of the nozzle forgings. Vessel fabrication is simpler if a single full ring forging can be produced containing the cross vessel nozzle(s). The size of such a ring forging is determined by the height of the nozzle. However, current fabrication limits determine the maximum ingot and forging size that can be obtained. It is feasible to obtain a full ring forging for the three cross vessel (1.55 m) configuration including all three nozzles. However, it is uncertain whether a single ring forging large enough for the single or two loop nozzles can be obtained in the near term. Single nozzle forgings can be obtained up to the size of a single cross vessel. A vessel can be fabricated using individual nozzle forgings and rolled plate, but it is simpler to have a single ring forging.

Adopting a multiple cross vessel configuration could have a small safety impact, since multiple cross vessels may increase the cross vessel failure probability. However, the theoretical rupture size would be smaller, reducing blowdown rates, thrust loads, and internal loadings during blowdown. The net impact is expected to be small but beneficial, since cross vessel failure is considered as a beyond design basis event. Given that the cross vessels would continue to be designed to the vessel code, that classification would not likely change and the resulting assessment would be less severe.

The reduction in hot duct diameter resulting from using multiple cross vessels would also reduce the reactor outlet plenum height. This would result in direct cost savings for graphite, reactor internals, the reactor vessel, and the reactor building. It would also provide increased margin against buckling of the core support posts.

Considering all of these factors, multiple cross vessels are preferred for both vessel fabrication and core outlet plenum height considerations. Either single or multiple loop cross vessel configurations are feasible.

3.4.2.2 Assessment of IHX Size Limitations

The recommended number of IHXs is determined primarily by IHX size limitations.

Compact heat exchangers require a modular approach. The size of a "single" IHX tends to be limited by how many modules can be connected together within a single IHX vessel.

Tubular heat exchangers are limited by the physical size of the tube bundle necessary to obtain the required heat transfer surface with reasonable primary and secondary side pressure drops. Tubular IHX capacity usually depends on the fluid. For the anticipated NGNP heat load, three loops would be required for PCS using a nitrogen-helium mixture in a tubular IHX. Two loops would be adequate for a commercial hydrogen plant using helium as the secondary heat transport fluid.

A tubular IHX heat transfer core is usually significantly larger than a compact heat exchanger, but less internal manifolds and piping is required for the tubular IHX. In addition, the vessel to house a tubular IHX is significantly smaller than compact IHX vessel. Of course, multiple such vessels would be necessary for the tubular.

Table 3-4 provides a comparison of key characteristics of compact and tubular IHX concepts. In general, tubular heat exchangers are more robust than compact heat exchangers, and they are more maintainable. Compact heat exchangers are expected to have lower initial capital cost, but they will also have shorter lifetimes and potentially higher maintenance costs.

Tubular IHX concepts are more maintainable, because individual tubes can be tested, inspected, and plugged. For compact IHX concepts, entire IHX modules would have to be replaced in the event of an IHX failure or leak. This requires a longer outage including opening of the IHX vessel for module replacement.

Another important point is that large tubular heat exchangers have been demonstrated in HTR IHX service environments. A tubular IHX built and tested at 950°C for the Prototype Nuklear Process heat (PNP), a past process heat HTR development program in Germany, is shown in Figure 3-15. Substantial development is underway on compact heat exchangers for high temperature applications, but no compact IHX concepts have been demonstrated in HTR service conditions.

The initial conclusion regarding the selection of IHX concept is that a tubular IHX is the preferred overall technology for NGNP conditions. As discussed in Section 3.1.2.2, the compact heat exchanger IHX is not judged to be feasible for the main PCS heat transfer interface in the required NGNP timeframe. In addition, the compact heat exchanger component lifetime under NGNP conditions is a concern.

In the context of the NGNP's role as a commercial demonstrator, it is important to remember that the tubular IHX is well suited to a commercial hydrogen plant configuration. For a commercial hydrogen plant using helium as the intermediate heat transport fluid, two loops should be adequate using tubular IHXs. The cost of such a system has not been evaluated, but it is likely the lowest cost system.

For NGNP conditions with a nitrogen-helium mixture on the secondary side of the PCS, three loops are required for tubular IHXs. This is because the indirect cycle CCGT is being operated at hydrogen plant temperatures.

The situation is somewhat different for the NGNP hydrogen plant IHX. This IHX is expected to be substantially smaller than the PCS IHX. Therefore, it is expected to be more practical to design a more robust compact heat exchanger configuration, even if the design is not fully optimized economically. In addition, the hydrogen process IHX will likely be designed for ready replacement in order to support future testing of alternate IHX concepts.

The resulting IHX technology recommendation is two-fold:

- Three tubular IHXs for large PCS loops
- Modular compact IHX for hydrogen loop with change out capability to demonstrate alternate concepts

This strategy is intended to maximize reliability for the PCS IHX while allowing advanced performance development and demonstration in the smaller hydrogen IHX.

3.4.2.3 Assessment of Circulator Size Limitations

Circulator size constraints determine the number of primary circulators that are needed. A commercial size HTR (i.e., approximately 500-600 MWth) requires approximately 15 MWe of circulator pumping power with a prismatic core. (Significantly more would be required for a pebble bed core due increased flow resistance.) This value may increase slightly due to NGNP conditions.

As part of AREVA's ANTARES program, various circulator suppliers were previously queried to determine the upper bounds on a practical circulator size. There are several factors which limit circulator size including rotor dynamics, impellor aerodynamics, rotor stress, motor electrical design, etc. The existing experience of probable vendors will support machines in 4-5 MWe range. This is the limit of current experience. A machine in the 12-15 MWe range is believed to be feasible by the vendors. However, significant development would be required, and they are not prepared to formally commit to such a machine today.

Therefore, it is recommended to base the NGNP preconceptual design on using three main primary circulators for the PCS heat supply and one smaller primary circulator for heat supply to the hydrogen loop IHX. This is based on the current circulator technology status and the target NGNP deployment schedule.

3.4.2.4 Assessment of IHX Vessel Considerations

The number of IHX loops will determine the number of IHX vessels, so the ramifications on the IHX vessel design are also considered.

While compact IHX concepts allow all IHX modules to be placed in a single IHX vessel, the resulting vessel is comparable in size to the reactor vessel. As a result, the compact IHX vessel requires on-site fabrication at landlocked sites (such as INL).

Multiple vessels are required for tubular IHXs, but these vessels are smaller. Therefore, vessels for multi-loop tubular IHX are transportable by land. These vessels are smaller in diameter, and they have a smaller wall thickness compared to the compact IHX vessel. This reduces the material requirements and fabrication cost for the individual vessels substantially. The cost of each tubular IHX vessel is expected to be less than half the cost of the compact IHX vessel.

Given that circulator power may be limiting for a single loop configuration and that a multiple loop arrangement facilitates the use of multiple circulators, a multiple loop configuration will likely have more circulator power margin. This means that the optimization between system pressure and system pumping power will likely arrive at a lower system pressure. Therefore, the whole system would benefit from a reduced operating pressure. This will reduce the reactor vessel cost and fabrication difficulty.

3.4.2.5 Summary of NHS Considerations for Number of Loops

The major NHS feasibility issues are summarized in Table 3-5 for configurations with one, two, and three main PCS loops.

For the cross vessel(s), the three cross vessel configurations is clearly feasible. The one and two cross vessel configurations are feasible provided that a welded vessel nozzle ring is acceptable.

For the IHX, a compact IHX would be required for a single loop configuration, and this is not judged to be ready for deployment in 2018. For a two loop configuration, a tubular IHX would work with helium secondary fluid, but not with the nitrogen-helium mixture used in the CCGT secondary circuit. For the three loop configuration, a tubular IHX is feasible with the nitrogen-helium mixture.

For the circulator, a single circulator is not expected to be available by 2018. A half size circulator for two loop operation may be achievable by 2018, but it would be a stretch from current technology. A one third size circulator could be supplied based on current technology.

The IHX vessel is feasible for any of the considered configurations. The vessel for the single loop system has a slight disadvantage, since it would require on-site fabrication.

Overall, a single loop configuration for heat supply to the NGNP PCS is theoretically possible with major development effort, but it is probably not achievable by 2018. For a commercial hydrogen production plant, a two loop configuration is probably achievable in the near-term using helium as the heat transport fluid.

For the NGNP demonstration mission, the best configuration from the NHS feasibility perspective is with three loops supplying heat to the indirect CCGT (and a 4th loop for the hydrogen production process).

3.4.2.6 Economic Considerations for Number of Loops

In assessing the number of loops, the main consideration balancing NHS feasibility is the plant economics. It is a question of fewer larger components versus a larger number of smaller components. Table 3-6 summarizes the relevant economic factors qualitatively. The two main economic factors are initial IHX capital cost and IHX component lifetime. These two factors largely offset one another.

The compact IHX concept required for the single loop configuration is expected to be significantly cheaper than the tubular concept. However this advantage is of limited value, since the concept would probably not be ready for full size deployment in the required NGNP timeframe.

Conversely, the tubular IHX is expected to have a significantly longer lifetime under NGNP conditions. This is due to lower design stresses, thicker wall sections, and a more easily optimized configuration. Required component replacement intervals could differ by as much as a factor of three or four for the two IHX concepts.

Most other factors are of secondary importance. The circulator cost is expected to be slightly lower for a single large machine, although feasibility is questionable. Vessel cost is considered to be largely a wash between the two concepts. The need for less on-site fabrication and a reduced reactor outlet plenum height are slight advantages for the multi-loop configuration. The building cost may be slightly higher for the multi-loop configuration due to a larger footprint.

Better maintainability and availability are expected for the multi-loop system, although these are only qualitative estimates at this time.

Overall, it is hard to draw a clear conclusion one way or the other from this economic evaluation. The results in Table 3-6 suggest perhaps a slight advantage for the multi-loop configuration although the assessment is certainly very qualitative.

A slightly more quantitative assessment was performed to compare the cost difference of the single loop compact IHX configuration to the multiple loop tubular IHX configuration. The results were again largely inconclusive, since one configuration or the other was judged to have a 2-4% overall cost advantage, depending on how the IHX capital cost and replacement interval and cost were estimated. (A more detailed estimate of the selected concept will be performed as part of the remaining preconceptual design studies work scope.)

What is clear is that the multiple loop configuration does not appear to have a large cost disadvantage as might have been anticipated.

3.4.2.7 Other Considerations for Number of Loops

One other consideration regarding the number of loops should be touched upon briefly. That is the point that multiple loops offer greater operational flexibility. In particular, the use of a robust tubular IHX in a multiple loop configuration may permit elimination of the Shutdown Cooling System. The regulatory implications of this decision have not been investigated, but the potential savings should be investigated further as part of future NGNP design activity.

Table 3-4: Key Comparison of Compact and Tubular IHX Concepts

Compact Heat Exchanger IHX	Tubular IHX
Compact heat exchanger (CHE) IHX can use single loop (with multiple modules)	Tubular requires multiple loops
	Tubular is more robust
	Tubular is more maintainable • Individual tube testing • ISI • Tube plugging without IHX removal
CHE may have higher operating costs • Shorter lifetime • Repair requires module replacement outage	Tubular may be more expensive initially • More IHX material • More vessels

He-He intermediate heat exchanger
before installation

Figure 3-15: Tubular IHX Built and Tested for PNP

Table 3-5: NHS Feasibility Issues for 1, 2, and 3 Main Loops

	1	2	3
Cross Vessel	Forging*	Forging*	OK
IHX	CHE Reqd.	Tubular w/ He	Tubular w/N_2
Circulator	By 2018?	Stretch	Current
IHX Vessel	Onsite Fab?	OK	OK

* Individual nozzle forgings are feasible for 1 and 2 cross vessel configurations, but full vessel nozzle ring forging probably not feasible.

Table 3-6: Economic Considerations for Single vs Multiple Loop Configurations

	Single Loop CHE IHX	3 loop Tubular IHX
IHX Cost (+ = lower cost)	+++	
IHX Lifetime (+ = longer life)		+++
Circulator Cost (+ = lower cost)	+	
System pressure (+ = lower P)		+
Vessel Cost (+ = lower cost)	=	=
Reduce On-Site Fabrication (+ = factory fabrication)		+
Reduce Outlet Plenum Height (+ = lower plenum height)		+
Building Cost (+ = lower cost)	+	
Maintainability		+
Availability		+

3.4.3 Number of Loops Issue Conclusion

Table 3-7 summarizes the comparison of the important factors in the decision on the number of loops (and the related decision on IHX type). The feasibility of the single loop system is judged to be low, primarily due to doubts about the compact IHX and large circulator feasibility in the required NGNP timeframe. In contrast, the feasibility of the multiple loop system based on the tubular IHX and smaller circulator is expected to be ready for deployment in the near-term.

The plant capital cost is judged to be better for the single loop system, primarily due to the lower IHX cost. However, the operating cost of the multiple loop system is expected to be better, because the tubular IHX is predicted to have a significantly longer service lifetime. Plant operability is also better for the multiple loop configuration.

Therefore, the three loop configuration (+ hydrogen process loop) is recommended for the NGNP. This selection maximizes feasibility and minimizes schedule risk. It also maximizes operational flexibility and maintainability. Any increase in plant cost is compensated by significantly reduced risk. Figure 3-16 shows the recommended arrangement schematically, including three loops supporting the single PCS and one loop supporting the hydrogen plant.

This recommendation is also fully compatible with future two loop commercial hydrogen plants.

More specific recommendations from the preceding discussion support this overall recommendation. These include:

- Separate cross vessels are preferred for each loop
- Tubular IHXs are recommended for large PCS loops
- Compact IHX is recommended for hydrogen loop with change out capability to demonstrate alternate concepts
- Separate circulator is recommended for each loop

Table 3-7: Overall Comparison of Single Versus Multiple Loops

		Single Loop Compact IHX	3 Loop Tubular IHX
NHS feasibility	HIGH	Low	High
Plant capital cost	HIGH	Good	OK
Plant operating cost	low	OK	Good
Plant performance - Maintainability - Availability	low	OK	Good
Operating flexibility	low	OK	Good

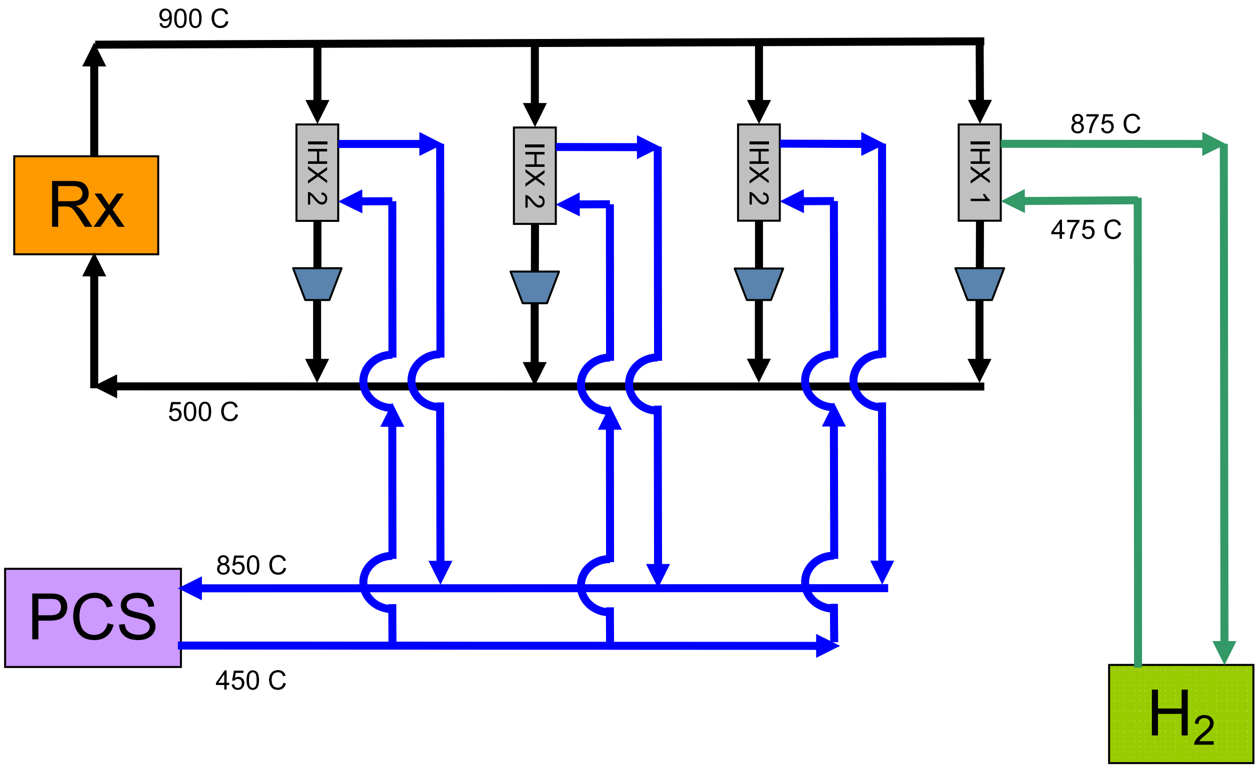


Figure 3-16: Schematic of preferred 4-loop configuration

3.4.4 Recommended Loop Arrangement

With the selection of a multiple loop configuration, a decision must be made on the recommended arrangement for the loops.

At the outset of this study, four basic arrangement types were considered as shown in Figure 3-17 through Figure 3-20. These configurations result from the number of cross vessels and IHX vessels to be used. The reference arrangement in the Design Baseline [1] was Option C (Figure 3-19).

There are two main considerations in the selection of the new NGNP loop arrangement. The first is the recommendation from the preceding sections for the number of IHXs, circulators, and cross vessels. The second is the impact of the configuration on building size and cost. A secondary consideration is how well the proposed NGNP configuration supports the deployment of future commercial plants.

Given the recommendation to use multiple cross vessels, Option D is preferred. The use of three main cross vessels in Option D (plus one for the hydrogen loop) provides somewhat improved vessel nozzle ring fabrication capability compared to the two vessels in Option C. This is reinforced by the recommendation to use a separate circulator for each loop. In Option C a hub separates each pair of IHX vessels. If each pair of IHXs was to be supported by a shared circulator, then this hub would have provided a convenient circulator mounting location.

However, with dedicated circulators attached to each IHX vessel, this hub only adds additional design complexity. The simplest design solution is the completely independent loops of Option D.

Support of the IHX vessels along the cross vessel axis is also more straightforward for Option D.

Originally it had been assumed that Option C might have an advantage in compactness compared to Option D. However, evaluation of the arrangement has not born this out. At the current level of detail, the two options are indistinguishable. Option C does increase the primary circuit complexity, since two IHXs must be connected to a single cross vessel.

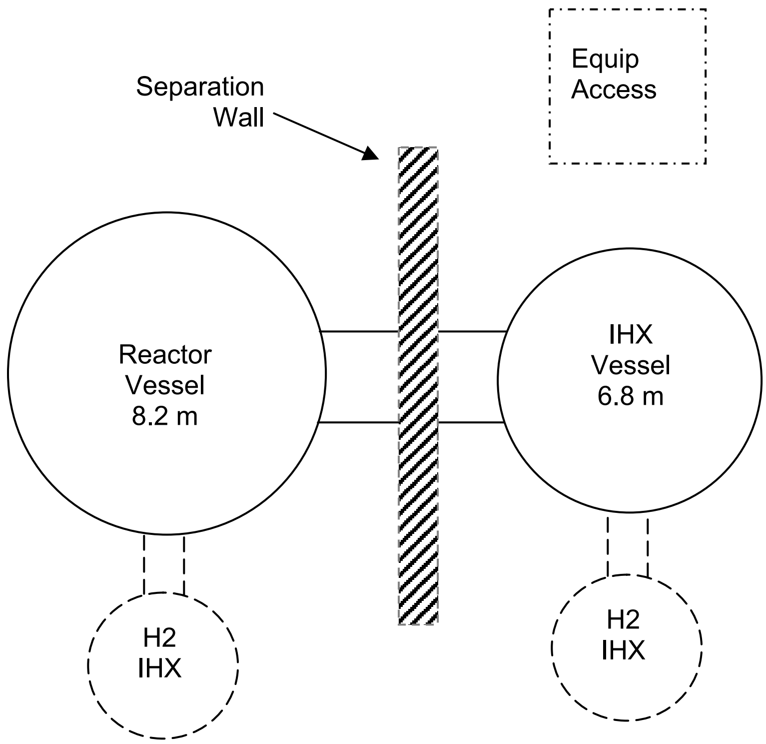
Option B was considered as an early alternative to the single IHX vessel with compact heat exchangers. The close arrangement of four tubular IHX's was intended to be compatible with the building footprint of Option A. Option B would also allow four IHXs to share a single cross vessel and possibly a single circulator. However, Option B does require a significant increase in complexity to connect the ducts from four IHX vessels to a single cross vessel. In addition, in order to provide a compact arrangement, Option B does not provide significant separation between IHX modules. This eliminates any benefit of physical separation between redundant systems, and it complicates maintenance.

The Option D building size is slightly larger than for Option A, but this disadvantage applies only to the NGNP. As shown in Figure 3-21, for a three loop commercial electricity plant or a two loop commercial hydrogen plant, the building design is smaller than the Option A layout. (Note that the diameter shown for the two loop configuration may not be practical, since it does not allow significant space for other utilities and access paths within the structure. The actual size might be closer to the three loop layout.)

Of course, this is a very preliminary analysis. Building cost is more complex than simply the diameter of the building. Nonetheless, this initial comparison suggests that the Option A arrangement does not have a significant building cost advantage.

The Option D configuration also provides the greatest fidelity to the anticipated commercial plant configurations in terms of operational characteristics. This will allow demonstration of anticipated loop control strategies and plant protection schemes in the NGNP plant.

Hence the recommended arrangement is Option D with completely independent loops. This is a change for the existing baseline [1].

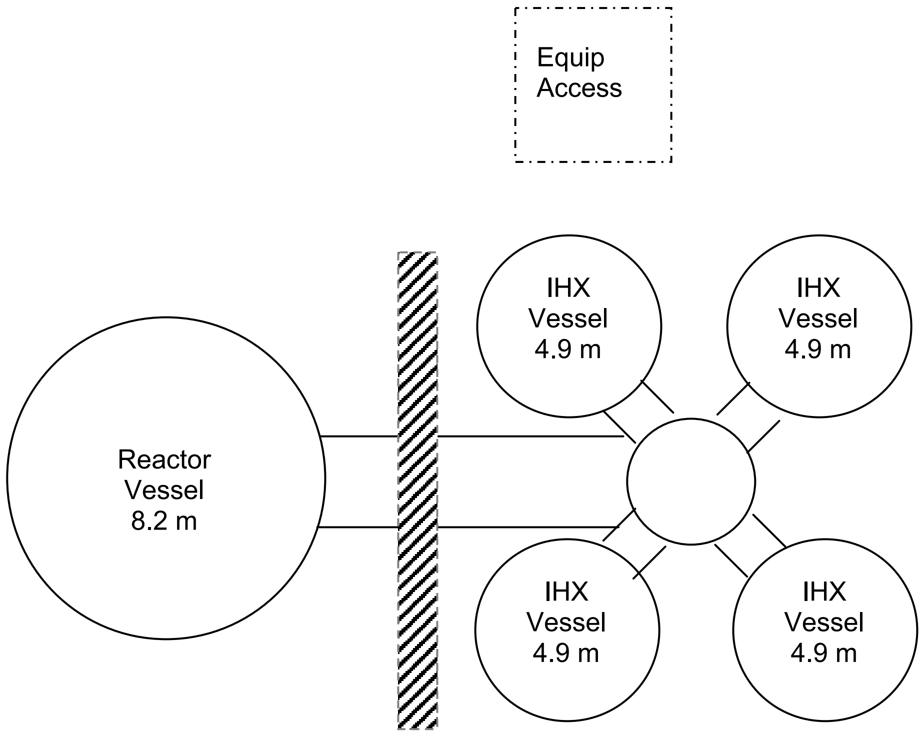


Circulator on bottom of IHX vessel

Possible IHX for Hydrogen Plant shown with dashed lines

Option	Cross Vessels	IHX Vessels	Circulators	H_2 Plant IHX Location
A-1	1	1	1	Integral in IHX vessel
A-2	1	1	2	Integral in reactor vessel
A-3	1	2	TBD	adjacent to PCS IHX vessel
A-4	2	2	2	Separate IHX vessel and cross vessel

Figure 3-17: Loop Arrangement Option A – Single Loop with Compact IHX



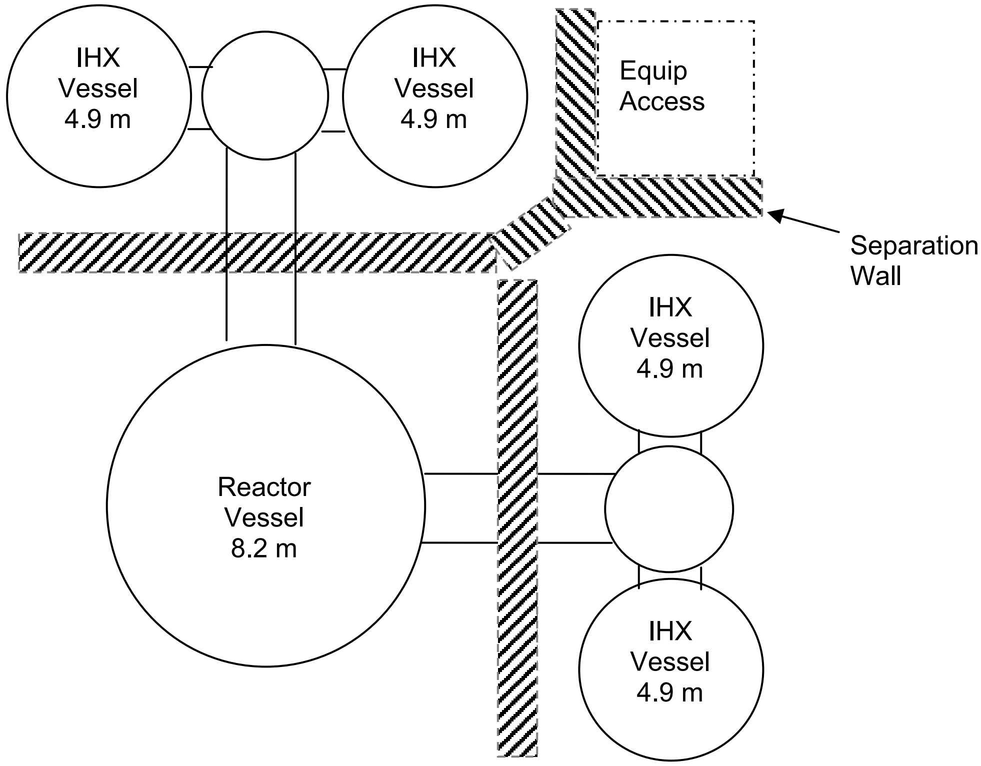
Assume 3 IHX provide heat to PCS

Assume 1 IHX provide heat to H_2 Plant *

Option	Cross Vessels	IHX Vessels	Circulators	Circulator Location
B-1	1	4	1	Central IHX Dist Hub
B-2	1	4	2	Header between IHX pairs
B-3	1	4	4	Bottom of IHX Vessel

* H_2 Plant IHX vessel may be smaller than PCS IHX vessels

Figure 3-18: Loop Arrangement Option B – Single Cross Vessel with Multiple Tubular IHXs



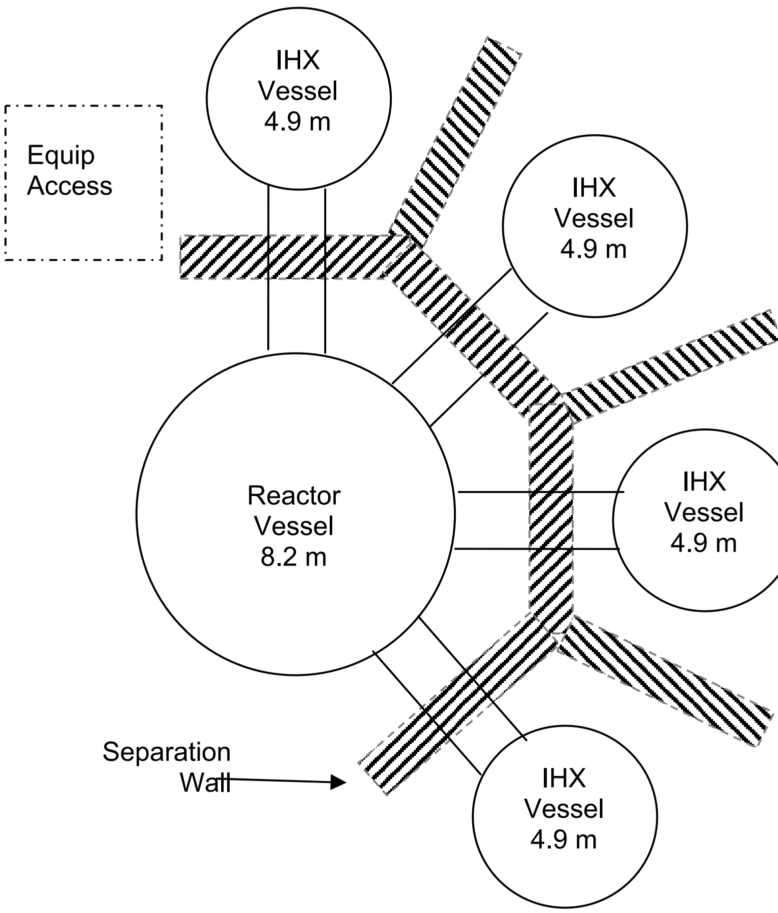
Assume 3 IHX provide heat to PCS

Assume 1 IHX provide heat to H_2 Plant *

Option	Cross Vessels	IHX Vessels	Circulators	Circulator Location
C-1	2	4	2	Header between IHX pairs
C-2	2	4	4	Bottom of IHX Vessel

* H_2 Plant IHX vessel may be smaller than PCS IHX vessels

Figure 3-19: Loop Arrangement Option C – Double Cross Vessels with Two Pairs of Tubular IHXs



Assume 3 IHX provide heat to PCS

Assume 1 IHX provide heat to H_2 Plant *

Option	Cross Vessels	IHX Vessels	Circulators	Circulator Location
D-1	4	4	4	Bottom of IHX Vessel

* H_2 Plant IHX vessel may be smaller than PCS IHX vessels

Figure 3-20: Loop Arrangement Option D – Four Tubular IHXs and Four Cross Vessels

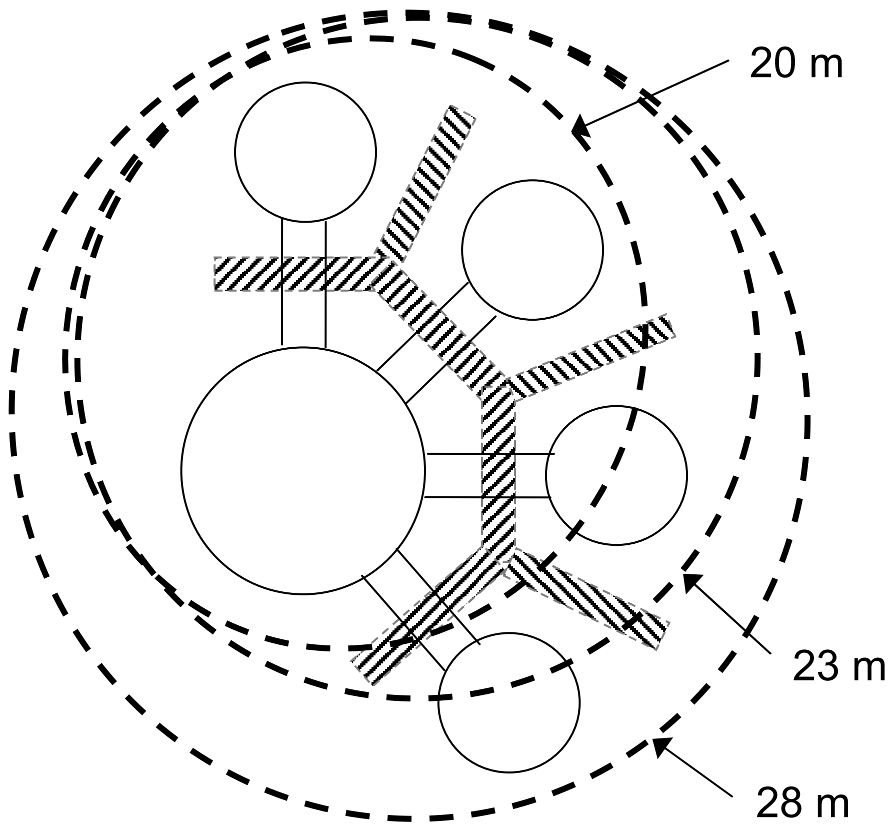


Figure 3-21: Reactor Building Outline Comparison for 2, 3, and 4 Loop Arrangements

3.5 Issue 4 – Evaluation of Secondary Temperatures

The next decision to be addressed is what the secondary loop temperatures should be. A summary of the issue is reviewed in the following subsection. The next subsection provides the evaluation of the discriminating considerations. Then the conclusion for this decision is summarized in the final subsection on this decision.

3.5.1 Summary of Secondary Temperatures Issue

Key question:

"What is the secondary side T_{hot} and T_{cold}?"

Range of options:

IHX approach temperatures between 25-50°C

Major considerations:

Hydrogen process performance

PCS performance

Component cost

Other discriminators:

NHS feasibility

3.5.2 Assessment of Secondary Temperatures Considerations

The evaluation of the secondary temperatures is primarily a tradeoff of improved plant performance weighed against the cost of increased IHX effectiveness. System performance can improve significantly if heat is delivered to the process or PCS at higher temperature. However, the IHX cost also increases significantly as higher effectiveness is specified.

To some extent, this question is the other side of the reactor outlet temperature question examined in Section 3.1. In the reactor outlet discussion, the approach temperatures in the IHX were assumed as given values. In this discussion, the reactor outlet temperature is assumed as a boundary condition and the incentive for modifying the approach temperature is examined. In a sense, the evaluation of the secondary temperature question is intended to confirm the assumption made in setting the reactor outlet temperature.

With a reactor outlet temperature of 900°C, the PCS and hydrogen plant will operate somewhere in the 800-900°C range. System performance can improve significantly with increased heat delivery temperature in this range. As discussed in Section 3.1, the PCS performance improves modestly within this range. The hydrogen process performance is expected to improve more significantly, so it offers a greater incentive for a reduced approach temperature.

The approach temperature between the primary and secondary fluid depends on the IHX effectiveness. In order to achieve a 50°C approach temperature such that the hot secondary fluid leaving the IHX is at 850°C, an IHX effectiveness of 89% is required. For a 25°C approach, the required effectiveness increases to 94%. A key point which makes this difference very significant is that, for high effectiveness heat exchangers, small improvements in effectiveness typically require large increases in surface area. An increase in effectiveness from 89% to 94% would approximately double the size and cost of an IHX.

Thus, it does not make sense to significantly increase the effectiveness for the PCS loop IHX. The anticipated performance benefit is not that large, but the cost of increasing the effectiveness would be significant for the large tubular IHXs with nitrogen-helium on the secondary. The size of the PCS IHX necessitates controlling cost.

For the hydrogen production IHX, the benefit of increasing the effectiveness and the resulting process heat delivery temperature is greater. Moreover, this IHX is smaller, so the overall cost of improving the effectiveness is not as significant. The small IHX for hydrogen production provides a greater incentive for pursuing high effectiveness.

3.5.3 Secondary Temperatures Issue Conclusion

Based on the assessment in the preceding section, the secondary temperatures will be kept at their current baseline values. This corresponds to an approach temperature of 50°C in the PCS IHX, with a relatively conservative 89% effectiveness, and an approach temperature of 25°C in the hydrogen production IHX, with a more aggressive 94 effectiveness.

The PCS secondary system temperatures are:

- T_{hot} (PCS supply) = 850°C
- T_{cold} (PCS return) = 450°C

This will provide good PCS performance, and it provides a reasonable effectiveness goal for the large IHX.

The temperatures in the High Temperature Heat Transport Loop which carries heat to the hydrogen plant are:

- T_{hot} (Heat transport loop supply) = 875°C
- T_{cold} Heat transport loop return) = 475°C

The hydrogen process performance characteristics encourage higher temperatures, and the more aggressive effectiveness goal is reasonable for the smaller IHX.

The recommended conditions are for the NGNP demonstration plant. An obvious question would be what the recommended conditions should be for a dedicated commercial hydrogen production plant. A detailed answer to that question is not available but one would anticipate a compromise in which reactor, IHX, and hydrogen process design and performance margins are reevaluated to develop the optimized solution. One might anticipate a solution in which the reactor outlet temperature is increased by a few degrees, the IHX effectiveness is slightly above 90%, and the hydrogen process is further optimized, resulting in good overall performance without placing an unacceptable burden on any specific part of the overall system. The fact that pure helium would be used in the high temperature heat transport loop would certainly help. Of course, the first commercial plant would benefit from the additional knowledge obtained in the NGNP development and demonstration program.

3.6 Issue 5 – Evaluation of System Pressure

The final decision to be addressed is the primary and secondary coolant pressures. A summary of the issue is reviewed in the following subsection. The next subsection provides the evaluation of the discriminating considerations. Then the conclusion for this decision is summarized in the final subsection on this decision.

3.6.1 Summary of System Pressure Issue

Key question:

"What are the primary and secondary system pressures?"

Range of options:

Overall range 4.0-8.0 MPa with emphasis on 5.0 MPa, 5.5 MPa, 6.0 MPa, and 6.5 MPa

Major considerations:

Operating cost

Plant cost

Other discriminators:

NHS feasibility

Secondary system performance

3.6.2 Assessment of System Pressure Considerations

In the simplest sense, determination of the primary coolant pressure is an optimization exercise which attempts to balance the vessel cost with the system pumping power. Increasing pressure generally increases vessel cost and reduces pumping power requirements.

The vessel cost includes the actual capital cost of the primary coolant vessels as well as the development and fabrication issues that are either minimized or exaggerated as a function of increasing system pressure. It also includes any operational costs associated with increased surveillance requirements due to higher system pressure.

The main element of the pumping power is the electrical power required to drive the circulator in indirect cycle systems or the shaft power required to drive the compressor in direct Brayton systems. However, several other factors are implicitly included in the pumping power. Systems with higher pumping power also require larger circulators or compressors which increases capital cost. As circulator size increases, technology development and feasibility issues may also arise.

The impact on cycle performance varies differently for different power conversion systems. Optimization of system pressure has a more fundamental effect on direct cycle systems than on indirect systems, although all systems are affected to one degree or another.

For indirect cycle systems, the pressure difference across the IHX can be important in determining the operating stress in the IHX and the corresponding component lifetime. A slight pressure bias across the IHX is preferred to avoid locations that are nearly perfectly balanced and therefore frequently crossing back and forth from tension to compression. The direction of this bias is normally dependent on the optimization of stress in the IHX and on the impact of potential leaks and the transport of contaminants from one fluid stream to the other.

For the NGNP preconceptual design based on the adapted ANTARES HTR concept, the system pressure does not have a significant impact on the fundamental characteristics of the system. Therefore, only a brief qualitative assessment of system pressure has been performed. More detailed optimization calculations will be necessary during the Conceptual Design phase.

The key difference between the reference ANTARES concept and the AREVA NGNP preconceptual design is the decision to use multiple loops in the NGNP. The resulting availability of multiple circulators leads to optimization at a lower primary pressure than the ANTARES. The ANTARES reference primary coolant pressure was 6.0 MPa, and the recommended NGNP pressure is 5.0 MPa. The use of a lower pressure reduces vessel cost and it reduces fabrication difficulties. This is beneficial for 2018 startup at the INL site.

3.6.3 System Pressure Issue Conclusion

The recommended NGNP primary coolant pressure is 5.0 MPa.

The secondary coolant pressure is essentially balanced with the primary circuit. A slight bias (perhaps 0.1 MPa) will be imposed between the two circuits to make the long-term stress regimes more predictable. For the PCS the system pressure would likely be set slightly above the primary pressure in order to minimize stress in the hot end of the IHX and to minimize contamination of the PCS circuit in the event of an IHX leak. For the heat transport loop to the hydrogen plant, the possible presence of contaminants in the circuit due to potential process heat exchanger leaks in the hydrogen plant must be considered. In each case, detailed analysis will be required to optimize the pressure bias for each of the secondary loops.

3.7 Review Overall Consistency of Results

Having completed the initial evaluation of each of the decisions to be made, it is necessary to review the initial results to make sure that in the aggregate the results are reasonable and self-consistent. Since most of the later decisions depend to some extent on the prior decisions, it is important to review them to make sure that the initial decisions still make sense in light of the final outcome. The following questions are addressed with this purpose in mind:

Is the reactor outlet temperature of 900°C still reasonable considering that it indirectly requires an inlet temperature of 500°C?

> Yes. The 900°C hot end temperature provides good performance, and the 500°C cold end temperature can be reasonably accommodated without adversely impacting core flow distributions, etc.

Do the selected parameters and configuration result in a reasonable system?

> Yes, all key elements are feasible, and the level of technical difficulty is reasonably balanced within the system.

Is the recommended concept compatible with 2018 deployment?

> Yes, assuming prompt action is taken on design, development, and procurement.

Does the concept support hydrogen process development?

Yes, the dedicated loop providing heat to the high temperature heat transport loop provides significant flexibility. It provides a way to test alternate components and it can support a range of temperature and operational conditions without adversely impacting electricity generation.

Does the concept support direct commercialization?

Yes, the rugged system design based on near-term technology and reasonable performance requirements provides a good foundation for rapid commercial deployment following initial NGNP demonstration.

The selection of the parallel configuration for PCS and hydrogen process heat loads provides best support for future commercialization, since dedicated commercial plant will have to directly match process and reactor inlet and outlet temperature inderface.

The selected approach allows results in reasonable requirements for key component and a well balanced overall system. The emphasis on maintainable technologies will allow rapid transfer to a commercial environment.

4.0 RESULTS AND CONCLUSIONS

The main results and recommended operating conditions for the NGNP are provided in Section 4.2. First, more general conclusions are provided in Section 4.1.

4.1 General Conclusions

A 900°C reactor outlet temperature provides a good balance between hydrogen production process performance and NHS feasibility

A 500°C reactor inlet temperature provides a reasonable core temperature rise without severe consequences on the rest of the design.

The multiple loop configuration addresses several otherwise difficult development challenges, and leads to designs based on existing experience and near-term technology for all key components.

The recommended concept is well suited to future commercial hydrogen plants. Two loops with tubular IHXs are expected to be adequate for transferring the full reactor output to a hydrogen production facility, potentially resulting in reduced system cost compared to a single loop compact heat exchanger configuration. The selected configuration is also most easily adapted to future advanced heat transport systems such as the use of molten salt.

4.2 Summary of Results and Recommendations

The following parameters are recommended for use in the NGNP preconceptual design:

- Reactor outlet temperature 900°C
- Reactor inlet temperature 500°C
- System configuration H2 and PCS in parallel
- Number of loops 4 loops
  3 with tubular IHXs for PCS
  1 with compact IHX for H2
- Secondary temperatures 450-850°C for PCS (50°C approach)
  475-875°C for H2 (25°C approach)
- System pressure 5.0 MPa

It is further recommended that these parameters be reconfirmed early in the conceptual design phase in order to take into consideration the final results of the preconceptual design studies and any modification of mission requirements at the beginning of the next project phase.

5.0 REFERENCES

1. 51-9041508-000, NGNP Design Baseline, April, 2007.
2. AREVA NGNP Preconceptual Design Work Plan, 2006.
3. INL, NGNP Technical Functions and Requirements, 2003.
4. 51-9039141-000, NGNP System Requirements Manual, March, 2007.
5. K.-F. Knoche and J.E. Funk, “Entropy production, efficiency, and economics in the thermochemical generation of synthetic fuels. I. The hybrid sulfuric acid process", *Int. J. Hydrogen Energy*, **2**, pp. 377-385 (1977).
6. K.-D. Jung et al., “Decomposition of Sulfuric Acid to Produce Sulfur Dioxide and Oxygen in IS cycle”, paper 230c, AIChE Annual Meeting, San Francisco (2006).
7. A.J. Goodjohn and R.N. Quade, “The Cogeneration Potential of the High-Temperature Gas-Cooled Reactor,” GA-A16706, GA Technologies (March 1982).

APPENDIX A: IMPACT OF HIGH TEMPERATURE HEAT TRANSPORT STRATEGY ON PRIMARY AND SECONDARY CYCLE CONCEPT

A.1 High Temperature Heat Transport Alternatives

The Primary and Secondary Cycle Concept Study has assumed that an intermediate heat transport loop using high pressure helium would be used to transport the heat to be provided to the hydrogen production process facilities. This is an effective means of high temperature heat transport which presents the most straightforward interface with the NHS for service in the 850-900°C temperature range with minimal basic technology development.

The study also assumes an indirect cycle CCGT using a nitrogen-helium gas mixture as adapted from the reference ANTARES design as the basis for the PCS.

It is important to keep in mind that several alternatives to these assumptions exist which may have significant advantages depending on the specific plant mission requirements and design constraints. The evaluation of these alternatives falls most clearly under the High Temperature Heat Transport Special Study and the Power Conversion System Special Study, and it is clearly outside the scope of the Primary and Secondary Cycle Concept Study. Nonetheless, these alternative approaches could have significant impacts on the decisions made within this study if they had been adopted as the reference assumptions at the beginning of the study. Moreover, these alternatives may present important opportunities for future technology deployment, and it is not the intent of the study to discourage them from being further evaluated.

Some of the most obvious alternatives to high pressure helium include the following:

- Molten salt
- Liquid metal
- Chemical energy transfer
- High temperature steam
- Electrical energy supply
- Hybrid systems

A.2 Effectiveness of Heat Transport Alternatives

- Molten salt – Has the potential to significantly reduce pumping power and piping cost by allowing low pressure operation. It has good high temperature capability.
- Liquid metal – Similar characteristics as molten salt, although high temperature capability is not quite as good.
- Chemical energy transfer – Proposed in the past as a means to transfer the energy at ambient temperature. NHS heat exchanger drives high temperature endothermic reaction and products are cooled regeneratively prior to transport to user facility. Energy is released at user facility in inverse exothermic reaction. Systems have been evaluated for specific applications in the past. Substantial component and system development work would be required to deploy a system for current applications.

- High temperature steam – This is the most mature thermal energy transfer technology. High temperature steam is routinely transported over significant distances, potentially alleviating the challenge of locating the NHS and hydrogen production facility in close proximity to one another. However, the maximum temperatures achievable are not as high as high pressure helium and molten salt (perhaps 600°C). Use of high pressure steam for thermochemical hydrogen production would require modification of currently envisioned systems or adoption of processes with lower temperature requirements than S-I.

- Electrical energy supply – The most convenient form of energy supply is electrical heating. This simplifies the design of both the process equipment and the NHS interface. The key drawback is the substantial loss in efficiency due to the electricity generation process. Using HTRs offers the potential to increase this efficiency, but it is still a significant loss compared to the ideal of direct thermal energy transfer.

- Hybrid systems – The most attractive approach may ultimately be a system that combines elements of more that one of the above concepts. Mature systems will likely take this route to minimize some of the difficulties associated with individual approaches and achieve the best optimized system. This is one of the areas of AREVA's internally funded research.

A.3 Impact of Heat Transport Alternatives on Nuclear Heat Source

- Molten salt –Trace heating of all piping, heat exchangers, and components would be required, including the IHX. Significant material compatibility questions remain to be resolved. The issue of pressure balance within the IHX could be a serious problem. The issue of molten salt leaking into the primary system must also be resolved.

  Molten salt is not viewed as ready for deployment in the NGNP and probably not in the initial wave of commercial plants.

- Liquid metal – Similar concerns as molten salt. Material compatibility data may be more widely available, but remaining issues still preclude near term deployment.

- Chemical energy transfer – Significant development and analysis would be required to assess the impact of such a system for NGNP. Significant effort would be anticipated in developing the required heat exchanger/chemical reactor for interfacing a proposed application with the primary circuit of current reactor concepts. Safety implications of large scale chemical reactions on secondary side of NHS heat exchanger would have to be evaluated with the regulator.

  Chemical energy transfer systems are perhaps the least ready for deployment due to the required development.

- High temperature steam – Production of high temperature steam using HTRs is well established technology, having been the focus of numerous design studies and having been demonstrated successfully in several previous plants. In general, NHS design constraints are relaxed significantly, because reactor operating temperatures are somewhat lower, and because the IHX is replace by a steam generator which is more robust. The issue of water ingress would have to be reintroduced into the safety analysis space, but existing approaches used to resolve this issue in the past would be expected to be adequate. (Note that the water ingress issue is not the serious circulator water bearing issue that plagued Fort St. Vrain.)

High temperature steam supply is the most readily available technology and could accelerate deployment of the NHS.

- Electrical energy supply – The impact of this approach on the NHS depends on which generating technology is employed. If a Rankine system is used, the issues are almost identical to the high temperature steam option. If a Brayton or CCGT is used, then other considerations are introduced.

- Hybrid systems – The impact on the NHS obviously depends on which options are adapted in the hybrid approach. If a combination of high temperature steam and electricity were selected, then there would be no impact on the NHS, since the NHS product would be steam in either case.

APPENDIX C: RESEARCH AND DEVELOPMENT

C.0 RESEARCH AND DEVELOPMENT

C.1 R&D Survey Participants

The R&D needs were identified by surveying the subject matter experts as described in Section 19. A total of 46 survey forms were submitted by these subject matter experts (Table C.1). Dominique Hittner of AREVA provided a general review of the survey forms and added cost/schedule estimates and Importance/Knowledge inputs for some of the survey results.

Table C-1: VHTGR R&D Survey Subject Matter Experts

Area	Name	Organization
Nuclear Heat Source / Nuclear Island	Bernard Riou	AREVA
Power Conversion System		MHI
Heat Transfer Loop	Bernard Riou	AREVA
Support Systems	Bernard Riou /Eric Breuil/John Mayer	AREVA
Support systems (NI & BOP Bldgs)	Bernard Riou / Bob Lenyk	AREVA / Burns & Roe
Fuel Supply and Qualification	John Mayer / Jeff Halfinger	AREVA / BWXT
Materials (Metallic)	Sophie Dubiez Le Goff	AREVA
Materials (Ceramics & Graphite)	Odile Gelineau	AREVA
Licensing and Permitting	Bill Szymczak / Farshid Shahrokhi	AREVA
Computer Codes	Denis Verrier	AREVA

C.2 R&D Survey Forms

The individual survey forms are provided in this section by the NGNP project WBS number. (Additional, lower level WBS numbers have been added by AREVA.)

C.2.1 WBS C Q 10 2 22 NHS 2.2.1 Fuel
WBS C Q 10 2 22 NHS 2.2.1.1 Fuel Development

VHTGR H/W Survey Form

NGNP VHTGR H/W R&D Survey Form ***(Design Baseline is listed in*** *Reference* ***)***

WBS # (Reference)	**2.2.1.1.1**	**WBS Title**	**Kernel materials**
Subject Matter Expert Name		**Email**	**Phone**
Organization			

Current Technology Readiness Level
Identify TRL, including rationale

TRL	**4**	*TRL Definitions are listed in at Reference* *If can't decide please describe in the rationale block; if more space is required please insert "rows" or adjust "height"*
Rationale & Assumptions	**Historical programs have demonstrated the feasibility of TRISO fuels containing either UCO or UO2 kernels in gas reactors at reasonable performance temperatures and burnups. Production capabilities do not exist in US infrastructure. Irradiation testing ongoing of preliminary UCO kernels (contained in particles) produced in pilot-scaled facility.**	

R&D Needs
Identify and characterize the needs for R&D work to resolve critical issues affecting design, fabrication, testing and operation of the VHTGR plant

Importance and Knowledge definitions are listed in Reference; Estimated Cost or Schedule in a range (e.g., $5 M to $7M or > 12 months) is acceptable
If more rows/space are needs please insert "rows" and/or adjust "height"

1 **Development of advanced carbon source for UCO kernel production ($5M to $10M to produce carbon source material and test materials in pilot-facility fabricating UCO kernels).**

Which Program Phase Will This R&D Support? (please check one)

Design	Construction	Initial Operations	Operate Commercially
		x	x

Importance (H/M/L) *(Reference)*	Knowledge (K/P/U) *(Reference)*	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Fallback Option *(Reference for definition)*
M	U	$5M-$10M	12 mo		existing facilities at BWXT	Utilize best-available surface modified carbon tested under existing programs.

2

Which Program Phase Will This R&D Support? (please check one)

Design	Construction	Initial Operations	Operate Commercially

Importance (H/M/L) *(Reference)*	Knowledge (K/P/U) *(Reference)*	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Fallback Option *(Reference for definition)*

Kernel Manufacturing (2.2.1.1.2)

VHTGR H/W Survey Form

NGNP VHTGR H/W R&D Survey Form ***(Design Baseline is listed in Reference)***

WBS # (Reference)	**2.2.1.1.2**	**WBS Title**	**Kernel Manufacturing**
Subject Matter Expert Name		**Email**	**Phone**
Organization			

Current Technology Readiness Level
Identify TRL, including rationale

TRL	**4**	*TRL Definitions are listed in at Reference* *If can't decide please describe in the rationale block; if more space is required please insert "rows" or adjust "height"*
Rationale & Assumptions	**Historical programs have demonstrated the feasibility of TRISO fuels containing either UCO or UO2 kernels in gas reactors at reasonable performance temperatures and burnups. Production capabilities do not exist in US infrastructure. Irradiation testing ongoing of preliminary UCO kernels (contained in particles) produced in pilot-scaled facility.**	

R&D Needs
Identify and characterize the needs for R&D work to resolve critical issues affecting design, fabrication, testing and operation of the VHTGR plant

Importance and Knowledge definitions are listed in Reference; Estimated Cost or Schedule in a range (e.g., $5 M to $7M or > 12 months) is acceptable
If more rows/space are needs please insert "rows" and/or adjust "height"

1

Development of advanced kernel wash and dry system to cost effectively increase throughput of kernel line with no degradation in kernel quality.

Development of enhanced sintering for either UCO (large fluidized bed sintering) or UO2 (static bed sintering) with a focus on increased throughput and reduced cost.

Which Program Phase Will This R&D Support? (please check one)

Design	Construction	Initial Operations	Operate Commercially
x		x	x

Importance (H/M/L) *(Reference)*	Knowledge (K/P/U) *(Reference)*	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Fallback Option *(Reference for definition)*
H	**P**	**$15M-$20M**	**24 mo**		**existing, modified facilities at BWXT site**	**Utilize current high cost equipment and processes that have been demonstrated to be effective.**

2

Which Program Phase Will This R&D Support? (please check one)

Design	Construction	Initial Operations	Operate Commercially

Importance (H/M/L) *(Reference)*	Knowledge (K/P/U) *(Reference)*	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Fallback Option *(Reference for definition)*

Coating Materials (2.2.1.1.3)

VHTGR H/W Survey Form

NGNP VHTGR H/W R&D Survey Form (Design Baseline is listed in Reference)					
WBS # (Reference)	2.2.1.1.3	WBS Title	Coating Materials		
Subject Matter Expert Name		Email		Phone	
Organization					

Current Technology Readiness Level
Identify TRL, including rationale

TRL	7	*TRL Definitions are listed in at Reference* *If can't decide please describe in the rationale block; if more space is required please insert "rows" or adjust "height"*
Rationale & Assumptions	**Historical programs have demonstrated the feasibility of TRISO coated fuels containing either UCO or UO2 kernels in gas reactors at reasonable performance temperatures and burnups. Production capabilities do not exist in US infrastructure. Irradiation testing ongoing of preliminary TRISO particles produced in laboratory-scaled coating furnace.**	

R&D Needs
Identify and characterize the needs for R&D work to resolve critical issues affecting design, fabrication, testing and operation of the VHTGR plant

Importance and Knowledge definitions are listed in Reference; Estimated Cost or Schedule in a range (e.g., $5 M to $7M or > 12 months) is acceptable
If more rows/space are needs please insert "rows" and/or adjust "height"

1	**No materials development needs identified for the coating materials.** **R&D need of coating materials qualification has been included in WBS 2.2.1.1.5 Compact Materials and 2.2.1.1.6 Compact Manufacturing**

Which Program Phase Will This R&D Support? (please check one)

Design	Construction	Initial Operations	Operate Commercially

Importance (H/M/L) (Reference)	Knowledge (K/P/U) (Reference)	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Fallback Option (Reference for definition)

2	

Which Program Phase Will This R&D Support? (please check one)

Design	Construction	Initial Operations	Operate Commercially

Importance (H/M/L) (Reference)	Knowledge (K/P/U) (Reference)	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Fallback Option (Reference for definition)

Coating Manufacturing (2.2.1.1.4)

VHTGR H/W Survey Form							
NGNP VHTGR H/W R&D Survey Form *(Design Baseline is listed in Reference)*							
WBS # (Reference)	2.2.1.1.4	WBS Title	Coating Manufacturing				
Subject Matter Expert Name		Email		Phone			
Organization							
Current Technology Readiness Level Identify TRL, including rationale							
TRL	4	*TRL Definitions are listed in at Reference* *If can't decide please describe in the rationale block; if more space is required please insert "rows" or adjust "height"*					
Rationale & Assumptions	**Historical programs have demonstrated the feasibility of TRISO coated fuels containing either UCO or UO2 kernels in gas reactors at reasonable performance temperatures and burnups. Production capabilities do not exist in US infrastructure. Irradiation testing ongoing of preliminary TRISO particles produced in laboratory-scaled coating furnace.**						
R&D Needs Identify and characterize the needs for R&D work to resolve critical issues affecting design, fabrication, testing and operation of the VHTGR plant							
Importance and Knowledge definitions are listed in Reference; Estimated Cost or Schedule in a range (e.g., $5 M to $7M or > 12 months) is acceptable *If more rows/space are needs please insert "rows" and/or adjust "height"*							
1	**Investigate largest coating batches size capable in existing 6" coating retort and determine economic feasibility of using a 6" retort for production. Acceptability of coatings should initially be based on physical characteristics of the coatings after manufacture. Should a larger coater be required, plan on implementing the R&D of that coater as part of the facility expansion for production.**						
Which Program Phase Will This R&D Support? (please check one)							
Design	**Construction**	**Initial Operations**	**Operate Commercially**				
x		x	x				
Importance (H/M/L) *(Reference)*	**Knowledge (K/P/U)** *(Reference)*	**Estimated Cost ($M)**	**Estimated Schedule (Months)**	**Predecessor R&D, if any?**	**Facility Availability (exiting, modified, new) (which existing facilities?)**	**Fallback Option** *(Reference for definition)*	
H	**P**	**$5M-20M (if new furnace is needed)**	**6mo-24mo**		**Existing 6" furnace at BWXT**	**Use parameters and batch size developed by the government and accept the economic impact**	
2							
Which Program Phase Will This R&D Support? (please check one)							
Design	**Construction**	**Initial Operations**	**Operate Commercially**				
Importance (H/M/L) *(Reference)*	**Knowledge (K/P/U)** *(Reference)*	**Estimated Cost ($M)**	**Estimated Schedule (Months)**	**Predecessor R&D, if any?**	**Facility Availability (exiting, modified, new) (which existing facilities?)**	**Fallback Option** *(Reference for definition)*	

Compact Materials (2.2.1.1.5)

VHTGR H/W Survey Form							
NGNP VHTGR H/W R&D Survey Form *(Design Baseline is listed in Reference)*							
WBS # (Reference)		2.2.1.1.5		WBS Title	Compact Materials		
Subject Matter Expert Name				Email		Phone	
Organization							
Current Technology Readiness Level Identify TRL, including rationale							
TRL	3			*TRL Definitions are listed in at Reference* *If can't decide please describe in the rationale block; if more space is required please insert "rows" or adjust "height"*			
Rationale & Assumptions	**Compact fabrication using thermosetting resins has been developed and demonstrated on a laboratory scale, however, currently-available materials have not been irradiated and performance under relevant environment demonstrated.**						
R&D Needs Identify and characterize the needs for R&D work to resolve critical issues affecting design, fabrication, testing and operation of the VHTGR plant							
Importance and Knowledge definitions are listed in Reference; Estimated Cost or Schedule in a range (e.g., $5 M to $7M or > 12 months) is acceptable *If more rows/space are needs please insert "rows" and/or adjust "height"*							
1	**Selection of graphitic matrix, resin, etc. to produce thermosetting compacts. Performance of compacts must be demonstrated under normal and off-normal accident conditions.**						
Which Program Phase Will This R&D Support? (please check one)							
Design		**Construction**		**Initial Operations**		**Operate Commercially**	
x				x		x	
Importance (H/M/L) *(Reference)*	**Knowledge (K/P/U)** *(Reference)*	**Estimated Cost ($M)**	**Estimated Schedule (Months)**	**Predecessor R&D, if any?**	**Facility Availability (exiting, modified, new) (which existing facilities?)**	**Fallback Option** *(Reference for definition)*	
H	**P**	**$60**	**36**			**none**	
2							
Which Program Phase Will This R&D Support? (please check one)							
Design		**Construction**		**Initial Operations**		**Operate Commercially**	
Importance (H/M/L) *(Reference)*	**Knowledge (K/P/U)** *(Reference)*	**Estimated Cost ($M)**	**Estimated Schedule (Months)**	**Predecessor R&D, if any?**	**Facility Availability (exiting, modified, new) (which existing facilities?)**	**Fallback Option** *(Reference for definition)*	

Compact Manufacturing (2.2.1.1.6)

VHTGR H/W Survey Form							
NGNP VHTGR H/W R&D Survey Form *(Design Baseline is listed in Reference)*							
WBS # (Reference)		2.2.1.1.6		WBS Title	Compact Manufacture		
Subject Matter Expert Name					Email		Phone
Organization							
Current Technology Readiness Level Identify TRL, including rationale							
TRL	3			*TRL Definitions are listed in at Reference* *If can't decide please describe in the rationale block; if more space is required please insert "rows" or adjust "height"*			
Rationale & Assumptions	**Compact fabrication using thermosetting resins has been developed and demonstrated on a laboratory scale, however, currently-available materials and processes have not been irradiated and performance under relevant environment demonstrated.**						
R&D Needs Identify and characterize the needs for R&D work to resolve critical issues affecting design, fabrication, testing and operation of the VHTGR plant							
Importance and Knowledge definitions are listed in Reference; Estimated Cost or Schedule in a range (e.g., $5 M to $7M or > 12 months) is acceptable *If more rows/space are needs please insert "rows" and/or adjust "height"*							
1	**Establish compact manufacturing capabilities in the US based on the AREVA process. Develop (or confirm) compact pressures and temperatures to minimize fuel damage. Develop heat treat process to ensure complete graphitization of the matrix material.** **Perform irradiation tests on compacts to demonstrate performance for nominal and off-nominal operating conditions.**						
Which Program Phase Will This R&D Support? (please check one)							
Design		**Construction**		**Initial Operations**		**Operate Commercially**	
x				x		x	
Importance (H/M/L) (Reference)	**Knowledge (K/P/U)** (Reference)	**Estimated Cost ($M)**	**Estimated Schedule (Months)**	**Predecessor R&D, if any?**	**Facility Availability (exiting, modified, new) (which existing facilities?)**	**Fallback Option** (Reference for definition)	
H	P	$40	36		AREVA facilities in France. Expansion of BWXT fuel line for compacts is recommended		
2							
Which Program Phase Will This R&D Support? (please check one)							
Design		**Construction**		**Initial Operations**		**Operate Commercially**	
Importance (H/M/L) (Reference)	**Knowledge (K/P/U)** (Reference)	**Estimated Cost ($M)**	**Estimated Schedule (Months)**	**Predecessor R&D, if any?**	**Facility Availability (exiting, modified, new) (which existing facilities?)**	**Fallback Option** (Reference for definition)	

Quality Control Methods (2.2.1.1.7)

VHTGR H/W Survey Form							
NGNP VHTGR H/W R&D Survey Form *(Design Baseline is listed in Reference)*							
WBS # (Reference)	2.2.1.1.7		WBS Title	Quality Control Methods			
Subject Matter Expert Name				Email		Phone	
Organization							
Current Technology Readiness Level Identify TRL, including rationale							
TRL	6		*TRL Definitions are listed in at Reference* *If can't decide please describe in the rationale block; if more space is required please insert "rows" or adjust "height"*				
Rationale & Assumptions	Most quality control methods are available for pilot-scale fuel production.						
R&D Needs Identify and characterize the needs for R&D work to resolve critical issues affecting design, fabrication, testing and operation of the VHTGR plant							
Importance and Knowledge definitions are listed in Reference; Estimated Cost or Schedule in a range (e.g., $5 M to $7M or > 12 months) is acceptable *If more rows/space are needs please insert "rows" and/or adjust "height"*							
1	Development of highly reliable instrumentation and data acquisition software will be needed to ensure fuel particle quality is built into the fuel. Capturing of essential data for fuel certification will be necessary.						
Which Program Phase Will This R&D Support? (please check one)							
Design		Construction		Initial Operations		Operate Commercially	
x				x		x	
Importance (H/M/L) (Reference)	Knowledge (K/P/U) (Reference)	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Fallback Option (Reference for definition)	
M	K	$2	18				
2							
Which Program Phase Will This R&D Support? (please check one)							
Design		Construction		Initial Operations		Operate Commercially	
Importance (H/M/L) (Reference)	Knowledge (K/P/U) (Reference)	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Fallback Option (Reference for definition)	

Inspection Techniques (2.2.1.1.8)

VHTGR H/W Survey Form

NGNP VHTGR H/W R&D Survey Form *(Design Baseline is listed in Reference)*

WBS # (Reference)	2.2.1.1.8	WBS Title	Inspection Techniques		
Subject Matter Expert Name		Email		Phone	
Organization					

Current Technology Readiness Level
Identify TRL, including rationale

TRL	4	*TRL Definitions are listed in at Reference* *If can't decide please describe in the rationale block; if more space is required please insert "rows" or adjust "height"*
Rationale & Assumptions	Several inspection techniques are available for fuel kernels, particles, and compacts. However, a strong correlation between as-fabricated and inspected particles and compacts and irradiation performance has not been shown in all cases.	

R&D Needs
Identify and characterize the needs for R&D work to resolve critical issues affecting design, fabrication, testing and operation of the VHTGR plant

Importance and Knowledge definitions are listed in Reference; Estimated Cost or Schedule in a range (e.g., $5 M to $7M or > 12 months) is acceptable
If more rows/space are needs please insert "rows" and/or adjust "height"

1	Develop QC inspection techniques that directly relate to irradiation performance. Develop techniques for large-scale production capabilities that minimize the quantity of materials that require destructive evaluation to ensure statistically acceptable fuel is produced. Techniques to be investigated could be: micro focus x-ray of particles (dimensional inspection of particle layers), mercury porosymetry (buffer density), sink-float (IPyC, SiC, and OPyC density), anisotropy measurements of the IPyC and OPyC layers, leach-burn-leach test or weak irradiation techniques (particl leak tightness), etc. Irradiation testing of the compacts to attempt to relate as-measured attributes actually correlate to performance would be necessary to ensure the correct attributes are being measured and characterized.

Which Program Phase Will This R&D Support? (please check one)

Design	Construction	Initial Operations	Operate Commercially
x		x	x

Importance (H/M/L) (Reference)	Knowledge (K/P/U) (Reference)	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Fallback Option (Reference for definition)
H	P/U	$25	24			

2	

Which Program Phase Will This R&D Support? (please check one)

Design	Construction	Initial Operations	Operate Commercially

Importance (H/M/L) (Reference)	Knowledge (K/P/U) (Reference)	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Fallback Option (Reference for definition)

Fuel Mass Production (2.2.1.1.9)

VHTGR H/W Survey Form			
NGNP VHTGR H/W R&D Survey Form *(Design Baseline is listed in Reference)*			
WBS # (Reference)	2.2.1.1.9	WBS Title	Fuel Mass Production

Subject Matter Expert Name		Email		Phone	
Organization					

Current Technology Readiness Level Identify TRL, including rationale		
TRL	3	*TRL Definitions are listed in at Reference* *If can't decide please describe in the rationale block; if more space is required please insert "rows" or adjust "height"*
Rationale & Assumptions	**Many areas of the fuel fabrication process have been demonstrated on a pilot-scale. However, some chemical processing areas or the process will require significant scale-up to meet production demands. This scale-up is not expected to be linear and product quality must be demonstrated on the larger scale.**	

R&D Needs Identify and characterize the needs for R&D work to resolve critical issues affecting design, fabrication, testing and operation of the VHTGR plant	
Importance and Knowledge definitions are listed in Reference; Estimated Cost or Schedule in a range (e.g., $5 M to $7M or > 12 months) is acceptable *If more rows/space are needs please insert "rows" and/or adjust "height"*	
1	**R&D should focus on areas where product uniformity and quality are most at jeopardy. Initial R&D should focus on kernel wash & dry, sintering, coating (assuming larger than 6" coater required), compact matrix formulation, and compact fabrication. Many QC techniques need to be developed with mass production in mind (reference 2.2.1.1.8). Irradiation testing will be required to confirm fuel performance matches performance from the laboratory/pilot facilities.**

Which Program Phase Will This R&D Support? (please check one)			
Design	**Construction**	**Initial Operations**	**Operate Commercially**
x		x	x

Importance (H/M/L) *(Reference)*	Knowledge (K/P/U) *(Reference)*	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Fallback Option *(Reference for definition)*
H	P	$30	30		Modify existing facilities at BWXT to develop larger-scaled production	

2	

Which Program Phase Will This R&D Support? (please check one)			
Design	**Construction**	**Initial Operations**	**Operate Commercially**

Importance (H/M/L) *(Reference)*	Knowledge (K/P/U) *(Reference)*	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Fallback Option *(Reference for definition)*

C.2.2 WBS C Q 10 2 22 NHS 2.2.2 Materials Development and Qualification (2.2.2)

VHTGR H/W Survey Form

NGNP VHTGR H/W R&D Survey Form *(Design Baseline is listed in Reference)*

WBS # (Reference)	**006 2 2 2**	**WBS Title**	**Materials development and qualification**
Subject Matter Expert Name	**Bernard Riou**	**Email** bernard.riou@areva.com	**Phone** 434-832-4255
Organization	**AREVA NP**		

Current Technology Readiness Level
Identify TRL, including rationale

TRL	**5**	*TRL Definitions are listed in at Reference* *If can't decide please describe in the rationale block; if more space is required please insert "rows" or adjust "height"*
Rationale & Assumptions	**The HTR design relies on contact conditions between different materials (metal to metal, graphite to ceramics, ceramics to metal, etc.) and R&D actions have to be performed to assess the contact conditions to avoid unexpected situations (bonding, wear, etc).** **As an example, the core support interface to reactor vessel interface currently is assumed to be a sliding interface. R&D actions are required to make sure that the helium environment (together with the contact pressure) is not likely to create a bonding effect between the alloy 800 and the 9 CR 1Mo material which consequences for the design may not be acceptable.**	

R&D Needs
Identify and characterize the needs for R&D work to resolve critical issues affecting design, fabrication, testing and operation of the VHTGR plant

Importance and Knowledge definitions are listed in Reference; Estimated Cost or Schedule in a range (e.g., $5 M to $7M or > 12 months) is acceptable
If more rows/space are needs please insert "rows" and/or adjust "height"

1 **• Perform tribology tests on expected couples of materials in representative HTR conditions.**
Note: this type of tests requires dedicated facilities

Which Program Phase Will This R&D Support? (please check one)

Design	Construction	Initial Operations	Operate Commercially
x			

Importance (H/M/L) *(Reference)*	Knowledge (K/P/U) *(Reference)*	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Fallback Option *(Reference for definition)*
M	P	0.5	18		Yes in AREVA NP and CEA	

2

Which Program Phase Will This R&D Support? (please check one)

Design	Construction	Initial Operations	Operate Commercially

Importance (H/M/L) *(Reference)*	Knowledge (K/P/U) *(Reference)*	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Fallback Option *(Reference for definition)*

WBS C Q 10 2 22 NHS 2.2.2.1 Metallic Materials

Vessel Materials (High Temperature Steel) (2.2.2.1.1)

VHTGR H/W Survey Form

NGNP VHTGR H/W R&D Survey Form *(Design Baseline is listed in Reference)*

WBS # (Reference)	006 2 2 2 1 1	WBS Title	Vessel materials
Subject Matter Expert Name	Sophie Dubiez Le Goff	Email	sophie.dubiez-legoff@areva.com
Phone	33-47274 7282		
Organization	AREVA NP		

Current Technology Readiness Level
Identify TRL, including rationale

TRL	6	*TRL Definitions are listed in at Reference* *If can't decide please describe in the rationale block; if more space is required please insert "rows" or adjust "height"*
Rationale & Assumptions	9%Cr steel has been selected for the vessel system. This material is already used in conventional power plants and is also supported by significant R&D test results from past Fast Reactors R&D programs. An R&D program has already been launched in the context of HTR Antares activities to complete the required input data for the final selection and the qualification program.	

R&D Needs
Identify and characterize the needs for R&D work to resolve critical issues affecting design, fabrication, testing and operation of the VHTGR plant

Importance and Knowledge definitions are listed in Reference; Estimated Cost or Schedule in a range (e.g., $5 M to $7M or > 12 months) is acceptable
If more rows/space are needs please insert "rows" and/or adjust "height"

1
- Mechanical properties on heavy section products (base and weld metal)
- Effect of aging
- Effect of irradiation
- Corrosion in helium environment
- Weldability
- Emissivity
- Negligible creep conditions
- Creep fatigue
- A specific test program on representative plates and forging (including welded joints) will be required for the component qualification

Which Program Phase Will This R&D Support? (please check one)

Design	Construction	Initial Operations	Operate Commercially
x	x		

Importance (H/M/L) (Reference)	Knowledge (K/P/U) (Reference)	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Fallback Option (Reference for definition)
H	P	4	72			

2

Which Program Phase Will This R&D Support? (please check one)

Design	Construction	Initial Operations	Operate Commercially

Importance (H/M/L) (Reference)	Knowledge (K/P/U) (Reference)	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Fallback Option (Reference for definition)

Reactor Internals (2.2.2.1.2)

VHTGR H/W Survey Form

NGNP VHTGR H/W R&D Survey Form *(Design Baseline is listed in Reference)*

WBS # (Reference)	006.2.2.2.1.2	WBS Title	Internal materials
Subject Matter Expert Name	Sophie Dubiez Le Goff	Email	sophie.dubiez-legoff@areva.com
Phone	33-47274 7282		
Organization	AREVA NP		

Current Technology Readiness Level
Identify TRL, including rationale

TRL	8	*TRL Definitions are listed in at Reference* *If can't decide please describe in the rationale block; if more space is required please insert "rows" or adjust "height"*
Rationale & Assumptions	In view of past experience in gas cooled reactor, alloy 800H is a prime candidate for metallic internals operating in cold helium. Efforts are moreover in progress to extend its coverage up to 850°C in ASME III-NH. Modified 9Cr1Mo is also a candidate only if the temperature is kept below 750°C for any situation. An R&D program has already been launched in the context of HTR Antares activities to complete the required input data for the final selection and the qualification program.	

R&D Needs
Identify and characterize the needs for R&D work to resolve critical issues affecting design, fabrication, testing and operation of the VHTGR plant

Importance and Knowledge definitions are listed in Reference; Estimated Cost or Schedule in a range (e.g., $5 M to $7M or > 12 months) is acceptable
If more rows/space are needs please insert "rows" and/or adjust "height"

1	It concerns for both alloys: - emissivity measurement under likely representative state of surface (as machined and oxidized after machining) - corrosion behavior under representative primary helium environment For extension of 800H coverage in ASME III-NH the following items are needed : - long term tests at temperature higher than 760°C - tensile tests at temperature higher than 870°C - extension of allowables to cover 60 years lifetime. Needs for mod 9Cr1Mo are already covered in the R&D needs for the vessel system. Needs for metallic materials in operation above 850°C are covered in the R&D needs for IHX materials.

Which Program Phase Will This R&D Support? (please check one)

Design	Construction	Initial Operations	Operate Commercially
x			

Importance (H/M/L) (Reference)	Knowledge (K/P/U) (Reference)	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Fallback Option (Reference for definition)
M	K	0.5	24			

2	

Which Program Phase Will This R&D Support? (please check one)

Design	Construction	Initial Operations	Operate Commercially

Importance (H/M/L) (Reference)	Knowledge (K/P/U) (Reference)	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Fallback Option (Reference for definition)

IHX (2.2.2.1.3)

VHTGR H/W Survey Form

NGNP VHTGR H/W R&D Survey Form *(Design Baseline is listed in Reference)*

WBS # (Reference)	006.2.2.2.1.3	WBS Title	IHX materials
Subject Matter Expert Name	Sophie Dubiez Le Goff	Email	sophie.dubiez-legoff@areva.com
Phone	33-47274 7282		
Organization	AREVA NP		

Current Technology Readiness Level
Identify TRL, including rationale

TRL	6	*TRL Definitions are listed in at Reference* *If can't decide please describe in the rationale block; if more space is required please insert "rows" or adjust "height"*
Rationale & Assumptions	**Two available conventional nickel-base alloys (617 and 230) have been selected as structural materials for the heat exchanger : 617 (NiCr22Co12Mo), which has been widely studied in the early 80's for HTR application and 230 (NiCr22W14), which has been developed more recently but it exhibits better corrosion resistance.** **An extensive research program has been launched in France within the framework of the ANTARES program to evaluate mechanical properties, thermal stability and corrosion resistance in the temperature range [700°C – 1000°C] over long time.**	

R&D Needs
Identify and characterize the needs for R&D work to resolve critical issues affecting design, fabrication, testing and operation of the VHTGR plant

Importance and Knowledge definitions are listed in Reference; Estimated Cost or Schedule in a range (e.g., $5 M to $7M or > 12 months) is acceptable
If more rows/space are needs please insert "rows" and/or adjust "height"

1	**The program addresses the following issues:** **- Baseline mechanical property data, including creep-fatigue data,** **- Long-term thermal stability,** **- Effect of helium coolant chemistry on material degradation,** **- Effect of 80%nitrogen-20%helium mixture on material degradation,** **- Corrosion effects on mechanical properties.**

Which Program Phase Will This R&D Support? (please check one)

Design	Construction	Initial Operations	Operate Commercially
x			

Importance (H/M/L) (Reference)	Knowledge (K/P/U) (Reference)	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Fallback Option (Reference for definition)
H	P	4	30			

2	

Which Program Phase Will This R&D Support? (please check one)

Design	Construction	Initial Operations	Operate Commercially

Importance (H/M/L) (Reference)	Knowledge (K/P/U) (Reference)	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Fallback Option (Reference for definition)

WBS C Q 10 2 22 NHS 2.2.2.2 Ceramics

Reactor Internals (2.2.2.2.2-1 and 2.2.2.2.2-2)

VHTGR H/W Survey Form

NGNP VHTGR H/W R&D Survey Form *(Design Baseline is listed in Reference)*

WBS # (Reference)	006.2.2.2.2.2	WBS Title	Ceramics for reactor internals		
Subject Matter Expert Name	Odile Gelineau	Email	odile.gelineau@areva.com	Phone	33 472747226
Organization	AREVA NP				

Current Technology Readiness Level
Identify TRL, including rationale

TRL	2	*TRL Definitions are listed in at Reference* *If can't decide please describe in the rationale block; if more space is required please insert "rows" or adjust "height"*
Rationale & Assumptions	No structural structures made of composites were used for the past HTRs, neither for other concepts of reactors. The use of composites is interesting because of their high resistance under high or very high temperature. A R&D program has been launched in the frame of Antares to confirm the possible use of such material inside the primary circuit.	

R&D Needs
Identify and characterize the needs for R&D work to resolve critical issues affecting design, fabrication, testing and operation of the VHTGR plant

Importance and Knowledge definitions are listed in Reference; Estimated Cost or Schedule in a range (e.g., $5 M to $7M or > 12 months) is acceptable
If more rows/space are needs please insert "rows" and/or adjust "height"

1	Thermal-physical properties (K, CTE, Cp) Mechanical properties including multiaxial strength Fracture properties Fatigue properties Behavior under oxidized atmosphere and oxidation effects on properties Codification Materials envisioned so far are C/C or C/SiC composites.

Which Program Phase Will This R&D Support? (please check one)

Design	Construction	Initial Operations	Operate Commercially
x			

Importance (H/M/L) (Reference)	Knowledge (K/P/U) (Reference)	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Fallback Option (Reference for definition)
M	U	4 (covers WBS 6.2.2.2.2.2 and 6.2.2.2.2.3)	54			

2	

Which Program Phase Will This R&D Support? (please check one)

Design	Construction	Initial Operations	Operate Commercially

Importance (H/M/L) (Reference)	Knowledge (K/P/U) (Reference)	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Fallback Option (Reference for definition)

VHTGR H/W Survey Form

NGNP VHTGR H/W R&D Survey Form *(Design Baseline is listed in Reference)*							
WBS # (Reference)	006.2.2.2.2.2	WBS Title	Insulation				
Subject Matter Expert Name	Odile Gelineau		Email	odile.gelineau@areva.com	Phone	33 472747226	
Organization	AREVA NP						

Current Technology Readiness Level
Identify TRL, including rationale

TRL	7	*TRL Definitions are listed in at Reference* *If can't decide please describe in the rationale block; if more space is required please insert "rows" or adjust "height"*
Rationale & Assumptions	**Thermal insulation will be needed to provide thermal protection of metallic materials which would be otherwise subjected to He at very high temperature**	

R&D Needs
Identify and characterize the needs for R&D work to resolve critical issues affecting design, fabrication, testing and operation of the VHTGR plant

Importance and Knowledge definitions are listed in Reference; Estimated Cost or Schedule in a range (e.g., $5 M to $7M or > 12 months) is acceptable
If more rows/space are needs please insert "rows" and/or adjust "height"

1	**Thermal-physical properties (K, CTE, Cp)** **Behavior under oxidation**

Which Program Phase Will This R&D Support? (please check one)

Design	Construction	Initial Operations	Operate Commercially
x			

Importance (H/M/L) *(Reference)*	Knowledge (K/P/U) *(Reference)*	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Fallback Option *(Reference for definition)*
M	P	See Ceramics for Reactor Internals	54			

2	

Which Program Phase Will This R&D Support? (please check one)

Design	Construction	Initial Operations	Operate Commercially

Importance (H/M/L) *(Reference)*	Knowledge (K/P/U) *(Reference)*	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Fallback Option *(Reference for definition)*

Control Rod Sheaths (2.2.2.2.3)

VHTGR H/W Survey Form

NGNP VHTGR H/W R&D Survey Form *(Design Baseline is listed in Reference)*

WBS # (Reference)	006.2.2.2.2.3	WBS Title	Ceramics for Control rods		
Subject Matter Expert Name	Odile Gelineau	Email	odile.gelineau@areva.com	Phone	33 472747226
Organization	AREVA NP				

Current Technology Readiness Level
Identify TRL, including rationale

TRL	2	*TRL Definitions are listed in at Reference* *If can't decide please describe in the rationale block; if more space is required please insert "rows" or adjust "height"*
Rationale & Assumptions	No control rods made of composites were used for the past HTRs, neither for other concepts of reactors. The use of composites C/C is interesting because of their high resistance under high or very high temperature. Other composites such as C/SiC could also be envisioned. A R&D program has been launched in the frame of Antares to confirm the possibility to employ such composite for the control rods.	

R&D Needs
Identify and characterize the needs for R&D work to resolve critical issues affecting design, fabrication, testing and operation of the VHTGR plant

Importance and Knowledge definitions are listed in Reference; Estimated Cost or Schedule in a range (e.g., $5 M to $7M or > 12 months) is acceptable
If more rows/space are needs please insert "rows" and/or adjust "height"

1	Thermal-physical properties (K, CTE, Cp) Mechanical properties including multiaxial strength and design of the component Fracture properties Fatigue properties Irradiation effects on properties including irradiation induced dimensional change and irradiation induced creep Behavior under oxidized atmosphere including oxidation effects on properties Tribology Codification Materials envisioned so far are C/C or C/SiC composites. SiC/SiC composites are not considered mature enough to be compatible with NGNP 2018 schedule.

Which Program Phase Will This R&D Support? (please check one)

Design	Construction	Initial Operations	Operate Commercially
x			

Importance (H/M/L) (Reference)	Knowledge (K/P/U) (Reference)	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Fallback Option (Reference for definition)
M	U	Covered under WBS 6.2.2.2.2.2 Ceramics for Reactor internals	54			

2	

Which Program Phase Will This R&D Support? (please check one)

Design	Construction	Initial Operations	Operate Commercially

Importance (H/M/L) (Reference)	Knowledge (K/P/U) (Reference)	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Fallback Option (Reference for definition)

WBS C Q 10 2 22 NHS 2.2.2.3 Graphite Materials (2.2.2.3)

VHTGR H/W Survey Form

NGNP VHTGR H/W R&D Survey Form *(Design Baseline is listed in Reference)*

WBS # (Reference)	006.2.2.2.3	WBS Title	Graphite		
Subject Matter Expert Name	Odile Gelineau	Email	odile.gelineau@areva.com	Phone	33 472747226
Organization	AREVA NP				

Current Technology Readiness Level
Identify TRL, including rationale

TRL	7	*TRL Definitions are listed in at Reference* *If can't decide please describe in the rationale block; if more space is required please insert "rows" or adjust "height"*
Rationale & Assumptions	Nuclear graphite grades were used in the past HTRs, for which data exist on old graphite grades. These grades are no more available at present time. A R&D program has been launched within Antares program to first select the best candidates among the new grades available (or to ask for a new grade development) and second to acquire data for designing. Grades presently under consideration are PCEA, NBG17 and/or NBG18	

R&D Needs
Identify and characterize the needs for R&D work to resolve critical issues affecting design, fabrication, testing and operation of the VHTGR plant

Importance and Knowledge definitions are listed in Reference; Estimated Cost or Schedule in a range (e.g., $5 M to $7M or > 12 months) is acceptable
If more rows/space are needs please insert "rows" and/or adjust "height"

1	Thermal-physical properties (K, CTE, Cp, emissivity) Mechanical properties including multiaxial strength Fracture properties Fatigue properties Irradiation effects on properties including irradiation induced dimensional change and irradiation induced creep Behavior under oxidized atmosphere including oxidation effects on properties Tribology Codification including fracture models

Which Program Phase Will This R&D Support? (please check one)

Design	Construction	Initial Operations	Operate Commercially
x			

Importance (H/M/L) (Reference)	Knowledge (K/P/U) (Reference)	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Fallback Option (Reference for definition)
H	P	$6 (initial) $14 (subsequent)	54+			

2	

Which Program Phase Will This R&D Support? (please check one)

Design	Construction	Initial Operations	Operate Commercially

Importance (H/M/L) (Reference)	Knowledge (K/P/U) (Reference)	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Fallback Option (Reference for definition)

C.2.3 WBS C Q 10 2 22 NHS 2.2.3 Components Testing

Circulators (2.2.3.3)

VHTGR H/W Survey Form							
NGNP VHTGR H/W R&D Survey Form *(Design Baseline is listed in Reference)*							
WBS # (Reference)	008 2 2 3 3	WBS Title	Gas Circulators for Combined Cycle				
Subject Matter Expert Name	Eric Breuil	Email	eric. breuil@areva.com	Phone	+33.4.72.74.70.87		
Organization	AREVA NP						
Current Technology Readiness Level Identify TRL, including rationale							
TRL	8	*TRL Definitions are listed in at Reference* *If can't decide please describe in the rationale block; if more space is required please insert "rows" or adjust "height"*					
Rationale & Assumptions	Circulators up to 4 Mwe have already operated in HTR reactors. Test program is more dedicated to component qualification than R&D						
R&D Needs Identify and characterize the needs for R&D work to resolve critical issues affecting design, fabrication, testing and operation of the VHTGR plant							
1	Air tests of the impeller (at scale 0.2 to 0.4) Helium tests of Magnetic and Catcher bearings Tests of the circulator shutoff valve Integrated tests at scale one of the whole machine should be required on a very large dedicated loop or during the NGNP commissioning phase.						
H							
Design	Construction	Initial Operations	Operate Commercially				
x							
Importance (H/M/L) *(Reference)*	Knowledge (K/P/U) *(Reference)*	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Fallback Option *(Reference for definition)*	
M	K	2	18				
2							
Which Program Phase Will This R&D Support? (please check one)							
Design	Construction	Initial Operations	Operate Commercially				
Importance (H/M/L) *(Reference)*	Knowledge (K/P/U) *(Reference)*	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Fallback Option *(Reference for definition)*	

Tube IHX (2.2.3.4-1)

VHTGR H/W Survey Form

NGNP VHTGR H/W R&D Survey Form *(Design Baseline is listed in Reference)*

WBS # (Reference)	008 2 2 3 4	WBS Title	IHX

Subject Matter Expert Name	Eric Breuil	Email	eric. breuil@areva.com	Phone	+33.4.72.74.70.87
Organization	AREVA NP				

Current Technology Readiness Level
Identify TRL, including rationale

TRL	5	*TRL Definitions are listed in at Reference* *If can't decide please describe in the rationale block; if more space is required please insert "rows" or adjust "height"*
Rationale & Assumptions	The Tube IHX design is based on extrapolated German past experience. The NGNP requirements leads to operate at high temperature with an innovative secondary fluid mixture He+N2	

R&D Needs
Identify and characterize the needs for R&D work to resolve critical issues affecting design, fabrication, testing and operation of the VHTGR plant

vHTGR H/W Survey Form

1	**Urgent to launch an R&D program:** • tests to confirm fabrication feasibility (tube bending, tube welding, nozzles on hot header, ISIR and assembly, etc) • corrosion and nitriding tests on base and coated materials in representative environment • representative IHX mock-ups from thermo-hydraulic and manufacturing point of view. Test in helium and He+N2 mixture are recommended, that leads to the availability on time of a large test facility (around 10MW) For the component qualification, it is considered that the qualification on a mock-up at scale 1 on a large test facility will be sufficient (no need for intermediate testing on small size mock-ups), subject that manufacturing issues be addressed by dedicated actions

Which Program Phase Will This R&D Support? (please check one)

Design	Construction	Initial Operations	Operate Commercially
x			

Importance (H/M/L) (Reference)	Knowledge (K/P/U) (Reference)	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Fallback Option (Reference for definition)
H	P	1) 1MW loop: $20 2) 10 MW loop - facility: $72 - 112 test article: $1 testing: $7	42, including 12 months testing		Large test facilities (about 10 MW) to be designed and built	

2	

Which Program Phase Will This R&D Support? (please check one)

Design	Construction	Initial Operations	Operate Commercially

Importance (H/M/L) (Reference)	Knowledge (K/P/U) (Reference)	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Fallback Option (Reference for definition)

Plate IHX (2.2.3.4-2)

VHTGR H/W Survey Form

NGNP VHTGR H/W R&D Survey Form ***(Design Baseline is listed in*** *Reference* ***)***

WBS # (Reference)	**008 2 2 3 4**	**WBS Title**	**IHX**		
Subject Matter Expert Name	**Eric Breuil**	**Email**	**eric. breuil@areva.com**	**Phone**	**+33.4.72.74.70.87**
Organization	**AREVA NP**				

Current Technology Readiness Level

Identify TRL, including rationale

TRL	**2**	*TRL Definitions are listed in at Reference* *If can't decide please describe in the rationale block; if more space is required please insert "rows" or adjust "height"*
Rationale & Assumptions	**The Plate IHX feasibility is a concern since NGNP requirements lead to operate at high temperature on the 60 MW loop**	

R&D Needs

Identify and characterize the needs for R&D work to resolve critical issues affecting design, fabrication, testing and operation of the VHTGR plant

vHTGR H/W Survey Form

1	**Urgent to launch an R&D program:** **• for development of visco-plastic model (material data-base to be completed)** **• for corrosion tests on base and coated materials in representative environment** **• for the development of manufacturing techniques (fusion welding, diffusion bonding, brazing, forming, etc)** **• for tests on representative IHX mock-ups from thermo-hydraulic and manufacturing point of view (diffusion bonding, brazing, ISIR).** **For the component qualification, a 3 steps approach is proposed:** **1) tests with small mock-ups in air** **2) tests with small mock-ups in He (about 1 MW test loop). These tests should be used as a basis for providing recommendations on the type of concept to be used for the NGNP** **3) final qualification on a mock-up at scale 1 (at least for the channels and the plates) on a large test facility (around 10MW)**

Which Program Phase Will This R&D Support? (please check one)

Design	Construction	Initial Operations	Operate Commercially

Importance (H/M/L) *(Reference)*	Knowledge (K/P/U) *(Reference)*	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Fallback Option *(Reference for definition)*
M	**P**	**1) 1MW loop: $20** **2) 10 MW loop - facility: $72 - 112** **test article: $1** **testing: $7**	**42, including 12 months testing**	**R&D on manufacturing techniques; heat transfer tests with representative fluids; materials corrosion tests**	**Small test facilities << 1 MW available. Test facilities of 1 MW and 10 MW to be built.**	

2	

Which Program Phase Will This R&D Support? (please check one)

Design	Construction	Initial Operations	Operate Commercially

Importance (H/M/L) *(Reference)*	Knowledge (K/P/U) *(Reference)*	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Fallback Option *(Reference for definition)*

Isolation Valves (2.2.3.5)

VHTGR H/W Survey Form							
NGNP VHTGR H/W R&D Survey Form *(Design Baseline is listed in Reference)*							
WBS # (Reference)	008 2 2 3 5	WBS Title	Hot gas isolation valves				
Subject Matter Expert Name	Bernard Riou	Email	bernard.riou@areva.com	Phone	434-832-4255		
Organization	AREVA NP						
Current Technology Readiness Level Identify TRL, including rationale							
TRL	7	*TRL Definitions are listed in at Reference* *If can't decide please describe in the rationale block; if more space is required please insert "rows" or adjust "height"*					
Rationale & Assumptions	A hot gas isolation valve was designed in the context of the German HTR development program and tested in KVK test facilities. The corresponding valve was designed for operation in He at 900°C and is very close to what is envisioned in the context of the NGNP.						
R&D Needs Identify and characterize the needs for R&D work to resolve critical issues affecting design, fabrication, testing and operation of the VHTGR plant							
Importance and Knowledge definitions are listed in Reference; Estimated Cost or Schedule in a range (e.g., $5 M to $7M or > 12 months) is acceptable *If more rows/space are needs please insert "rows" and/or adjust "height"*							
1	The qualification should be performed in 2 steps: 1) Elementary tests to characterize the fiber conditions, assembly techniques, spacers, etc. 2) Tests on a full scale mock-up in a big test facility in He/N2 (around 10 MW) Test should at least cover: - manufacturing parameters - depressurization tests - pressure loss, heat loss, temperature of the support tube (in mixture He/N2 conditions) - leak tightness tests of the valve - closing and opening - Fatigue and creep-fatigue of specific areas						
Which Program Phase Will This R&D Support? (please check one)							
Design	Construction	Initial Operations	Operate Commercially				
x							
Importance (H/M/L) (Reference)	Knowledge (K/P/U) (Reference)	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Fallback Option (Reference for definition)	
H	P	3.5	12		No for 10 MWth test loops		
2							
Which Program Phase Will This R&D Support? (please check one)							
Design	Construction	Initial Operations	Operate Commercially				
Importance (H/M/L) (Reference)	Knowledge (K/P/U) (Reference)	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Fallback Option (Reference for definition)	

Helium Purification System (2.2.3.6)

VHTGR H/W Survey Form							
NGNP VHTGR H/W R&D Survey Form *(Design Baseline is listed in Reference)*							
WBS # (Reference)	RAD.008.2.2.3.6		WBS Title	Helium Purification System			
Subject Matter Expert Name	John Mayer			Email	john.mayer@areva.com	Phone	508-573-6574
Organization	AREVA NP Inc.						
Current Technology Readiness Level Identify TRL, including rationale							
TRL	8		*TRL Definitions are listed in at Reference* *If can't decide please describe in the rationale block; if more space is required please insert "rows" or adjust "height"*				
Rationale & Assumptions	The configuration and major components of the proposed NGNP Helium Purification System are effectively identical to those currently in use, or used, in various other helium cooled reactors, including AVR, THTR-300, HTTR, and HTR-10. Use of this system for the NGNP will require sizing of the various components for the desired flow rates.						
R&D Needs Identify and characterize the needs for R&D work to resolve critical issues affecting design, fabrication, testing and operation of the VHTGR plant							
Importance and Knowledge definitions are listed in Reference; Estimated Cost or Schedule in a range (e.g., $5 M to $7M or > 12 months) is acceptable *If more rows/space are needs please insert "rows" and/or adjust "height"*							
1	Selection and qualification of appropriate charcoal						

Which Program Phase Will This R&D Support? (please check one)			
Design	Construction	Initial Operations	Operate Commercially
X			

Importance (H/M/L) *(Reference)*	Knowledge (K/P/U) *(Reference)*	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Fallback Option *(Reference for definition)*
M	P	0.2	6			

2	

Which Program Phase Will This R&D Support? (please check one)			
Design	Construction	Initial Operations	Operate Commercially

Importance (H/M/L) *(Reference)*	Knowledge (K/P/U) *(Reference)*	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Fallback Option *(Reference for definition)*

Shutdown Cooling System (2.2.3.7)

VHTGR H/W Survey Form								
NGNP VHTGR H/W R&D Survey Form *(Design Baseline is listed in Reference)*								
WBS # (Reference)	RAD.008.2.2.3.7		WBS Title	Shutdown Cooling System				
Subject Matter Expert Name	John Mayer			Email	john.mayer@areva.com		Phone	508-573-6574
Organization	AREVA NP Inc.							
Current Technology Readiness Level Identify TRL, including rationale								
TRL	7		*TRL Definitions are listed in at Reference* *If can't decide please describe in the rationale block; if more space is required please insert "rows" or adjust "height"*					
Rationale & Assumptions	The proposed Shutdown Cooling System is very similar to the Auxiliary Cooling System of the HTTR reactor. The primary difference is the location of the helium-to-water heat exchanger. The NGNP design places the heat exchanger within the lower plenum of the reactor vessel. In the HTTR, the heat exchanger is housed in a separate vessel. All of the components of the system are fairly common and well understood, including pumps, valves, heat exchangers, circulators.							
R&D Needs Identify and characterize the needs for R&D work to resolve critical issues affecting design, fabrication, testing and operation of the VHTGR plant								
Importance and Knowledge definitions are listed in Reference; Estimated Cost or Schedule in a range (e.g., $5 M to $7M or > 12 months) is acceptable *If more rows/space are needs please insert "rows" and/or adjust "height"*								
1	There are no critical R&D issues related to this system.							
Which Program Phase Will This R&D Support? (please check one)								
Design		Construction		Initial Operations		Operate Commercially		
Importance (H/M/L) (Reference)	Knowledge (K/P/U) *(Reference)*	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Fallback Option *(Reference for definition)*		
2								
Which Program Phase Will This R&D Support? (please check one)								
Design		Construction		Initial Operations		Operate Commercially		
Importance (H/M/L) (Reference)	Knowledge (K/P/U) *(Reference)*	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Fallback Option *(Reference for definition)*		

Fuel Handling System (2.2.3.8)

VHTGR H/W Survey Form

NGNP VHTGR H/W R&D Survey Form *(Design Baseline is listed in Reference)*

WBS # (Reference)	RAD.008.2.2.3.8	WBS Title	Fuel Handling System		
Subject Matter Expert Name	John Mayer	Email	john.mayer@areva.com	Phone	508-573-6574
Organization	AREVA NP Inc.				

Current Technology Readiness Level
Identify TRL, including rationale

TRL	6	*TRL Definitions are listed in at Reference* *If can't decide please describe in the rationale block; if more space is required please insert "rows" or adjust "height"*
Rationale & Assumptions	This TRL applies to the Fuel Server portion of the Fuel Handling System, which has been described only as a design concept at this point. The remainder of the Fuel Handling System components, including the Fuel Elevator, Adaptor Plate and Fuel Handling Machine, have been demonstrated at the Fort St. Vrain reactor. In addition, the HTTR reactor utilized a similar set of components. These portions of the system should be considered TRL 8.	

R&D Needs
Identify and characterize the needs for R&D work to resolve critical issues affecting design, fabrication, testing and operation of the VHTGR plant

Importance and Knowledge definitions are listed in Reference; Estimated Cost or Schedule in a range (e.g., $5 M to $7M or > 12 months) is acceptable
If more rows/space are needs please insert "rows" and/or adjust "height"

1	The Fuel Server system needs to be designed based on the current system concept. Key activities should include mechanical design of the shield enclosure, design of the robotic fuel cart, and development of the control software. Testing of the Fuel Server, beyond initial component testing, should be included in the testing program developed for the complete Fuel Handling System.

Which Program Phase Will This R&D Support? (please check one)

Design	Construction	Initial Operations	Operate Commercially
X	X	X	

Importance (H/M/L) (Reference)	Knowledge (K/P/U) (Reference)	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Fallback Option (Reference for definition)
M	K	0.5	12			

2	

Which Program Phase Will This R&D Support? (please check one)

Design	Construction	Initial Operations	Operate Commercially

Importance (H/M/L) (Reference)	Knowledge (K/P/U) (Reference)	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Fallback Option (Reference for definition)

Reactor Cavity Cooling System (Rccs) (2.2.3.10)

VHTGR H/W Survey Form							
NGNP VHTGR H/W R&D Survey Form (Design Baseline is listed in Reference)							
WBS # (Reference)	RAD.08.2.2.3.10		WBS Title	RCCS			
Subject Matter Expert Name	John Mayer			Email	john.mayer@areva.com	Phone	508-573-6574
Organization	AREVA NP Inc.						
Current Technology Readiness Level Identify TRL, including rationale							
TRL	5		TRL Definitions are listed in at Reference If can't decide please describe in the rationale block; if more space is required please insert "rows" or adjust "height"				
Rationale & Assumptions	Use of an unisulated reactor vessel coupled with a water-cooled panel heat exchanger as a core cooling mechanism for accident conditions has not been demonstrated. The basic components of the system are fairly common and well understood. Proper design and sizing of the system will require a demonstrated understanding of key heat transfer parameters for the vessel wall and panel surfaces.						
R&D Needs Identify and characterize the needs for R&D work to resolve critical issues affecting design, fabrication, testing and operation of the VHTGR plant							
Importance and Knowledge definitions are listed in Reference; Estimated Cost or Schedule in a range (e.g., $5 M to $7M or > 12 months) is acceptable If more rows/space are needs please insert "rows" and/or adjust "height"							
1	Characterization of the heat transfer characteristics of the anticipated, or proposed, surface treatments for the reactor vessel and the panel heat exchanger will need to be accomplished. A large scale demonstration of the capability of the RCCS to release the decay heat for the reactor is recommended.						
Which Program Phase Will This R&D Support? (please check one)							
Design	Construction		Initial Operations		Operate Commercially		
X	X						
Importance (H/M/L) (Reference)	Knowledge (K/P/U) (Reference)	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Fallback Option (Reference for definition)	
H	P	1	24		Available facility in ANL to be adapted to the needs of the large scale demonstration of RCCS operation		
2							
Which Program Phase Will This R&D Support? (please check one)							
Design	Construction		Initial Operations		Operate Commercially		
Importance (H/M/L) (Reference)	Knowledge (K/P/U) (Reference)	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Fallback Option (Reference for definition)	

Hot Gas Duct (2.2.3.12)

VHTGR H/W Survey Form

NGNP VHTGR H/W R&D Survey Form *(Design Baseline is listed in Reference)*

WBS # (Reference)	008 2 2 3 12	WBS Title	Hot gas ducts
Subject Matter Expert Name	Bernard Riou	Email	bernard.riou@areva.com
Phone	434-832-4255		
Organization	AREVA NP		

Current Technology Readiness Level
Identify TRL, including rationale

TRL	6	*TRL Definitions are listed in at Reference* *If can't decide please describe in the rationale block; if more space is required please insert "rows" or adjust "height"*
Rationale & Assumptions	The reference design for the primary and secondary hot gas duct is the Vee-shaped metallic concept. This design is considered to be compatible with the core outlet temperature envisaged, subject to demonstrating that no significant hot streaks should be expected. The ceramic concept is envisioned as a fall back option for the primary hot gas duct.	

R&D Needs
Identify and characterize the needs for R&D work to resolve critical issues affecting design, fabrication, testing and operation of the VHTGR plant

Importance and Knowledge definitions are listed in Reference; Estimated Cost or Schedule in a range (e.g., $5 M to $7M or > 12 months) is acceptable
If more rows/space are needs please insert "rows" and/or adjust "height"

1

The qualification should be performed in 3 steps:
1) Elementary tests to characterize the fiber conditions, assembly techniques, spacers, etc.
2) Tests on a small mock-up in a test facility of about 1 MWth or less to validate the fiber specification and the ceramic spacer specification (if possible in He).
3) Tests on a full scale mock-up in a big test facility in He (around 10 MW)

Test should at least cover:
- depressurization tests
- pressure loss, heat loss, temperature of the support tube (in He conditions)
- leak tightness tests of the connection areas
- Fatigue and creep-fatigue tests (e.g. bellows, Vee-shape spacers,etc)

In the first stages of the design, tests should cover both the metallic and ceramic design (pending the confirmation of the feasibility of the metallic design)

Which Program Phase Will This R&D Support? (please check one)

Design	Construction	Initial Operations	Operate Commercially
x			

Importance (H/M/L) (Reference)	Knowledge (K/P/U) (Reference)	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Fallback Option (Reference for definition)
H	P	4.5			No for 1 to 10 MWth test loops	

2

Which Program Phase Will This R&D Support? (please check one)

Design	Construction	Initial Operations	Operate Commercially

Importance (H/M/L) (Reference)	Knowledge (K/P/U) (Reference)	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Fallback Option (Reference for definition)

Instrumentation (2.2.3.13)

VHTGR H/W Survey Form

NGNP VHTGR H/W R&D Survey Form *(Design Baseline is listed in Reference)*

WBS # (Reference)	008 2 2 3 13	WBS Title	Instrumentation				
Subject Matter Expert Name	Bernard Riou			Email	bernard.riou@areva.com	Phone	434-832-4255
Organization	AREVA NP						

Current Technology Readiness Level
Identify TRL, including rationale

TRL	7	*TRL Definitions are listed in at Reference* *If can't decide please describe in the rationale block; if more space is required please insert "rows" or adjust "height"*
Rationale & Assumptions	NGNP will be the test bed for testing and validating HTR technology and specific instrumentation might be required for operation at high temperature. The detail of this instrumentation (in particular the operating conditions) will be a function of the type of testing and experiments envisioned and will depend also of the monitoring strategy.	

R&D Needs
Identify and characterize the needs for R&D work to resolve critical issues affecting design, fabrication, testing and operation of the VHTGR plant

voter H/W Survey Form

1 **Below example of R&D which might be envisioned:**
• Neutron flux detectors – Some R&D and qualification efforts may be desirable to select detector technology and verify adequate sensitivity and lifetime.
• Temperature Measurements – Standard thermocouples used in nuclear plants today are capable of measuring operating temperatures up to 1200ºC. Monitoring accident conditions may require the use of Pt-Rh thermocouples for operation at higher temperatures. These types of thermocouples are not used today and limited data about their reliability in nuclear environments exists. R&D may be needed to qualify Pt-Rh thermocouples for use in the NGNP, particularly if measurement of temperatures within the core is desired.
Further needs should arise together with the definition of the monitoring strategy.

Which Program Phase Will This R&D Support? (please check one)

Design	Construction	Initial Operations	Operate Commercially
		x	x

Importance (H/M/L) (Reference)	Knowledge (K/P/U) (Reference)	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Fallback Option (Reference for definition)
H	P	2				

2

Which Program Phase Will This R&D Support? (please check one)

Design	Construction	Initial Operations	Operate Commercially

Importance (H/M/L) (Reference)	Knowledge (K/P/U) (Reference)	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Fallback Option (Reference for definition)

C.2.4 WBS C Q 10 2 22 NHP 2.2.4 Computer Codes, Methods Development and Qualification

WBS C Q 10 2 22 NHP 2.2.4.4 Code Development

Neutronics Codes (2.2.4.4.1-1 and 2.2.4.4.1-2)

VHTGR Computer Code / Modeling Survey Form						
NGNP VHTGR Computer Code R&D Survey Form *(Design Baseline is listed in References)*						
WBS # *(References)*	2.2.4.4.1	WBS Title	Code Development - Neutronics Codes			
Subject Matter Expert Name	Denis Verrier	Email	denis.verrier@areva.com	Phone	33 (0)4 72 74 88 96	
Organization	AREVA NP					
Computer Code Usage Category (Please check all applicable) Please check all applicable, if existing code please provide code name with version						
Reactor System Analysis		Autonomous Control				
Neutronics		Materials Analysis				
Thermal-Hydraulic		Structure Analysis				
Severe Accident		PCS Analysis				
FP Transport		Heat Exchanger Analysis				
Containment Analysis		Human Factor Simulation				
PRA		Economic				
Fuel Performance						
coupled neutronics/TH	CABERNET (=NEPHTYS / STAR-CD)					
Objectives of Modeling						
Reactivity, power and temperature, burnup and fluence distribution calculation in steady state and transient conditions for block type cores (input to fuel performance assessment).						
Status of Computer Code Please check applicable						
Current Code Fully Applicable	Current Code Needs Modification	Current Code Needs Major Modification	Current Code Needs Validation	No Existing Code		
X	X		X			
Current Technology Readiness Level Identify TRL, including rationale						
TRL	6 - Integrated Modeling (Prototype) Completed	*TRL Definitions are listed in at References* *If can't decide please describe in the rationale block; if more space is required please insert "rows" or adjust "height"*				
Rationale & Assumptions						
R&D Needs (Description of Technical Approach) Identify and characterize the needs for R&D work to resolve critical issues affecting design, fabrication, testing and operation of the VHTGR plant						
Importance and Knowledge definitions are listed in References; Estimated Cost or Schedule in a range (e.g., \$5 M to \$7M or > 12 months) is acceptable *If more rows/space are needs please insert "rows" and/or adjust "height"*						
1	Enhancement of capabilities for the calculation of transient analyses.					
Importance (H/M/L) *(Reference)*	Knowledge (K/P/U) *(Reference)*	Estimated Cost (\$M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Validation Data Required (What kind?)	Alternative Model(s)
H	P	0.5	12			
2						
Importance (H/M/L) *(Reference)*	Knowledge (K/P/U) *(Reference)*	Estimated Cost (\$M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Validation Data Required (What kind?)	Alternative Model(s)

VHTGR Computer Code / Modeling Survey Form

NGNP VHTGR Computer Code R&D Survey Form *(Design Baseline is listed in References)*					
WBS # *(References)*	2.2.4.4.1	WBS Title	Code Development - Neutronics Codes		
Subject Matter Expert Name	Denis Verrier	Email	denis.verrier@areva.com	Phone	33 (0)4 72 74 88 96
Organization	AREVA NP				

Computer Code Usage Category (Please check all applicable) Please check all applicable, if existing code please provide code name with version			
Reactor System Analysis		Autonomous Control	
Neutronics	NEPHTYS	Materials Analysis	
Thermal-Hydraulic		Structure Analysis	
Severe Accident		PCS Analysis	
FP Transport		Heat Exchanger Analysis	
Containment Analysis		Human Factor Simulation	
PRA		Economic	
Fuel Performance			
(please add-on)			

Objectives of Modeling

- **Reactivity effects (first criticality, moderator and doppler coefficients, control rod worth, reactivity loss versus depletion)**
- **3D neutron flux and nuclear power distribution within the reactor core.**
- **3D burnup distribution and nuclide inventory for back-end cycle and decay heat issues**

Status of Computer Code
Please check applicable

Current Code Fully Applicable	Current Code Needs Modification	Current Code Needs Major Modification	Current Code Needs Validation	No Existing Code
	X		X	

Current Technology Readiness Level

Identify TRL, including rationale

TRL	6 - Integrated Modeling (Prototype) Completed	*TRL Definitions are listed in at References* *If can't decide please describe in the rationale block; if more space is required please insert "rows" or adjust "height"*
Rationale & Assumptions		

R&D Needs (Description of Technical Approach)
Identify and characterize the needs for R&D work to resolve critical issues affecting design, fabrication, testing and operation of the VHTGR plant

Importance and Knowledge definitions are listed in References; Estimated Cost or Schedule in a range (e.g., $5 M to $7M or > 12 months) is acceptable
If more rows/space are needs please insert "rows" and/or adjust "height"

1

Importance (H/M/L) *(Reference)*	Knowledge (K/P/U) *(Reference)*	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Validation Data Required (What kind?)	Alternative Model(s)

2

Importance (H/M/L) *(Reference)*	Knowledge (K/P/U) *(Reference)*	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Validation Data Required (What kind?)	Alternative Model(s)

Thermal/ Hydraulics/Pneumatics Codes (2.2.4.4.2)

VHTGR Computer Code / Modeling Survey Form						
NGNP VHTGR Computer Code R&D Survey Form *(Design Baseline is listed in References)*						
WBS # *(References)*	2.2.4.4.2	WBS Title	Code development - TH			
Subject Matter Expert Name	Denis Verrier	Email	denis.verrier@areva.com	Phone	33 (0)4 72 74 88 96	
Organization	AREVA NP					

Computer Code Usage Category (Please check all applicable) Please check all applicable, if existing code please provide code name with version			
Reactor System Analysis		Autonomous Control	
Neutronics		Materials Analysis	
Thermal-Hydraulic	STAR-CD	Structure Analysis	
Severe Accident		PCS Analysis	
FP Transport		Heat Exchanger Analysis	
Containment Analysis		Human Factor Simulation	
PRA		Economic	
Fuel Performance			
(please add-on)			

Objectives of Modeling

Determination of: (a) thermal loadings on the components (vessels, internals, fuel…) during normal or upset conditions, (b) the thermal behavior of the core, (c) the mixing inside the primary system, (d) head losses and performances of components, (e flow repartition across the components, (f) pressure shok waves.

Status of Computer Code
Please check applicable

Current Code Fully Applicable	Current Code Needs Modification	Current Code Needs Major Modification	Current Code Needs Validation	No Existing Code
X			X	

Current Technology Readiness Level

Identify TRL, including rationale

TRL	6 - Integrated Modeling (Prototype) Completed	*TRL Definitions are listed in at References* *If can't decide please describe in the rationale block; if more space is required please insert "rows" or adjust "height"*
Rationale & Assumptions		

R&D Needs (Description of Technical Approach)
Identify and characterize the needs for R&D work to resolve critical issues affecting design, fabrication, testing and operation of the VHTGR plant

Importance and Knowledge definitions are listed in References; Estimated Cost or Schedule in a range (e.g., $5 M to $7M or > 12 months) is acceptable
If more rows/space are needs please insert "rows" and/or adjust "height"

1 Development of graphite oxidation model for air ingress transients on reactor internal structures

Importance (H/M/L) *(Reference)*	Knowledge (K/P/U) *(Reference)*	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Validation Data Required (What kind?)	Alternative Model(s)
H	P	0.2	12			

2

Importance (H/M/L) *(Reference)*	Knowledge (K/P/U) *(Reference)*	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Validation Data Required (What kind?)	Alternative Model(s)

Other Codes (2.2.4.4.3)

VHTGR Computer Code / Modeling Survey Form

NGNP VHTGR Computer Code R&D Survey Form *(Design Baseline is listed in References)*

WBS # *(References)*	2.2.4.4.3	WBS Title	Code Development - Other codes

Subject Matter Expert Name	Denis Verrier	Email	denis.verrier@areva.com	Phone	33 (0)4 72 74 88 96
Organization	AREVA NP				

Computer Code Usage Category (Please check all applicable)
Please check all applicable, if existing code please provide code name with version

Reactor System Analysis		Autonomous Control	
Neutronics		Materials Analysis	
Thermal-Hydraulic		Structure Analysis	
Severe Accident		PCS Analysis	
FP Transport	X	Heat Exchanger Analysis	
Containment Analysis		Human Factor Simulation	
PRA		Economic	
Fuel Performance			
(please add-on)			

Objectives of Modeling

Transport of radio contaminant species from the fuel block graphite walls into the primary coolant in normal operation and upto the environment in case of accidents.

Status of Computer Code
Please check applicable

Current Code Fully Applicable	Current Code Needs Modification	Current Code Needs Major Modification	Current Code Needs Validation	No Existing Code
		X		

Current Technology Readiness Level

Identify TRL, including rationale

TRL	TRL 4 - Individual Module Modeling at Laboratory Environment Completed	*TRL Definitions are listed in at References* *If can't decide please describe in the rationale block; if more space is required please insert "rows" or adjust "height"*
Rationale & Assumptions		

R&D Needs (Description of Technical Approach)
Identify and characterize the needs for R&D work to resolve critical issues affecting design, fabrication, testing and operation of the VHTGR plant

Importance and Knowledge definitions are listed in References; Estimated Cost or Schedule in a range (e.g., $5 M to $7M or > 12 months) is acceptable
If more rows/space are needs please insert "rows" and/or adjust "height"

1 Models for (a) the assessment of product activation in the primary circuit (in particular tritium and 14C), (b) radio-contaminants distribution in the primary circuit, making distinction between circulating activity, plated out / deposited activity and purification system, during both normal operation and accidental situations, (c) radio-contaminants releases outside the primary pressure boundary, (d) radio-contaminants releases in the environment for accidental situations. Also, experimental work required for model qualification and the actual qualification effort.

Importance (H/M/L) *(Reference)*	Knowledge (K/P/U) *(Reference)*	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Validation Data Required (What kind?)	Alternative Model(s)
H	P	6	60			

2

Importance (H/M/L) *(Reference)*	Knowledge (K/P/U) *(Reference)*	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Validation Data Required (What kind?)	Alternative Model(s)

VHTGR Computer Code / Modeling Survey Form

NGNP VHTGR Computer Code R&D Survey Form *(Design Baseline is listed in References)*

WBS # *(References)*	2.2.4.4.3	WBS Title	Code development (other codes)				
Subject Matter Expert Name	Nadim MOUSSALLAM			Email	nadim.moussallam@areva.com	Phone	+33 4 72 74 74 69
Organization							

Computer Code Usage Category (Please check all applicable)
Please check all applicable, if existing code please provide code name with version

Reactor System Analysis		Autonomous Control	
Neutronics		Materials Analysis	
Thermal-Hydraulic		Structure Analysis	x
Severe Accident		PCS Analysis	
FP Transport		Heat Exchanger Analysis	
Containment Analysis		Human Factor Simulation	
PRA		Economic	
Fuel Performance			
(please add-on)			

Objectives of Modeling

Assessment of component behavior under normal operation and accident mechanical and thermal loadings

Status of Computer Code
Please check applicable

Current Code Fully Applicable	Current Code Needs Modification	Current Code Needs Major Modification	Current Code Needs Validation	No Existing Code
	X			

Current Technology Readiness Level

Identify TRL, including rationale

TRL	3	*TRL Definitions are listed in at References* *If can't decide please describe in the rationale block; if more space is required please insert "rows" or adjust "height"*
Rationale & Assumptions	The main tools for structural analysis exist but specific modeling and correlations relative to NGNP configurations and materials have to be developed. The proposed safety approach excludes the vessel rupture and thus relies on a leak-before-break (LBB) approach that has not been established for gas cooled reactors yet.	

R&D Needs (Description of Technical Approach)
Identify and characterize the needs for R&D work to resolve critical issues affecting design, fabrication, testing and operation of the VHTGR plant

Importance and Knowledge definitions are listed in References; Estimated Cost or Schedule in a range (e.g., \$5 M to \$7M or > 12 months) is acceptable
If more rows/space are needs please insert "rows" and/or adjust "height"

1
- Intoduction in structural mechanics codes of specific constitutive laws for HTR material (graphite, visco-plastic behavior of Ni base alloys): completing the experimental databases and developing numerical models
- Seismic behavior of a block type core: development of a block type core modeling and experimental determination of input data for the model through tests on a vibration table
- Fluid structure interaction and flow induced vibrations
- LBB methodology for gas cooled reactors

Importance (H/M/L) *(Reference)*	Knowledge (K/P/U) *(Reference)*	Estimated Cost (\$M)	Estimated Schedule (Months)		Validation Data Required (What kind?)	Alternative Model(s)
H	P/U	1	18	• Databases on material properties • Preliminary tests on fuel block seismic behavior • A preliminary study has shown that LBB approach could be applied to gas reactors		

2

Importance (H/M/L) *(Reference)*	Knowledge (K/P/U) *(Reference)*	Estimated Cost (\$M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Validation Data Required (What kind?)	Alternative Model(s)

Fuel Performance Models and Codes (2.2.4.4.5)

VHTGR Computer Code / Modeling Survey Form

NGNP VHTGR Computer Code R&D Survey Form *(Design Baseline is listed in References)*

WBS # *(References)*	2.2.4.4.5	WBS Title	Code Development - Fuel

Subject Matter Expert Name	Denis Verrier	Email	denis.verrier@areva.com	Phone	33 (0)4 72 74 88 96
Organization	AREVA NP				

Computer Code Usage Category (Please check all applicable)
Please check all applicable, if existing code please provide code name with version

Reactor System Analysis		Autonomous Control	
Neutronics		Materials Analysis	
Thermal-Hydraulic		Structure Analysis	
Severe Accident		PCS Analysis	
FP Transport		Heat Exchanger Analysis	
Containment Analysis		Human Factor Simulation	
PRA		Economic	
Fuel Performance	ATLAS		
(please add-on)			

Objectives of Modeling

Assessment of coated particles performance during normal operation and accidental conditions. Calculation of the failure fraction and fission product release rate from a fuel load in normal operation or accidental conditions

Status of Computer Code
Please check applicable

Current Code Fully Applicable	Current Code Needs Modification	Current Code Needs Major Modification	Current Code Needs Validation	No Existing Code
	X		X	

Current Technology Readiness Level

Identify TRL, including rationale

TRL	TRL 5 - Individual Module Modeling Completed and	*TRL Definitions are listed in at References* *If can't decide please describe in the rationale block; if more space is required please insert "rows" or adjust "height"*
Rationale & Assumptions		

R&D Needs (Description of Technical Approach)
Identify and characterize the needs for R&D work to resolve critical issues affecting design, fabrication, testing and operation of the VHTGR plant

Importance and Knowledge definitions are listed in References; Estimated Cost or Schedule in a range (e.g., $5 M to $7M or > 12 months) is acceptable
If more rows/space are needs please insert "rows" and/or adjust "height"

1 Coated particle irradiation at relevant operating conditions (burnup, temperature, fluence); heat-up experiment of irradiated fuel particles

Importance (H/M/L) *(Reference)*	Knowledge (K/P/U) *(Reference)*	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Validation Data Required (What kind?)	Alternative Model(s)
H	P	1.5	24	Several international benchmarks in the frame of IAEA and European programs		

2 Development of UCO models

Importance (H/M/L) *(Reference)*	Knowledge (K/P/U) *(Reference)*	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Validation Data Required (What kind?)	Alternative Model(s)
H	P	Included above	24			

VHTGR Computer Code / Modeling Survey Form

NGNP VHTGR Computer Code R&D Survey Form *(Design Baseline is listed in References)*

WBS # *(References)*	2.2.4.5	WBS Title	Code development/qualification - TH

Subject Matter Expert Name	Robert Martin	Email	robertp.martin@areva.com	Phone	01 434 832 2319
Organization	AREVA NP				

Computer Code Usage Category (Please check all applicable)

Please check all applicable, if existing code please provide code name with version

Reactor System Analysis	RELAP5-3D	Autonomous Control	
Neutronics		Materials Analysis	
Thermal-Hydraulic		Structure Analysis	
Severe Accident		PCS Analysis	
FP Transport		Heat Exchanger Analysis	
Containment Analysis		Human Factor Simulation	
PRA		Economic	
Fuel Performance			
(please add-on)			

Objectives of Modeling

Full systems integration for analysis of thermal-hydraulic response during both normal and transient/accident conditions. Capability exists to couple I&C, neutronics, fission product transport, containment analysis, fuel performance, and severe accident models. Unique capability to model water ingress. Unique capability to interface with other computation tools.

Status of Computer Code

Please check applicable

Current Code Fully Applicable	Current Code Needs Modification	Current Code Needs Major Modification	Current Code Needs Validation	No Existing Code
	X		X	

Current Technology Readiness Level

Identify TRL, including rationale

TRL	8	*TRL Definitions are listed in at References* *If can't decide please describe in the rationale block; if more space is required please insert "rows" or adjust "height"*
Rationale & Assumptions	RELAP5 was originally developed as a best-estimate thermal-hydraulic systems analysis tool for the performance of safety analysis of light water reactors. The US DOE has sponsored code developed to extend its capability to the broadest extent possible, including Generation IV needs. Application to NGNP concepts and subsequent code development is ongoing through US DOE sponsorship at the Idaho National Laboratory. The report INEEL/EXT-04-02293, Next Generation Nuclear Plant - Design Methods Development and Validation Research and Development Program Plan, highlights development needs for RELAP5-3D.	

R&D Needs (Description of Technical Approach)

Identify and characterize the needs for R&D work to resolve critical issues affecting design, fabrication, testing and operation of the VHTGR plant

Importance and Knowledge definitions are listed in References; Estimated Cost or Schedule in a range (e.g., $5 M to $7M or > 12 months) is acceptable
If more rows/space are needs please insert "rows" and/or adjust "height"

1 Several areas with regard to both modeling and validation are identified in the report INEEL/EXT-04-02293. Validation beyond that identified in INEEL/EXT-04-02293 and consistent with that planned for MANTA should be pursued.

Importance (H/M/L) *(Reference)*	Knowledge (K/P/U) *(Reference)*	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Validation Data Required (What kind?)	Alternative Model(s)
H	P					

2 The INL has recognize a need to couple Computation Fluid Dynamics models to RELAP5-3D. Currently, RELAP5-3D is capable of coupling to the FLUENT CFD code. If the role of RELAP5-3D expands, there may be value to the project coupling the CFD code STAR-CD with RELAP5-3D to best utilize our investment in our STAR-CD models for the VHTR.

Importance (H/M/L) *(Reference)*	Knowledge (K/P/U) *(Reference)*	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Validation Data Required (What kind?)	Alternative Model(s)
H	P					Stand-alone STAR-3D, coupled RELAP5-FLUENT

WBS C Q 10 2 22 NHP 2.2.4.5 Code Qualification (2.2.4.5-1 to 2.2.4.5-7)

VHTGR Computer Code / Modeling Survey Form

NGNP VHTGR Computer Code R&D Survey Form *(Design Baseline is listed in References)*

WBS # *(References)*	**2.2.4.5**	**WBS Title**	**Code Qualification - Fuel**			
Subject Matter Expert Name	**Denis Verrier**	**Email**	**denis.verrier@areva.com**	**Phone**	**33 (0)4 72 74 88 96**	
Organization	**AREVA NP**					

Computer Code Usage Category (Please check all applicable)
Please check all applicable, if existing code please provide code name with version

Reactor System Analysis		**Autonomous Control**	
Neutronics		**Materials Analysis**	
Thermal-Hydraulic		**Structure Analysis**	
Severe Accident		**PCS Analysis**	
FP Transport		**Heat Exchanger Analysis**	
Containment Analysis		**Human Factor Simulation**	
PRA		**Economic**	
Fuel Performance	**ATLAS**		
(please add-on)			

Objectives of Modeling

Assessment of coated particles performance during normal operation and accidental conditions. Calculation of the failure fraction and fission product release rate from a fuel load in normal operation or accidental conditions

Status of Computer Code
Please check applicable

Current Code Fully Applicable	Current Code Needs Modification	Current Code Needs Major Modification	Current Code Needs Validation	No Existing Code
	X		X	

Current Technology Readiness Level

Identify TRL, including rationale

TRL	**TRL 5 - Individual Module Modeling Completed and**	*TRL Definitions are listed in at References* *If can't decide please describe in the rationale block; if more space is required please insert "rows" or adjust "height"*
Rationale & Assumptions		

R&D Needs (Description of Technical Approach)
Identify and characterize the needs for R&D work to resolve critical issues affecting design, fabrication, testing and operation of the VHTGR plant

Importance and Knowledge definitions are listed in References; Estimated Cost or Schedule in a range (e.g., $5 M to $7M or > 12 months) is acceptable
If more rows/space are needs please insert "rows" and/or adjust "height"

1 **Coated particle irradiation at relevant operating conditions (burnup, temperature, fluence); heat-up experiment of irradiated fuel particles**

Importance (H/M/L) *(Reference)*	Knowledge (K/P/U) *(Reference)*	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Validation Data Required (What kind?)	Alternative Model(s)
H	P	5	30	Several international benchmarks in connection with IAEA and European programs		

2

Importance (H/M/L) *(Reference)*	Knowledge (K/P/U) *(Reference)*	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Validation Data Required (What kind?)	Alternative Model(s)

VHTGR Computer Code / Modeling Survey Form

NGNP VHTGR Computer Code R&D Survey Form *(Design Baseline is listed in References)*

WBS # *(References)*	**2.2.4.5**	**WBS Title**	**Code Qualification**		
Subject Matter Expert Name	**Denis Verrier**	**Email**	**denis.verrier@areva.com**	**Phone**	**33 (0)4 72 74 88 96**
Organization	**AREVA NP**				

Computer Code Usage Category (Please check all applicable)
Please check all applicable, if existing code please provide code name with version

Reactor System Analysis		**Autonomous Control**	
Neutronics	**MCNP**	**Materials Analysis**	
Thermal-Hydraulic		**Structure Analysis**	
Severe Accident		**PCS Analysis**	
FP Transport		**Heat Exchanger Analysis**	
Containment Analysis		**Human Factor Simulation**	
PRA		**Economic**	
Fuel Performance			
(please add-on)			

Objectives of Modeling

Reference steady state core calculation for all type of cores

Status of Computer Code
Please check applicable

Current Code Fully Applicable	Current Code Needs Modification	Current Code Needs Major Modification	Current Code Needs Validation	No Existing Code
X			X	

Current Technology Readiness Level

Identify TRL, including rationale

TRL	**6 - Integrated Modeling (Prototype) Completed**	*TRL Definitions are listed in at References* *If can't decide please describe in the rationale block; if more space is required please insert "rows" or adjust "height"*
Rationale & Assumptions		

R&D Needs (Description of Technical Approach)
Identify and characterize the needs for R&D work to resolve critical issues affecting design, fabrication, testing and operation of the VHTGR plant

Importance and Knowledge definitions are listed in References; Estimated Cost or Schedule in a range (e.g., $5 M to $7M or > 12 months) is acceptable
If more rows/space are needs please insert "rows" and/or adjust "height"

1 **Dedicated critical experiments, with an asymptotic spectrum representative of the expected prismatic fuel assembly and core, with full access to pin-by-pin power distributions, and control rod and burnable poisons worths are needed.**

Importance (H/M/L) *(Reference)*	Knowledge (K/P/U) *(Reference)*	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Validation Data Required (What kind?)	Alternative Model(s)
H	U	8 (including R&D Need 2)	36 (including R&D Need 2)	FSV and HTTR first criticality		No

2 **Experimental data of neutronic characteristics (spectrum, fission and capture rates) at the interface between a prismatic fuel assembly and a graphite reflector assembly.**

Importance (H/M/L) *(Reference)*	Knowledge (K/P/U) *(Reference)*	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Validation Data Required (What kind?)	Alternative Model(s)
H	U	*Included above*				

VHTGR Computer Code / Modeling Survey Form							
NGNP VHTGR Computer Code R&D Survey Form *(Design Baseline is listed in References)*							
WBS # *(References)*	2.2.4.5	WBS Title	Code Qualification				
Subject Matter Expert Name	Denis Verrier	Email	denis.verrier@areva.com	Phone	33 (0)4 72 74 88 96		
Organization	AREVA NP						

Computer Code Usage Category (Please check all applicable) Please check all applicable, if existing code please provide code name with version			
Reactor System Analysis		Autonomous Control	
Neutronics	MONTEBURNS	Materials Analysis	
Thermal-Hydraulic		Structure Analysis	
Severe Accident		PCS Analysis	
FP Transport		Heat Exchanger Analysis	
Containment Analysis		Human Factor Simulation	
PRA		Economic	
Fuel Performance			
(please add-on)			

Objectives of Modeling

Reference depletion and decay heat core calculations for all types of cores

Status of Computer Code
Please check applicable

Current Code Fully Applicable	Current Code Needs Modification	Current Code Needs Major Modification	Current Code Needs Validation	No Existing Code
X			X	

Current Technology Readiness Level

Identify TRL, including rationale

TRL	6 - Integrated Modeling (Prototype) Completed	*TRL Definitions are listed in at References* *If can't decide please describe in the rationale block; if more space is required please insert "rows" or adjust "height"*
Rationale & Assumptions		

R&D Needs (Description of Technical Approach)
Identify and characterize the needs for R&D work to resolve critical issues affecting design, fabrication, testing and operation of the VHTGR plant

Importance and Knowledge definitions are listed in References; Estimated Cost or Schedule in a range (e.g., $5 M to $7M or > 12 months) is acceptable
If more rows/space are needs please insert "rows" and/or adjust "height"

1 Experimental results of fuel irradiation experiments (compacts or pebbles) at representative burnup, temperature and fluence

Importance (H/M/L) *(Reference)*	Knowledge (K/P/U) *(Reference)*	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Validation Data Required (What kind?)	Alternative Model(s)
H	P	0.5	12	isotopic analysis of a pebble irradiated to 15% FIMA in HFR (ongoing)		No

2 Experimental results of decay heat at short term (<100 hours) for representative fuel composition and burnup.

Importance (H/M/L) *(Reference)*	Knowledge (K/P/U) *(Reference)*	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Validation Data Required (What kind?)	Alternative Model(s)
H	P	1.5	24			

VHTGR Computer Code / Modeling Survey Form

NGNP VHTGR Computer Code R&D Survey Form *(Design Baseline is listed in References)*

WBS # *(References)*	2.2.4.5	WBS Title	Code Qualification
Subject Matter Expert Name	Denis Verrier	Email	denis.verrier@areva.com
Phone	33 (0)4 72 74 88 96		
Organization	AREVA NP		

Computer Code Usage Category (Please check all applicable)

Please check all applicable, if existing code please provide code name with version

Reactor System Analysis		Autonomous Control	
Neutronics	NEPHTYS	Materials Analysis	
Thermal-Hydraulic		Structure Analysis	
Severe Accident		PCS Analysis	
FP Transport		Heat Exchanger Analysis	
Containment Analysis		Human Factor Simulation	
PRA		Economic	
Fuel Performance			
(please add-on)			

Objectives of Modeling

- **Reactivity effects (first criticality, moderator and doppler coefficients, control rod worth, reactivity loss versus depletion)**
- **3D neutron flux and nuclear power distribution within the reactor core.**
- **3D burnup distribution**

Status of Computer Code

Please check applicable

Current Code Fully Applicable	Current Code Needs Modification	Current Code Needs Major Modification	Current Code Needs Validation	No Existing Code
X			X	

Current Technology Readiness Level

Identify TRL, including rationale

TRL	6 - Integrated Modeling (Prototype) Completed	*TRL Definitions are listed in at References* *If can't decide please describe in the rationale block; if more space is required please insert "rows" or adjust "height"*
Rationale & Assumptions		

R&D Needs (Description of Technical Approach)

Identify and characterize the needs for R&D work to resolve critical issues affecting design, fabrication, testing and operation of the VHTGR plant

Importance and Knowledge definitions are listed in References; Estimated Cost or Schedule in a range (e.g., $5 M to $7M or > 12 months) is acceptable
If more rows/space are needs please insert "rows" and/or adjust "height"

1 **Approach for qualification currently consists of comparisons against Monte-Carlo reference calculations and benchmarking against the few available experimental data (FSV, HTTR). Thus new dedicated critical experiments, with an asymptotic spectrum representative of the expected prismatic fuel assembly and core, with full access to pin-by-pin power distributions, and control rod and burnable poisons worths are needed.**

Importance (H/M/L) *(Reference)*	Knowledge (K/P/U) *(Reference)*	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Validation Data Required (What kind?)	Alternative Model(s)
H	U	Same data as MCNP		FSV and HTTR first criticality		No

2 **Experimental data of neutronic characteristics (spectrum, fission and capture rates) at the interface between a prismatic fuel assembly and a graphite reflector assembly.**

Importance (H/M/L) *(Reference)*	Knowledge (K/P/U) *(Reference)*	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Validation Data Required (What kind?)	Alternative Model(s)
H	U	Same data as MCNP				

VHTGR Computer Code / Modeling Survey Form

NGNP VHTGR Computer Code R&D Survey Form *(Design Baseline is listed in References)*

WBS # *(References)*	2.2.4.5	WBS Title	Code qualification		
Subject Matter Expert Name	Denis Verrier	Email	denis.verrier@areva.com	Phone	33 (0)4 72 74 88 96
Organization	AREVA NP				

Computer Code Usage Category (Please check all applicable)

Please check all applicable, if existing code please provide code name with version

Reactor System Analysis	MANTA	Autonomous Control	
Neutronics		Materials Analysis	
Thermal-Hydraulic		Structure Analysis	
Severe Accident		PCS Analysis	
FP Transport		Heat Exchanger Analysis	
Containment Analysis		Human Factor Simulation	
PRA		Economic	
Fuel Performance			
(please add-on)			

Objectives of Modeling

Calculation of main system parameters (temperature, pressure, flowrate) of the HTR plant during all transient (normal, abnormal) when the primary coolant flows in forced convection in order to define plant operation and control and to provide load data for primary components. Possibility to calculate generalized natural convection.

Status of Computer Code

Please check applicable

Current Code Fully Applicable	Current Code Needs Modification	Current Code Needs Major Modification	Current Code Needs Validation	No Existing Code
X			X	

Current Technology Readiness Level

Identify TRL, including rationale

TRL	6 - Integrated Modeling (Prototype) Completed	*TRL Definitions are listed in at References* *If can't decide please describe in the rationale block; if more space is required please insert "rows" or adjust "height"*
Rationale & Assumptions		

R&D Needs (Description of Technical Approach)

Identify and characterize the needs for R&D work to resolve critical issues affecting design, fabrication, testing and operation of the VHTGR plant

Importance and Knowledge definitions are listed in References; Estimated Cost or Schedule in a range (e.g., $5 M to $7M or > 12 months) is acceptable
If more rows/space are needs please insert "rows" and/or adjust "height"

1 **Global validation of MANTA currently consists of code-to-code benchmarking: comparisons with CATHARE from CEA (France), LEDA from EDF (France), ASURA from MHI (Japan), REALY2 from GA (USA) and RELAP5-3D from INL (USA) have already shown good agreement. Qualification against experimental data is also progressing (EVO loop, HE-FUS3 loop and PBMM). Nevertheless additional benchmarks against experimental data are required. Some facilities which could provide valuable data have been identified (non exhaustive): namely, HTTR reactor in Japan, HTR10 reactor in China, SBL-30 loop in the USA (SNL). The qualiification of component models will follow from the qualification tests of the components. The core model qualification follows from comparison with a detailed core calculation.**

Importance (H/M/L) *(Reference)*	Knowledge (K/P/U) *(Reference)*	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Validation Data Required (What kind?)	Alternative Model(s)
H	P	0.3	6	Comparison with other codes and with experimental results	Additional experimental data from HTTR and HTR-10 safety tests and from SBL-30 loop	

2

Importance (H/M/L) *(Reference)*	Knowledge (K/P/U) *(Reference)*	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Validation Data Required (What kind?)	Alternative Model(s)

VHTGR Computer Code / Modeling Survey Form

NGNP VHTGR Computer Code R&D Survey Form *(Design Baseline is listed in References)*

WBS # *(References)*	2.2.4.5	WBS Title	Code Qualification
Subject Matter Expert Name	Denis Verrier	Email	denis.verrier@areva.com
Phone	33 (0)4 72 74 88 96		
Organization	AREVA NP		

Computer Code Usage Category (Please check all applicable)

Please check all applicable, if existing code please provide code name with version

Reactor System Analysis		Autonomous Control	
Neutronics		Materials Analysis	
Thermal-Hydraulic		Structure Analysis	
Severe Accident		PCS Analysis	
FP Transport		Heat Exchanger Analysis	
Containment Analysis		Human Factor Simulation	
PRA		Economic	
Fuel Performance			
Coupled neutronics/TH	CABERNET (= NEPHTYS / STAR-CD)		

Objectives of Modeling

Reactivity, power and temperature, burnup and fluence distribution calculation in steady state and transient conditions for block type cores (input to fuel performance assessment).

Status of Computer Code

Please check applicable

Current Code Fully Applicable	Current Code Needs Modification	Current Code Needs Major Modification	Current Code Needs Validation	No Existing Code
X	X		X	

Current Technology Readiness Level

Identify TRL, including rationale

TRL	6 - Integrated Modeling (Prototype) Completed	*TRL Definitions are listed in at References* *If can't decide please describe in the rationale block; if more space is required please insert "rows" or adjust "height"*
Rationale & Assumptions		

R&D Needs (Description of Technical Approach)

Identify and characterize the needs for R&D work to resolve critical issues affecting design, fabrication, testing and operation of the VHTGR plant

Importance and Knowledge definitions are listed in References; Estimated Cost or Schedule in a range (e.g., $5 M to $7M or > 12 months) is acceptable
If more rows/space are needs please insert "rows" and/or adjust "height"

1 **Experimental data of coupled power and temperature distributions obtained on representative fuel assembly geometry. If not achievable before NGNP: (a) Partial qualification data (e.g. burn-up measurements on fuel columns after irradiation in HTTR, which can provide a code/experiment comparison on the axial power distribution on a cycle, certainly different with and without temperature feedback); (b) Additional power margins will be necessary for initial operation of NGNP, to account for the uncertainty on the coupled neutronics-thermo-fluid dynamics calculation; (c) Need to provide in-core measurements of power and temperature distributions in NGNP for qualification of coupled calculations and therefore for allowing reaching nominal power; (d) R&D needs for developing appropriate sensors for in-core measurements (never performed in HTRs)**

Importance (H/M/L) *(Reference)*	Knowledge (K/P/U) *(Reference)*	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Validation Data Required (What kind?)	Alternative Model(s)
H	U					No

2

Importance (H/M/L) *(Reference)*	Knowledge (K/P/U) *(Reference)*	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Validation Data Required (What kind?)	Alternative Model(s)

VHTGR Computer Code / Modeling Survey Form

NGNP VHTGR Computer Code R&D Survey Form *(Design Baseline is listed in References)*

WBS # *(References)*	2.2.4.5	WBS Title	Code qualification
Subject Matter Expert Name	Denis Verrier	Email	denis.verrier@areva.com
Phone	33 (0)4 72 74 88 96		
Organization	AREVA NP		

Computer Code Usage Category (Please check all applicable)
Please check all applicable, if existing code please provide code name with version

Reactor System Analysis		Autonomous Control	
Neutronics		Materials Analysis	
Thermal-Hydraulic	STAR-CD	Structure Analysis	
Severe Accident		PCS Analysis	
FP Transport		Heat Exchanger Analysis	
Containment Analysis		Human Factor Simulation	
PRA		Economic	
Fuel Performance			
(please add-on)			

Objectives of Modeling

Determination of: (a) thermal loadings on the components (vessels, internals, fuel…) during normal or upset conditions, (b) the thermal behavior of the core, (c) the mixing inside the primary system, (d) head losses and performances of components, (e flow repartition across the components, (f) pressure shock waves.

Status of Computer Code
Please check applicable

Current Code Fully Applicable	Current Code Needs Modification	Current Code Needs Major Modification	Current Code Needs Validation	No Existing Code
X			X	

Current Technology Readiness Level

Identify TRL, including rationale

TRL	6 - Integrated Modeling (Prototype) Completed	*TRL Definitions are listed in at References* *If can't decide please describe in the rationale block; if more space is required please insert "rows" or adjust "height"*
Rationale & Assumptions		

R&D Needs (Description of Technical Approach)
Identify and characterize the needs for R&D work to resolve critical issues affecting design, fabrication, testing and operation of the VHTGR plant

Importance and Knowledge definitions are listed in References; Estimated Cost or Schedule in a range (e.g., $5 M to $7M or > 12 months) is acceptable
If more rows/space are needs please insert "rows" and/or adjust "height"

1 **Qualification of conduction cooldown models on representative geometry, materials and temperature**

Importance (H/M/L) *(Reference)*	Knowledge (K/P/U) *(Reference)*	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Validation Data Required (What kind?)	Alternative Model(s)
H	P	1	18			

2 **Qualification of turbulence and mixing on representative mock-ups in critical areas (lower and upper reactor plena, hot gas duct, core bypass, IHX collectors…**

Importance (H/M/L) *(Reference)*	Knowledge (K/P/U) *(Reference)*	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Validation Data Required (What kind?)	Alternative Model(s)
H	P	0.5	18			

3 **Qualification of oxidation models**

Importance (H/M/L) *(Reference)*	Knowledge (K/P/U) *(Reference)*	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Validation Data Required (What kind?)	Alternative Model(s)
H	P	0.3	12	Several tests performed with different graphite grades at CEA and FZJ. NACOK experiments within the European RAPHAEL project (coupling of graphite models with thermo-fluid dynamic behavior).	Need of new tests with selected graphite grades in representative operating conditions	

C.2.5 WBS C Q 10 2 22 PCF 2.4 Power Conversion System

WBS C Q 10 2 22 PCF 2.4.2 Brayton Cycle

WBS C Q 10 2 22 PCF 2.4.2.2 Turbo-machinery

He/N2 Turbine (2.4.2.2.1)

VHTGR H/W Survey Form

NGNP VHTGR H/W R&D Survey Form ***(Design Baseline is listed in References)***

WBS # *(References)*	**2.4.2.2.1**	**WBS Title**	**HT He/N_2 Turbine**		
Subject Matter Expert Name		**Email**		**Phone**	
Organization	**MHI**				

Current Technology Readiness Level
Identify TRL, including rationale

TRL	**7**	*TRL Definitions are listed in at References* *If can't decide please describe in the rationale block; if more space is required please insert "rows" or adjust "height"*
Rationale & Assumptions	**There are three different design parameters in ANTARES design concept from the commercially available GT: working fluid, inlet pressure and pressure ratio. For ANTARES design the working fluid will be 20% He and 80% N_2, which will be different from most GT working fluid of air. Special design for different working fluid exists, but experience of actual machine using He/N_2 mixed gas is not. The high inlet pressure and low pressure ratio of ANTARES design, 4.94 MPa vs. 1.6 MPa to 3.2 MPa and 2.3 vs. 16 or 32, respectively, technically, should not pose any critical issues, but no existing data either to support it. Some modifications of the current GT design may be needed, so is the follow-on qualification. Risk on nitriding needs R&D efforts.**	

R&D Needs
Identify and characterize the needs for R&D work to resolve critical issues affecting design, fabrication, testing and operation of the VHTGR plant

Importance and Knowledge definitions are listed in References; Estimated Cost or Schedule in a range (e.g., $5 M to $7M or > 12 months) is acceptable
If more rows/space are needs please insert "rows" and/or adjust "height"

1	**Nitriding of metals will occur when exposed to hot nitrogen. This nitriding process tends to embrittle metals which could lead to failures of turbine blades and pressure boundaries such as boiler tubes, gas shells, etc. Propose to perform experimental approach for nitriding of potential PCS materials and how temperature effects on nitriding.**

Importance (H/M/L)	Knowledge (K/P/U)	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Recommendation for Government Lead & Rationale *(Reference)*
H	P	5	18	None	Existing	No

2	

Importance (H/M/L)	Knowledge (K/P/U)	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Recommendation for Government Lead & Rationale *(Reference)*

He/N2 Compressor (2.4.2.2.2)

VHTGR H/W Survey Form							
NGNP VHTGR H/W R&D Survey Form ***(Design Baseline is listed in*** *References* ***)***							
WBS # *(References)*	**2.4.2.2.2**	**WBS Title**	**He/N_2 Compressor**				
Subject Matter Expert Name		**Email**		**Phone**			
Organization	**MHI**						
Current Technology Readiness Level Identify TRL, including rationale							
TRL	**7**	*TRL Definitions are listed in at References* *If can't decide please describe in the rationale block; if more space is required please insert "rows" or adjust "height"*					
Rationale & Assumptions	**Compressor is proven technology, but to attain higher efficiency some blade design modifications will be required.**						
R&D Needs Identify and characterize the needs for R&D work to resolve critical issues affecting design, fabrication, testing and operation of the VHTGR plant							
Importance and Knowledge definitions are listed in References; Estimated Cost or Schedule in a range (e.g., $5 M to $7M or > 12 months) is acceptable *If more rows/space are needs please insert "rows" and/or adjust "height"*							
1	**The R&D for the blades performance should be required in order to attain higher efficiency.**						
Importance (H/M/L)	**Knowledge (K/P/U)**	**Estimated Cost ($M)**	**Estimated Schedule (Months)**	**Predecessor R&D, if any?**	**Facility Availability (exiting, modified, new) (which existing facilities?)**	**Recommendation for Government Lead & Rationale** *(Reference)*	
M	**P**	**5**	**12**	**None**	**Existing**	**No**	
2							
Importance (H/M/L)	**Knowledge (K/P/U)**	**Estimated Cost ($M)**	**Estimated Schedule (Months)**	**Predecessor R&D, if any?**	**Facility Availability (exiting, modified, new) (which existing facilities?)**	**Recommendation for Government Lead & Rationale** *(Reference)*	

Generator and Electrical Equipments (2.4.2.3)

VHTGR H/W Survey Form

NGNP VHTGR H/W R&D Survey Form *(Design Baseline is listed in References)*

WBS # *(References)*	2.4.2.3	WBS Title	Generator and Electrical Equipments
Subject Matter Expert Name		Email	Phone
Organization	MHI		

Current Technology Readiness Level
Identify TRL, including rationale

TRL	9	*TRL Definitions are listed in at References* *If can't decide please describe in the rationale block; if more space is required please insert "rows" or adjust "height"*
Rationale & Assumptions	Generator and electrical equipment is proven technology	

R&D Needs
Identify and characterize the needs for R&D work to resolve critical issues affecting design, fabrication, testing and operation of the VHTGR plant

Importance and Knowledge definitions are listed in References; Estimated Cost or Schedule in a range (e.g., $5 M to $7M or > 12 months) is acceptable
If more rows/space are needs please insert "rows" and/or adjust "height"

1	None

Importance (H/M/L)	Knowledge (K/P/U)	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Recommendation for Government Lead & Rationale *(Reference)*

2	

Importance (H/M/L)	Knowledge (K/P/U)	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Recommendation for Government Lead & Rationale *(Reference)*

He Cycle Control and Ducting (2.4.2.4)

VHTGR H/W Survey Form								
NGNP VHTGR H/W R&D Survey Form *(Design Baseline is listed in References)*								
WBS # *(References)*		**2.4.2.4**			**WBS Title**	**He/N_2 Cycle Control and Ducting**		
Subject Matter Expert Name				**Email**			**Phone**	
Organization		**MHI**						
Current Technology Readiness Level Identify TRL, including rationale								
TRL	**9**	*TRL Definitions are listed in at References* *If can't decide please describe in the rationale block; if more space is required please insert "rows" or adjust "height"*						
Rationale & Assumptions	**He/N2 cycle control and ducting are proven technology.**							
R&D Needs Identify and characterize the needs for R&D work to resolve critical issues affecting design, fabrication, testing and operation of the VHTGR plant								
Importance and Knowledge definitions are listed in References; Estimated Cost or Schedule in a range (e.g., $5 M to $7M or > 12 months) is acceptable *If more rows/space are needs please insert "rows" and/or adjust "height"*								
1	**Nitriding concern on using high temperature He/N2 mixed gas and R&D needs have been identified in WBS 3.5.2.2.1.**							
Importance (H/M/L)	**Knowledge (K/P/U)**	**Estimated Cost ($M)**	**Estimated Schedule (Months)**	**Predecessor R&D, if any?**	**Facility Availability (exiting, modified, new) (which existing facilities?)**	**Recommendation for Government Lead & Rationale** *(Reference)*		
2								
Importance (H/M/L)	**Knowledge (K/P/U)**	**Estimated Cost ($M)**	**Estimated Schedule (Months)**	**Predecessor R&D, if any?**	**Facility Availability (exiting, modified, new) (which existing facilities?)**	**Recommendation for Government Lead & Rationale** *(Reference)*		

Heat Recovery Steam Generator (2.4.3)

VHTGR H/W Survey Form						
NGNP VHTGR H/W R&D Survey Form *(Design Baseline is listed in References)*						
WBS # *(References)*	2.4.3	WBS Title	HRSG			
Subject Matter Expert Name		Email		Phone		
Organization	MHI					
Current Technology Readiness Level Identify TRL, including rationale						
TRL	8	*TRL Definitions are listed in at References* *If can't decide please describe in the rationale block; if more space is required please insert "rows" or adjust "height"*				
Rationale & Assumptions	HRSG is proven technology, however, system optimization, to determine the optimal pinch-point, has not been completed.					
R&D Needs Identify and characterize the needs for R&D work to resolve critical issues affecting design, fabrication, testing and operation of the VHTGR plant						
Importance and Knowledge definitions are listed in References; Estimated Cost or Schedule in a range (e.g., $5 M to $7M or > 12 months) is acceptable *If more rows/space are needs please insert "rows" and/or adjust "height"*						
1	None					
Importance (H/M/L)	Knowledge (K/P/U)	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Recommendation for Government Lead & Rationale *(Reference)*
2						
Importance (H/M/L)	Knowledge (K/P/U)	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Recommendation for Government Lead & Rationale *(Reference)*

Steam Cycle (2.4.4)

VHTGR H/W Survey Form							
NGNP VHTGR H/W R&D Survey Form *(Design Baseline is listed in References)*							
WBS # *(References)*	2.4.4		WBS Title	Steam Cycle			
Subject Matter Expert Name				Email		Phone	
Organization	MHI						
Current Technology Readiness Level Identify TRL, including rationale							
TRL	9		*TRL Definitions are listed in at References* *If can't decide please describe in the rationale block; if more space is required please insert "rows" or adjust "height"*				
Rationale & Assumptions	Steam cycle subsystem is proven technology						
R&D Needs Identify and characterize the needs for R&D work to resolve critical issues affecting design, fabrication, testing and operation of the VHTGR plant							
Importance and Knowledge definitions are listed in References; Estimated Cost or Schedule in a range (e.g., $5 M to $7M or > 12 months) is acceptable *If more rows/space are needs please insert "rows" and/or adjust "height"*							
1	None						
Importance (H/M/L)	Knowledge (K/P/U)	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Recommendation for Government Lead & Rationale *(Reference)*	
2							
Importance (H/M/L)	Knowledge (K/P/U)	Estimated Cost ($M)	Estimated Schedule (Months)	Predecessor R&D, if any?	Facility Availability (exiting, modified, new) (which existing facilities?)	Recommendation for Government Lead & Rationale *(Reference)*	

C.3 Summary of Survey Form Information

A summary of survey form information appears in Table C-2. For each identified R&D need, the table provides the Technical Readiness Level (TRL), importance of the R&D, knowledge state, and an estimated cost and schedule.

Table C-2: VHTGR Hardware and Computer Code TRL Summary

WBS		SME	Org.	TRL	Importance (H/M/L)	Knowledge (K/P/U)	Estimated Cost (M)	Estimated Schedule (Months)
C-Q-10-2-22-NHS								
2 2 1	**Fuel**							
2 2 1 1	**Fuel development and qualification**							
2 2 1 1 1	Kernel materials	John Mayer / Jeff Halfinger	Areva / BWXT	4	M	U	$5 - 10	12
2 2 1 1 2	Kernel manufacturing	John Mayer / Jeff Halfinger	Areva / BWXT	4	H	P	$15 - 20	24
2 2 1 1 3	Coating materials	John Mayer / Jeff Halfinger	Areva / BWXT	7				
2 2 1 1 4	Coating manufacturing	John Mayer / Jeff Halfinger	Areva / BWXT	4	H	p	$5-20 (if new furnace is needed)	6 - 24
2 2 1 1 5	Compact materials	John Mayer / Jeff Halfinger	Areva / BWXT	3	H	P	$60	36
2 2 1 1 6	Compact manufacturing	John Mayer / Jeff Halfinger	Areva / BWXT	3	H	P	$40	36
2 2 1 1 7	Quality control methods	John Mayer / Jeff Halfinger	Areva / BWXT	6	M	K	$2	18
2 2 1 1 8	Inspection techniques	John Mayer / Jeff Halfinger	Areva / BWXT	4	H	P/U	$25	24
2 2 1 1 9	Fuel mass production	John Mayer / Jeff Halfinger	Areva / BWXT	3	H	P	$30	30
2 2 2	**Materials Development & Qualification**							
	Tribology	Bernard Riou	Areva	5	M	P	$0.5	18
2 2 2 1	**Metallic Materials**							
2 2 2 1 1	Vessel Materials (High temperature steel)	Sophie Dubiez Le Goff	Areva	6	H	P	$4	72
2 2 2 1 2	Reactor Internals	Sophie Dubiez Le Goff	Areva	6	M	P	$0.5	24
2 2 2 1 3	IHX	Sophie Dubiez Le Goff	Areva	6	H	P	$4	30
2 2 2 2	**Ceramics**							
2 2 2 2 2	Reactor Internals	Odile Gelineau	Areva	2	M	U	$4	54
2 2 2 2 3	Control rod sheaths (Carbon-Carbon composites)	Odile Gelineau	Areva	2				
2 2 2 3	**Graphite Materials (initial)**	Odile Gelineau	Areva	7	H	P	$6	54+
	Graphite Materials (subsequent)						$14	
2 2 3	**Components Testing**							
2 2 3 2	**Core Internal Structures and Supports**							
2 2 3 3	**Circulators**	Eric Breuil	Areva	8	M	K	$2	18
2 2 3 4	**IHX (Tube)**	Eric Breuil	Areva	5	H	P	10 MW loop - facility: $72 - 112 test article: $1 testing: $7	42, including 12 months testing
	IHX (Plate)	Eric Breuil	Areva	2	H	U	1) 1MW loop: $20 2) 10 MW loop - facility: $72 - 112 test article: $1 testing: $7	42, including 12 months testing
2 2 3 5	**Isolation Valves**	Bernard Riou	Areva	7	H	P	$3.5 (total)	12
2 2 3 6	**Helium Purification System**	John Mayer	Areva	8	M	P	$0.20	6
2 2 3 7	**Shutdown Cooling System**	John Mayer	Areva	7				
2 2 3 8	**Fuel Handling System**	John Mayer	Areva	2	M	K	$0.5	12
2 2 3 9	**Neutron Control System Drive Mechanism**							
2 2 3 10	**RCCS**	John Mayer	Areva	2	H	P	$1	24
2 2 3 11	**Plant Control, Investment Protection, and Safety Protection**							
2 2 3 12	**Hot Gas Duct**	Bernard Riou	Areva	6	H	P	$4.5 (total)	
2 2 3 13	**Instrumentation**	Bernard Riou	Areva	7	H	P	$2	
2 2 4	**Computer Codes & Methods Development & Qualification**							
2 2 4 4	**Code Development**							
2 2 4 4 1	Neutronics codes (MCNP, MONTEBURNS, NEPHTYS)	Denis Verrier	Areva	6	H	P	$0.5	12
2 2 4 4 2	Thermal/ Hydraulics/ Pneumatics Codes	Denis Verrier	Areva	6	H	P	$0.20	12
2 2 4 4 3	Other codes	Denis Verrier	Areva	4	M	P	$6	60
	Mechanical Analysis			4	M	P	$1.0	18
2 2 4 4 5	Fuel Performance Models and Codes	Denis Verrier	Areva	5	H	P	$1.5	24
	Develop fuel performance models	Denis Verrier	Areva					
2 2 4 5	**Code Qualification**							
	ATLAS	Denis Verrier	Areva	5	H	P	$5	30
	MCNP	Denis Verrier	Areva	6	H	U	$8	36
	MONTERBURNS	Denis Verrier	Areva	6	H	P	$2	24
	NEPHTYS	Denis Verrier	Areva	6	H	U	Same data as for MCNP	
	MANTA	Denis Verrier	Areva	6	H	P	$0.3	6
	RELAP			6	H	P		
	CABERNET	Denis Verrier	Areva	6	H	U	During Commissioning Tests	
	STAR-CD	Denis Verrier	Areva	6	H	P	$1.8 (total)	18
C-Q-10-2-22-PCP								
2 4	**Power Conversion System R&D**							
2 4 2	**Brayton Cycle**							
2 4 2 2	**Turbo-machinery**		MHI	7	H	P	$10	18
2 4 2 3	**Generator and Electrical Equipments**		MHI	9				
2 4 2 4	**He Cycle Control & Ducting**		MHI	9				
2 4 3	**Heat Recovery Steam Generator**		MHI	8				
2 4 4	**Steam Cycle**		MHI	9				

APPENDIX D: SYSTEM ANALYSIS OF HIGH TEMPERATURE STEAM ELECTROLYSIS

D.0 SYSTEM ANALYSIS

The thermal and electrical energy from a nuclear power plant can be used to produce hydrogen by thermochemical and electrochemical means. The sulfur iodine system [1] is the traditional thermochemical approach, while high temperature steam electrolysis [1-4] is an example of an electrochemical approach. These are two competing processes for hydrogen generation for the NGNP project. A brief outline of each process is given here.

The sulfur iodine process revolves around three reactions coupled together to produce hydrogen and oxygen. The first reaction is the Bunsen reaction in which two water molecules react with SO2 and I2 to form sulfuric acid and two molecules of HI.

$I2 + SO2 + 2\ H2O \rightarrow H2SO4 + 2\ HI$

The reaction is exothermic and usually run at temperatures around 120°C. Two separate thermal decomposition reactions follow in which the hydrogen iodide and sulfuric acid are decomposed to form hydrogen and oxygen, respectively. The thermal decomposition of hydrogen iodide to H2 and I2 can be accomplished ~ 300°C, while the decomposition temperature for oxygen liberation from sulfuric acid is much higher, ~800°C. Smith and Santangelo [2] offer a more detailed overview of the process as well as others [1, 3, 4].

High temperature steam electrolysis (HTSE) is usually conducted over solid oxide electrolytic cells (SOEC) [5]. The high operating temperatures reduces the minimum energy, the ΔG of reaction, required for the electrochemical reaction to take place. Steam is raised to temperatures of ~ 800°C and passed over the cathode where it is dissociated into its component atoms. Hydrogen is evolved at the cathode while oxygen is transported through a solid electrolyte and recombines at the anode. The anode is swept by a carrier gas to reduce the concentration of oxygen at the surface of the electrode. Heat can be recovered from the effluent gases.

Each process has it own keys to process design and integration with the NGNP plant and deserves a highly detailed study to understand the best mode of operation before comparing designs. For purposes of this section of the report, a cursory study was conducted on a process for high temperature steam electrolysis for hydrogen production. Simulations of a generic SOEC-based process were conducted with the identification and quantification of key process variables. Optimal conditions depend on the specific electrolyzer technology employed and thus an overall direction for optimal process conditions is given. The sulfur iodine process is not covered in this section.

D.1 Background

Electrolysis of water uses electricity to overcome the energy barrier for splitting the molecule into its atomic components and recombining them to form hydrogen and oxygen. Electrolysis can be done at room temperature or at elevated temperature, with liquid water or water vapor [6]. The electrolysis configuration depends on the operating temperature and phase of the water, liquid or vapor. The method examined here is high temperature steam electrolysis using solid oxide electrolytic cells (SOEC). The cells are composed of three active layers of ceramic material: two electrodes and a solid electrolyte between them.

Figure D-1 is a schematic of a cell cross section. Superheated steam with a small amount of hydrogen enters at the inlet of the cell. The steam adsorbs on the cathode and is split into positively charged hydrogen and negatively charged oxygen atoms. Two positively charged hydrogen ions then react on the surface, requiring two electrons supplied by the cathode, to form molecular hydrogen and desorb back into the vapor phase. Oxygen ions diffuse through the cathode, migrate through the solid electrolyte and recombine to form molecular oxygen at the anode liberating four electrons in the process. Oxygen then desorbs into the sweep gas stream, represented by steam in Figure D-1. The potential applied across the electrodes serves as the energy required to execute the sequence of steps.

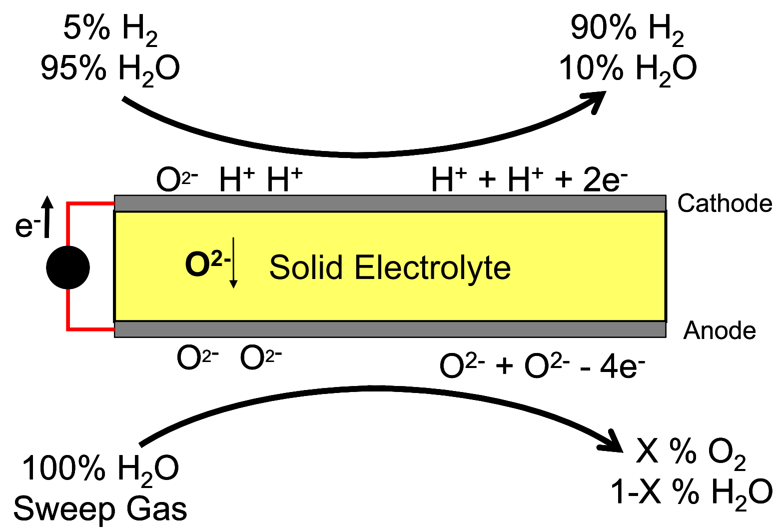


Figure D-1: Schematic of Solid Oxide Electrolytic Cell

Several to many single cells like that shown in Figure D-1 are assembled to form a single stack of cells. The stacks are modular in design and many would be assembled together to form a commercial electrolytic reactor. Each reactor would exhibit a composition, pressure and temperature profile along the flow direction. Several reactors would likely be needed to comprise a plant at the scale of hydrogen production in consideration here. Modeling of the cell, stack, reactor, and finally the plant adds additional layers of complexity when trying to understand the overall process design. The additional variable of changing the final oxygen/water vapor composition of the exiting sweep gas shown in Figure 1 further complicates the modeling efforts. Before delving into undo detail of building the overall model, the energy requirements to drive the reaction at the cell level need to first be understood.

Compositions and temperatures are likely to change over a given cell. These changes, while small, complicate the requirements on energy needed for the cell. Therefore, the over-idealized reaction of pure water to form separate streams of pure hydrogen and oxygen, shown below, at a pressure of 0.9MPa will be used to illustrate the change in Gibbs energy and enthalpy with temperature.

Pure $H_2O_{(g)}$ $\rightarrow$ Pure $H_{2\,(g)}$ + Pure $O_{2\,(g)}$

ΔH and ΔG at 0.9MPa and temperature specified

The minimum electrical energy required to operate a cell is the Gibbs energy change for the reaction, which, of course, is a function of temperature, composition, and pressure. Figure D-2 shows the Gibbs energy, the blue squares on the figure, as a function of temperature. The Gibbs energy decrease is approximately linearly between 300 and 1000°C. The Enthalpy of reaction, the red points on the figure, exhibits a weak temperature dependence and increases slightly over the temperature range. The difference between the curves is TΔS, a positive quantity for the reaction since the enthalpy is greater than the Gibbs energy. Points a through d are marked by black dots on the graph and will be used to illustrate several cell operating points.

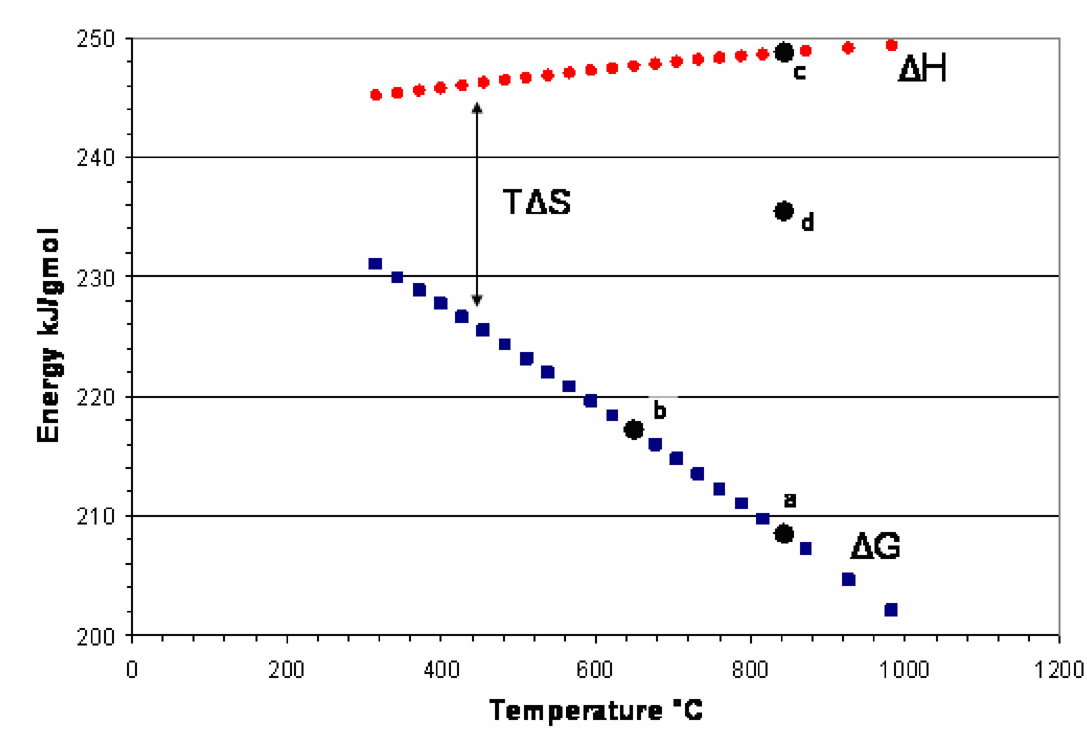


Figure D-2: Enthalpy and Gibbs Free Energy of Electrolysis Reaction Dependence on Temperature at an Operating Pressure of 0.9Mpa

Point a on Figure D-2 represents the Gibbs energy change, ΔG, for reaction at 843°C, 208.5 kJ/gmol. This would represent a minimum voltage of 1.080V applied to the electrolyzer if operated at 843°C and 0.9MPa. Point b represents the ΔG required at a lower temperature of 649°C, 217.2 kJ/gmol equivalent to 1.126V. The potential applied at point a conditions is less than the potential required at point b on Figure D-2. The voltages are both significantly lower than the 1.446V required for electrolysis at ambient conditions, 25°C and 0.1MPa. The advantage of operating at a lower voltage comes when calculating the overall power supplied to the cell. Power is the product of voltage and current. Two electrons per H2 molecule are also needed to complete the reaction and therefore current is closely tied to the production rate of hydrogen. This means the cell with the lowest voltage will generate the most hydrogen for a fixed amount of power supplied to a cell. Therefore, running at higher temperatures reduces the voltage needed to operate the cell and increases the amount of hydrogen that can be produced from the cell.

The energy supplied at point a, 208.5 kJ/gmol, however, is less than the enthalpy for reaction, which is represented by point c on Figure 2, 248.8 kJ/gmol at 843°C. Thus, operating at point a will result in cooling of the cell upon reaction. Conversely, operation of the cell at point c will result in running isothermally. The voltage equivalent of the 248.8 kJ/mol of energy at point c is 1.289V. This is 0.209V higher than the conditions for point a, 1.080V. The additional 0.209V applied to the cell is dissipated as heat within the cell which maintains the cell as well as the reactant/product mixture temperatures at 843°C. The voltage applied to the electrolyzer to run isothermally is commonly referred to as the thermal neutral voltage.

Point d represents operating at conditions between the minimum potential and the thermal neutral voltage, nominally 1.220V or 235.5 kJ/gmol. The potential is 0.14V or 27 kJ/gmol above the minimum energy of point a and is therefore dissipated as heat within the cell. However, the potential is 0.051V below the thermal neutral voltage. Therefore, the temperature of the cell as well as the reactant/product mixture will decrease. The temperature, however, will be between that of point a and point b.

Additional complexities in composition, temperature, and pressure fluctuations in real cells do not change the overall guiding principal outlined in the idealized case represented by points a through d on Figure D-2. The voltage applied to stacks of cells, however, would change based on local conditions within the reactor. This level of detail was left out of the models presented here. The focus of this study was on the overall process and the identification of and trends exhibited by key parameters in the overall process. Therefore, several assumptions on the electrolysis reactor as well as other parts of the process were made for simplification of the process model and are covered in the next section.

D.2 Assumptions for SOEC Process

The overall hydrogen plant is comprised of a nuclear island and a hydrogen production island. The process design was based on several key assumptions. The nuclear island was assumed to supply the thermal and electrical energy necessary to run the hydrogen production island. No other utilities, cooling water, water conditioning systems, pumps, compressors, etc., are assumed to be shared between the islands. There are no material flows between the two islands – only flow of thermal and electrical energy. The medium for thermal energy transport was assumed to be steam at a pressure of 2 MPa and 600°C, unless otherwise stated. This steam stream is similar to that of the IPST steam generated in the supercritical power conversion system (PCS) cycle outline in the Power Conversion System Special Study section of the Preconceptual Design Studies Report. Saturated water at slightly reduced pressure is returned to the nuclear island. This stream is self-contained within the nuclear loop; it transfers its thermal energy to another lower-pressure, 0.9 MPa, steam loop within the H2 plant island. A separate steam system utilizing the heat from the nuclear island supplied steam is contained within the hydrogen island.

Electrical energy is supplied to the hydrogen production island at the needed voltage with no losses associated with its transmission, or power conditioning (voltage adjustment, AC/DC conversion etc.). The nuclear power plant is designed to generate 565MWth of energy. This energy is fully utilized as thermal or electrical input into the hydrogen production island. Residual power, usually <0.5 MW, is assumed to be transmitted and sold as power to the grid. Only steady state operation has been modeled. The electrolyzer itself is modeled as a black box which converts 90% of the steam in the feed to hydrogen and oxygen at the specified inlet and outlet temperatures. Isothermal conditions were assumed in most of the cases; however, several cases were run at nonisothermal conditions.

The steam from the nuclear island is taken from the IPST reheat stream of similar conditions as defined in the Power Conversion System Special Study section of the Preconceptual Design Studies Report. Thus, steam taken as thermal input for the hydrogen production island reduces the net power output of the nuclear island. The energy balance for the hydrogen production island takes into account the loss of electrical power supplied by the nuclear island.

The overall H_2 production rate and nuclear island energy usage, both thermal and electric, was the main focus of this study with no detail given for a specific electrolyzer performance. This generalized electrolyzer basis has been used for all cases and data may not reflect the current operating conditions for the state of the art electrolyzer. Energy utilization will play a large role on the operating conditions of the electrolyzer which may take the current designs for HTSE electrolyzers outside feasible operating conditions. However, it can be used to set the direction of future R&D.

D.3 Process Design

The evaluation of key process variables within the process design linking the high temperature electrolysis based H_2 plant with a nuclear plant is the topic of this section. The process is similar to that evaluated by researchers at MIT [7, 8]. The cartoon in Figure 3 shows the overall flow of feed, sweep gas, and products in the system. Steam is supplied to the inlet of the electrolyzer with separate hydrogen and oxygen product streams leaving the electrolyzer. The products are then sent through heat recovery for generation of feed and sweep steam. Not all the heat can be usefully recovered by the heat recovery train. Cooling water is used to cool the product streams to ~38°C before each stream is sent to a knock out pot. The water removed in the knock out pots is then recycled to the beginning of the water system and is deareated before being preheated and sent back into the steam circuit. The resulting oxygen and water vapor stream from the knockout pot is vented to the atmosphere, but could be further dried and compressed for subsequent use. The hydrogen stream is sent to a drying system where the residual water is removed before being compressed to a pressure of 800 psia. A small stream of product hydrogen is recycled to the front of the electrolyzer to prevent oxidation of the electrolyzer cell.

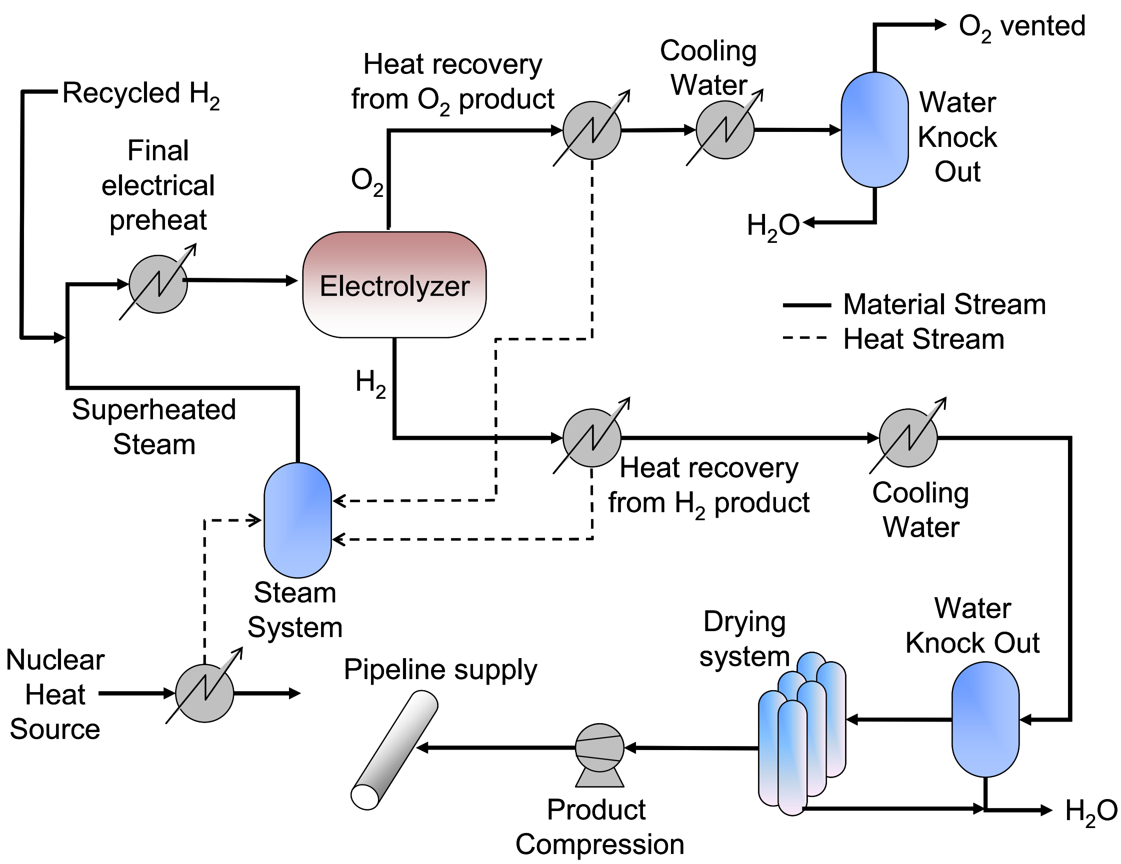


Figure D-3: High Temperature Steam Electrolysis Process Schematic

Both steam used for feed and sweep gas are generated in a common steam system. A portion of the heat, as described above, is supplied by heat recovery off the product streams. The product streams, however, do not possess the required amount and quality of heat needed to raise all the steam for the electrolyzer cell operation.

Nuclear steam is used to augment the required boiling duty, and represents the net thermal energy transfer from the nuclear plant to the H_2 plant. The remaining heat needed is supplied by resistively heating the steam to the desired electrolyzer cell inlet temperature. The remainder of electrical energy is used to power the electrolyzer, the boiler feed water pump, and the hydrogen product compressor.

A model of the process was created in ASPEN to simulate various operational cases. The inlet and outlet temperature of the SOEC and the composition of the sweep gas were the key process variables altered in the study. The subsequent change in the capacity of the plant, the balance of thermal and electrical energy supplied by the nuclear island, and the distribution of the heat in the heat recovery/steam generation system results are documented here. Furthermore, the energy balance from the converged simulation was used to estimate the operating voltage of the SOEC. Additional results using nuclear generated steam at a pressure of 2MPa and 538°C will be covered briefly.

D.4 Discussion/Results

A key input into understanding the overall process is the conversion of thermal energy into electrical energy within the power conversion system of the nuclear island. As indicated in the Power Conversion System Special Study section of the Preconceptual Design Studies Report, the conversion of thermal energy to electrical energy is less than 50% efficient. Therefore, thermal input is a relatively "cheaper" source of energy than electrical and is more preferable to use as an energy source for equivalent changes in enthalpy. However, the use of thermal energy may not be advisable if the steam is used for additional purposes such as preheat for make up water or deareator preheat. Additional use of the heat beyond the turbine effectively increases the energy penalty or "cost" of the 600°C steam used for heating within the hydrogen production island. Therefore, an estimate of 1 MWe in gross output of the plant for every 1 kg/s of steam usage was used in the process model. The steam supplied by the nuclear island for boiling duty was completely condensed in the exchanger in order to limit the thermal input required from the nuclear island. An approach temperature of ~17°C was used for the exchanger, which in turn sets the hydrogen plants steam pressure at 0.9 MPa. A subsequent drop of 0.2 MPa was assumed in the exchanger train resulting in 0.7 MPa operating pressure for the SOEC.

Twelve cases were simulated with the results of the simulations presented in Table D-1 and Table D-2. The operating temperature of the SOEC and the composition of the sweep gas were varied in cases 1 through 9. The operating temperatures used in the simulation were 600, 700, and 800°C while sweep gas compositions were varied from 25 to 50 and finally 75% O2. Cases 10-12 are tabulated results for nonisothermal operation of the SOEC cell at three different inlet and outlet temperatures. Table 1 results show result for the key process output variables, while the distribution of power is tabulated in Table D-2.

Table D-1: Process Conditions and Modeling Results*

	SOEC Inlet Temperature	SOEC Outlet Temperature	% O_2 in Sweep Gas	% Boiling duty from Nuclear Steam	% Boiling Duty from Process	Steam Temperature Exiting Process Heat Recovery	Cooling Water Duty‡	Electrolyzer Potential	H_2 Production Rate
Run#/Units	°C	°C	%	%	%	°C	MW	V	kg/s
1	800	800	25	56.5	43.5	384	49.6	1.288	1.62
2	800	800	50	67.5	32.5	561	19.4	1.288	1.78
3	800	800	75	75.0	25.0	604	8.0	1.288	1.85
4	700	700	25	64.4	35.6	347	50.4	1.285	1.65
5	700	700	50	73.7	26.3	458	19.6	1.285	1.80
6	700	700	75	79.9	20.1	532	8.0	1.285	1.86
7	600	600	25	72.1	27.9	311	51.3	1.282	1.68
8	600	600	50	79.7	20.3	400	19.8	1.282	1.82
9	600	600	75	84.8	15.2	460	8.1	1.282	1.87
10	800	600	25	84.8	15.2	460	8.2	1.226	1.87
11	800	400	25	94.2	5.8	314	8.2	1.165	1.89
12	600	400	25	94.2	5.8	314	8.1	1.221	1.89

*** Note** – All results shown are for a steam system pressure of 0.9MPa, 0.7MPa SOEC operating pressure, and nuclear steam at 2MPa and 600°C

† **Note** – The balance of the sweep gas is superheated steam, 0.9 MPa at specified inlet operating conditions

‡ **Note** – Trim cooler duty is the sum of the duties to cool both H_2 and O_2 product streams to ~37°C

A clear trend in operating temperature and sweep gas composition is seen by examining cases 1-9 in Table 1. The H2 production rate increases as operating temperature of the SOEC decreases from 800 to 700 to 600°C. Also, the H2 production rate increases with a decrease in percentage of steam in the sweep gas. Table 2 and Figure 4 show the trends in MW of energy consumed by thermal heating utilizing nuclear steam and resistive heating of the process feed for cases 1-9. Together, the results illustrate what is occurring as operating temperature and sweep gas composition are changed.

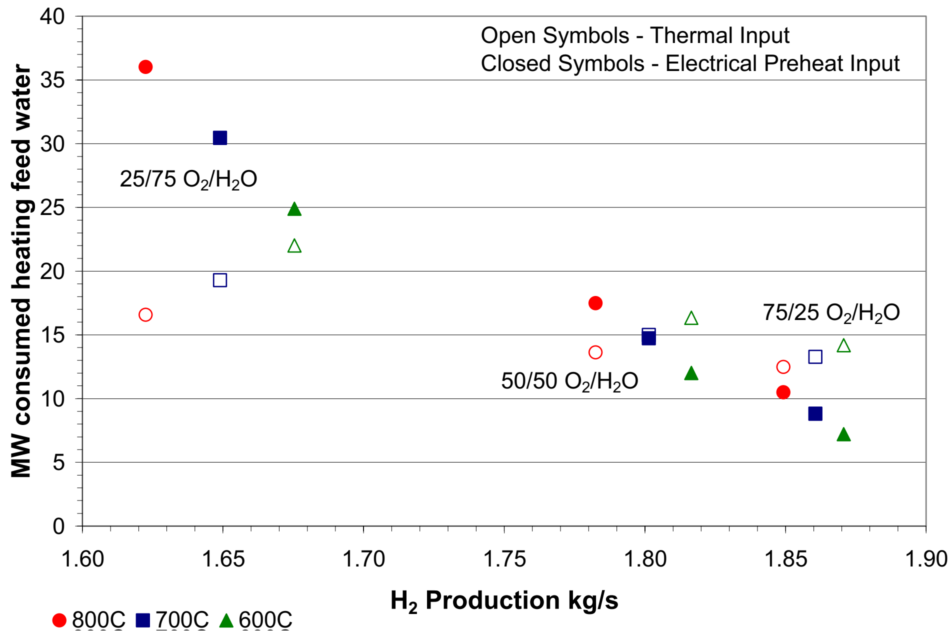


Figure D-4: Nuclear Thermal and Electrical Energy Consumed in Boiling and Preheating Feed

Three groupings of 6 points each are shown in Figure 4. The first set of points represent results for MW consumed for a sweep gas composition of 25% O_2, 75% H_2O at 600, 700, and 800C. The second set of 6 points is for the same set of temperatures, but for a sweep gas composition containing 50% H2O and 50% O_2. The third set of 6 points is for a sweep gas composition of 25% H_2O and 75% O_2 at the three operating temperatures. The open symbols in the figure represent the MW consumed by thermal heating using nuclear steam while the closed symbols are for the MW consumed by resistive preheating of the feed and sweep. These results are also tabulated for reference in Table D-2.

Table D-2: Distribution of Power within Process*

	BFW Pump Power	Resistive Preheat	Electrolyzer	Electrolyzer + Resistive Preheat	H_2 Compressor	Total H_2 Island Power Consumed	Nuclear Thermal Heat†	Net Nuclear Island Power Produced	Residual Power to Grid
Run #/Units	KW	MWe	MWe	MWe	MWe	MWe	MWe	MWe	MWe
1	46.8	36.0	201.7	237.7	11.0	248.7	16.6	249.0	0.24
2	31.7	17.5	221.6	239.1	12.1	251.2	13.6	251.9	0.15
3	26.1	10.5	229.9	240.4	12.5	252.9	12.5	253.1	0.77
4	47.5	30.4	204.5	234.9	11.2	246.2	19.3	246.3	0.10
5	32.0	14.7	223.4	238.1	12.2	250.4	15.0	250.6	0.22
6	26.2	8.8	230.8	239.6	12.6	252.2	13.3	252.3	0.09
7	48.3	24.9	207.2	232.1	11.3	243.5	22.0	243.5	0.02
8	32.3	12.0	224.7	236.7	12.3	249.0	16.3	249.2	0.19
9	26.2	7.2	231.4	238.6	12.7	251.3	14.2	251.4	0.08
10	26.4	17.4	221.2	238.6	12.7	251.3	14.2	251.4	0.10
11	26.7	24.2	212.5	236.7	12.8	249.5	15.9	249.7	0.15
12	26.7	14.0	222.7	236.7	12.8	249.5	15.9	249.7	0.15

* **Note** – All results shown are for a steam system pressure of 0.9MPa, 0.7MPa SOEC operating pressure, and nuclear steam at 2MPa and 600°C

† **Note** –Converted from MWth to MWe based on results on Power Conversion System Special Study section of the Preconceptual Design Studies Report

A decreasing trend is clearly seen in the energy required to resistively preheat the feed and sweep steam streams with decreasing SOEC operating temperature. Also, a slight increase is seen in the thermal MW usage with decreasing cell operating temperature. This trend continues in both the 50/50 and 75/25% oxygen/water sweep gas compositions. Also, Table 1 results show an increase in the percentage of boiling duty done by the nuclear steam as well as a decrease in process steam superheat temperature prior to the resistive preheat coil with decreasing operating temperature. The results are due to a reduction in quality of heat available in the process heat recovery train. A lower outlet temperature of the SOEC means less heat is available for superheating and boiling duty. Thus, the reduction in operating temperature forces more of the boiling duty to nuclear steam boiler. The lower superheat temperature does not adversely effect the MW consumed by the resistive preheat coil since the ΔT across the preheat coil is less for lower operating temperatures.

Similar trends with slight changes in the magnitude of the trends are seen as the percentage of oxygen in the sweep gas is increased. Figure 4 also clearly illustrates that higher O2% in the sweep gas increases production at a given operating temperature. Increasing the O2% in the sweep gas decreases the requirement for sweep gas steam generation. This results in a lower boiling duty on the steam system, but also leads to a reduced mass flow in the oxygen process heat recovery train. The reduction in mass flow in the oxygen process heat recovery train decreases the amount of boiling duty that can be done within the train. The two effects are not equal which results in a decrease in nuclear steam usage. Also, the overall reduction in the mass flow of steam allows for the steam to be superheated to a higher temperature and results in a lower ΔT for the resistive preheat coil. Thus, in addition to a mild reduction in the amount of thermal input into the process, the MW consumed in the resistive preheat coil is decreased. Therefore, more electric power is available to the electrolyzer and the hydrogen production rate is increased.

Operating at nonisothermal conditions also improves the overall hydrogen production rate of the plant, as shown in Figure 5. Operation of the SOEC at a reduced potential reduces the power consumption per kg H2 produced. Lowering the operating potential results in a lower outlet temperature from the SOEC cell, which in turn reduces the overall heat recovery from the product similar to that outlined above for changes in the operating temperature during isothermal operation. Additional thermal energy compensates the loss of boiling as seen when comparing the percent of boiling duty of case 3 to case 10 and 11 and case 9 to case 12 in Table 1 or the thermal MW of the corresponding cases in Table 2. The drop in the exit temperature of the SOEC also reduces the superheat temperature of the steam prior to the resistive heating coil, thus requiring more energy for resistive feed preheat. The energy required for boiling and final feed preheat is slightly less than the reduction in energy for the electrolyzer. Consequently, an increase is seen in production with nonisothermal operation.

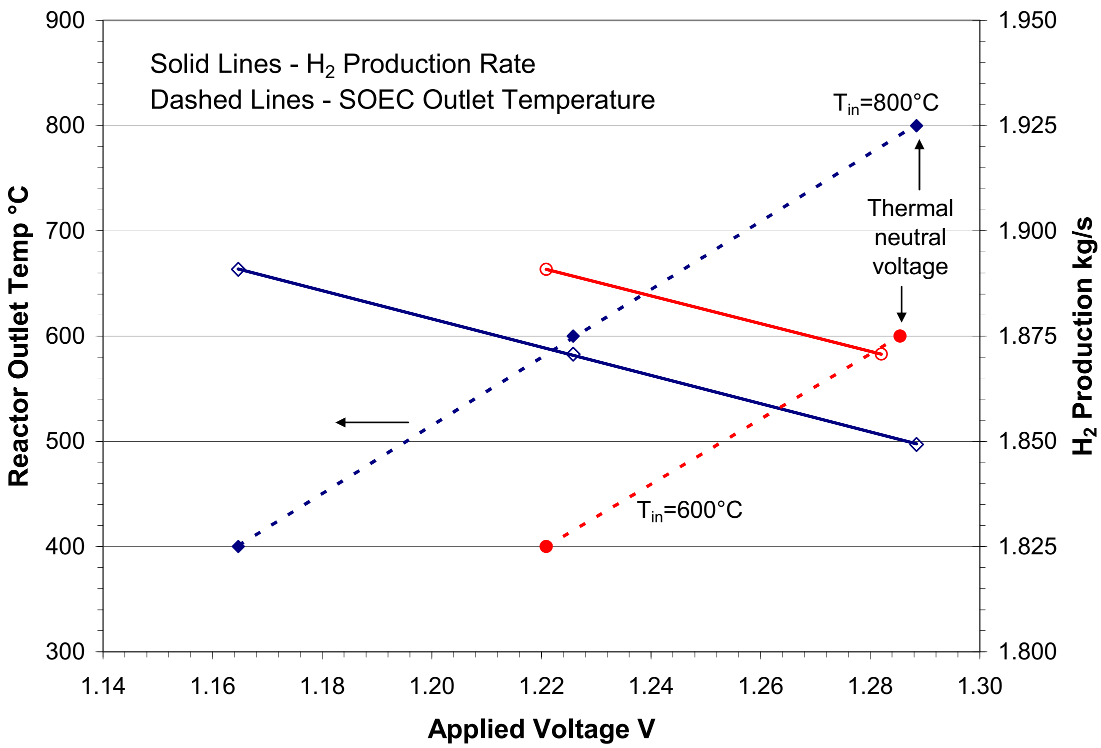


Figure D-5: H_2 Production and Reactor Outlet Temperature for Nonisothermal Operation of SOEC

Figure 5 has another interesting feature. The curves for the exit temperature of the SOEC at inlet temperatures of 800°C and 600°C are not parallel to each other yet at voltages which lead in each case to the same SOEC exit temperature, there H2 capacity is equivalent. Closer inspection of the distribution of power for cases 9 and 10 and also cases 11 and 12 indicate that the power consumption only differs between the resistive preheat coil and the SOEC. However, the total of the power consumption of these two components is equivalent. Therefore, setting the reactor exit temperature sets the process heat recovery as well as the thermal input required from the nuclear island. The H2 capacity is also set. Effectively, operating under nonisothermal conditions moves electrical energy which was originally dissipated as heat in the electrolyzer to the resistive preheat coil. This allows for more flexible operation of the SOEC over the life of the electrolysis reactor. Running at reduced voltages can extend the life of the cell as in general an increase in voltage is seen over the course of a cells life. As the cell ages, voltage can be transferred from the preheat coil and dissipated within the electrolyzer cell extending the effective lifetime of the electrolysis reactor. The results also indicate that design of heat transfer for the process can be disconnected from the type of SOEC used as long as an exit temperature is specified. Changes in the availability or quality of the nuclear thermal input still would have an effect on both the heat transfer and H2 capacity as seen by reduced electrical power for the electrolyzer and resistive preheat coil.

Overall, the energy supplied from the nuclear power plant can be divided into a percent of electric power utilized as thermal and electrical inputs to the hydrogen production island. The net efficiency of the nuclear plant for the supercritical steam cycle is ~47%. The plant is designed for 565 MW of thermal output, which coincides with a net electric power production of 265.6 MWe. The thermal input reduces production of electric power from the plant. The equivalent megawatts electric for this power production is given in Table 2 and ranges between 12.5 to 22 MWe. Subtracting the nuclear thermal heat from the 265.6 MWe of possible power generation yields the net nuclear island power production column in Table 2. On a percentage basis, the net thermal input is between 4.7 to 8.3% of the 265.6 MWe. The remaining power available, as shown in the net nuclear island power produced, is consumed by boiler feedwater pumps, product compressors, resistive preheat of the feed, and the electrolyzer or sent to the power grid for sale. The electrolyzer consumes the most of the net nuclear island power, 81 to 92%, while resistive preheat, 3 to 15%, and product compression, 4 to 5%, comprise the next two largest draws on power. Boiler feed water pump power and the residual power supplied to the grid together are typically less than 0.1% of the overall power consumption.

D.4.1 Results from Alternate Steam Conditions

In addition to simulating points utilizing reheat steam from the supercritical steam cycle systems such as that specified in the Power Conversion System Special Study section of this Preconceptual Design Studies Report, nominally 2MPa and 600°C, reduced steam conditions, 2MPa and 538°C, were also briefly modeled. Results of the simulation show an overall reduction in capacity of the hydrogen production island consistent with the drop in efficiency of power conversion, ~47% for supercritical vs. ~43% for subcritical. The increase in steam usage to achieve similar boiling duties was a smaller effect, < 0.5%, comparatively to the power conversion efficiency. Alternatively, the two He or two supercritical CO_2 cycles outline in the Power Conversion System Special Study would use less mass flow to accomplish the boiling duty as the temperature of the heat transfer medium is higher than that of the supercritical steam cycle, see Power Conversion System Special Study for more details. The power conversion efficiency of the He and supercritical CO_2 cycles would also have a bearing on the overall hydrogen plant capacity. Potentially, the He cycles with high quality heat, 900°C, could supplement or eliminate the need for resistive preheating. This would remove some of the coupling observed between the electrolyzer, resistive preheat coil, and the rest of the process in its current orientation. The higher quality of heat available in both CO_2 and He cycles would alternatively allow the system pressure to be raised and reduce the power consumed by product compression. The effectiveness of using either the CO_2 or He cycles would depend on cost and difficulties associated with the transport of the high temperature heat transfer medium used in each cycle.

D.4.2 Process Economics

Process Economics were not completed for any of the designs simulated as the intent of the study was to directionally ascertain what operating points would be beneficial from an overall process standpoint. A detailed economic estimate would also require knowledge of a specific electrolyzer and its properties. The electrolyzer operating conditions tested were not specific to a single electrolyzer and would have to be verified once an SOEC technology was chosen. Materials of construction and their quantities differ for SOEC technologies and further complicate any evaluation.

D.4.3 Practical Operation of Solid Oxide Electrolytic Cells

Researchers working on the state of the art SOEC are still faced with numerous challenges which need to be resolved. Currently, cells are run at or near atmospheric pressure. This condition would surely need to change to make an actual plant scale cell viable. The process evaluated in this document operates at only modest pressure, 0.9MPa and minimal cathode to anode differential pressure. However, higher pressure process design could be devised. Higher pressures may be possible as all the power conversion cycles presented here were modeled. In addition, size and scale of both cell and stack and characteristics of the cell, particularly higher current density, need to improve in order to reduce the number of modules required for the simulated hydrogen production rate. The mechanical seals of the cells have been problematic with both hydrogen and oxygen leakage noted. Also, corrosion of the anode can occur under the high temperature oxidative environment seen during normal operation. The system as a whole has a relatively small operating window as the solid electrolyte internal resistance is a strong function of temperature. This limitation may limit the temperature drop across the reactor leading to higher cell voltages being applied to maintain higher exit temperatures. Alternatively, the electrolysis reactor could be made up of banks of SOEC of differing materials which would be best suited for the local condition along the reactor temperature, composition, and pressure profile. In spite of the many challenges, over 2000hrs of operation at nearly 900 liters/hr has been reported in the literature with reasonable efficiencies 55% on a HHV basis [9].

D.5 Summary and Conclusions

A brief process modeling study of the electrochemical reaction of water to hydrogen and oxide in a high temperature steam electrolysis solid oxide electrolytic cell was conducted. Thermal and electrical inputs required to complete the reaction were taken from a nuclear power island designed to operate at 565 MWth. Steam at 2MPa and 600°C was used as the thermal input from the nuclear island to the hydrogen production island. A process incorporating the heat recovery from the product stream along with additional thermal input from the nuclear island for additional boiling duty and resistive heating for final preheat of the steam to cell inlet temperature was simulated at operating temperatures of 600, 700, and 800°C while varying the composition of the sweep gas at the anode, electrode where O_2 is generated, from 25 to 50 to 75% O_2 with the balance of the stream as steam. Hydrogen production rates ranged from 1.62 to 1.86 kg/s over the specified test conditions. The results show a trend toward optimal operating the process at a lower SOEC outlet temperature, 600°C, and higher concentrations of oxygen in the sweep gas, 75%. The percentage of the boiling duty utilizing nuclear thermal input increased and feed and sweep gas steam temperatures exiting the process heat recovery train decreased under these conditions. This was more than offset by the decrease in the energy required to resistively heat the feed and sweep gas to the inlet temperature of the SOEC. Further results indicate that the level of hydrogen production is set by the exit temperature of the SOEC as the subsequent heat recovery train is set.

An increase in hydrogen capacity was seen when the SOEC was run at lower operating potentials, nonisothermal operation. A maximum production rate of 1.89 kg/s was achieved for two nonisothermal cases in which the exit temperature of the electrolysis cell was 400°C. However, no change in hydrogen capacity was observed when comparing isothermal and nonisothermal cases in which the exit temperatures from the SOEC were equivalent. The transferring of electrical energy from the SOEC to the resistive preheat coil was observed after comparison of the cases. Steam at subcritical, 2 MPa and 538°C, was also simulated resulting in lower hydrogen plant capacities. The reduction in overall power conversion efficiency was the major factor in the decline with a small decrease in capacity noted by the increase in mass flow required to achieve boiling duties similar to those for the higher temperature steam.

The impact of hydrogen plant capacity on a pipeline application is reduced compared with a sole source mode of operation. Fluctuations on demand from the pipelines can cause plants supplying the pipeline to run below rated capacity. Thus, a range in the hydrogen plant capacity here does not immediately dismiss the less favorable operating cases from becoming realistic plant designs. The economic details of a particular design basis for an established pipeline would have to be evaluated before a decision on the final plant design.

D.6 Future Considerations and Work

The process was evaluated for the key areas where significant efforts are needed to improve technology. The key focus of future work is on the SOEC itself. Materials and design of the heat exchange network are manageable for both the supercritical and subcritical steam PCS cycles. The clean-up, purification, and compression of the hydrogen product stream is also within the current means of process design available.

The results of the process modeling study have implications on the direction of future advances in SOEC technology. The observed trend in operating conditions would further tax current problems observed with SOEC. High oxygen concentrations would be seen at the anode of the cell causing a potential increase in the oxidation of the electrode. Oxidation of the electrode material renders it nonconducting and results in an increase in the potential needed to maintain electron flux or a reduction in hydrogen production rate is necessary. A cell operating temperature of 600°C or lower may change the selection of materials currently used in the manufacture of some SOEC. The process design operates at relatively modest pressure, 0.7MPa, without a significant pressure differential through the cell cathode to anode, which is beneficial to the design of the commercial electrolysis reactor.

The focus for SOEC should be scale up of the cells and in electrolysis reactor design. These are the largest hurdles to the commercialization of technology. The second largest hurdle is reliability. Leak free, long-term operation of the SOEC will be key to reducing the maintenance cost of the plant. Additional advances in materials selection may help in addressing these problems and could extend the useful life of a cell stack further reducing the operating cost of the plant. The actual operating temperatures are less import from a process design viewpoint and will inevitably fall to the least expensive process economics.

D.7 References

1. "Progress Report for the DOE Hydrogen Program". 2005 DOE Hydrogen Program Annual Progress Report, October 2005
2. Smith, W. Novis, Santangelo , Joseph G. Hydrogen: Production and Marketing. ACS Symposium Series. 1980, American Chemical Society, Washington DC
3. Pham, A-Quac. Lawrence Livermore Lab, CA. Proc. of 2001 DOE H2 Program Review Publ. NREL. CAN: 138:79978 AN 2002: 682400
4. High Efficiency Generation of Hydrogen Fuels Using Thermochemical Cycles and Nuclear Power by L. C. Brown, G. E. Besenbruch, K. R. Schultz, S. K. Showalter, A. C. Marshall, P. S. Pickard and J. F. Funk; Sandia Nat. Labs, Univ. of Kentucky; Proceedings of the 2002 Spring Meeting of the American Institute of Chemical Engineers, March 10-14, 2002, in New Orleans, LA.
5. O'Brien, J.E., Stoots, C.M., and Hawkes, G.L. "Comparison of a One-Dimensional Model of a High-Temperature Solid-Oxide Electrolysis Stack with CFD and Experimental Results" IMECE2005-81921, Proceedings of IMECE2005 2005 ASME International Mechanical Engineering Congress and Exposition November 5-11, 2005, Orlando, Florida USA

6. Casper, M.S. Hydrogen Manufacture by Electrolysis, Thermal Decomposition and Unusual Techniques. 1978, Noyes Data Corporation, New Jersey, USA.
7. Yildiz, B. Hohnholt, K., Kazimi, M.S. "Hydrogen Production Using High Temperature Steam Electrolysis and Gas Reactors with Supercritical CO2 Cycles. MIT CANES Report, MIT-NES-TR-002, December 2004
8. Memmott, M.J., Driscoll, M.J., Kazimi, M.S., Hejzlar, P. "Hydrogen Production by Steam Electrolysis Using a Supercritical CO2-Cooled Fast Reactor. MIT CANES Report, MIT-NES-TR-007, February 2007.
9. Herring, S. et al., 2006 DOE Hydrogen Program Peer Review, Washington, DC, May17, 2006.

E.0 90% REVIEW MEETING – COMMENTS/ACTIONS

The 90% (i.e., project complete) review meeting between BEA/INL principals and AREVA NGNP Team principals was held on June 12, 13 and 14, 2007 in Idaho Falls, Idaho.

Table E-1, beginning on the next page, lists the official comments and actions assigned to AREVA for action.

DOCUMENT MANAGEMENT CONTROL SYSTEM (DMCS)
REVIEW COMMENTS AND RESOLUTIONS

TABLE E-1 90% Review Meeting - Comments and Actions

Technical Point of Contact:	Phone No.:	Return Comments To:	MS:	E-Mail:	Comments Due By:	Reviewer's Name/Discipline:	Phone No.:
Sam Bader	526-8929	Sam Bader					

Comments resolved by:	Date:	Signature of reviewer accepting comment resolutions:	Date:

All comments, submitted within the scope of the review, must be resolved between submitter and document owner. If an acceptable resolution cannot be negotiated, the comment will be escalated to management for resolution.

Document ID:		Document Title: AREVA PCDSR	Revision ID: DAR No.:
Item No.	Page No./ Section/Zone	Review Comment	Comment Resolution
1		For the econ analysis, I would recommend that a more across the board sensitivity analysis be performed similar to the one point analysis of the change in H2 escalation which was escalated approximately 35%. If one does this for all of the significant (and perceived to be insignificant variables) the cost and revenue drivers should become apparent. This not be a huge study but is typical of pre-conceptual studies in R&D.	AREVA agreed to provide the spreadsheets and data. BEA will perform the sensitivity analyses.
2	13, 28, D-1	While following the discussion regarding proposed Future Study 21.1.1 on expanding the role of the steam cycle, I noticed a repeating typo regarding Appendix D. See pages 13, 28, and D-1: "temperture."	AREVA action
3	Section 9.2.1 and 9.2.2	Are there two cranes on the rails over the reactor area in the Reactor Building and Reactor Services building? In section 9.2.1, a 300 ton crane is mentioned. In section 9.2.2, a 136 ton crane is mentioned.	AREVA action –correct the reference to the 136 ton crane in Section 9.
4	General	The recommended licensing strategy needs to have an ultimate objective of supporting the application for and receipt of a design certification for the commercial application of the HTGR technology.	AREVA Action. The PCDR shall be modified accordingly as required.
5	Presentation	AREVA indicates that the initial license will likely be as a test reactor under 104c. This has a maximum power level restriction less than that which we would want to achieve during initial operation of the plant. There is a provision in the code for licensing non-LWR plants as a prototype which does not have this power restriction. The licensing strategy should refer to the prototype license instead of 104c.	AREVA Action. The PCDR shall be modified accordingly

DOCUMENT MANAGEMENT CONTROL SYSTEM (DMCS)
REVIEW COMMENTS AND RESOLUTIONS

Document ID:		Document Title: AREVA PCDSR	Revision ID:	DAR No.:
Item No.	Page No./ Section/Zone	Review Comment	Comment Resolution	
6		A scoping calc indicates Rx vessel stresses in the 10 inch section of 14,700 psi and 24,500 psdi in the 6 inch section. These figures seem high relative to the creep rupture stress vs. time at temperature data presented. What are the assumed allowables for the vessel and what is their basis?	AREVA action to clarify in PCDSR – add the allowables and margins for the belt line area away from discontinuities.	
7	various	Regarding the projected 20 year service life for the PCS IHX and 5 year service life for the H2 IHX: The plant layout does not appear to show much access space for performing changeout, inspection, or maintenance activities in the cavities containing the IHXs. (1) Are there going to be manways or handholes in the IHX shells to perform some level of inspection or maintenance on the internal components? (2) Will the cavities contain permanently installed access ladders/stairs to such features, or will staging have to be built during maintenance evolutions?	This level of detail is not required at the pre-conceptual phase of design. Chapter 14 of the PCDSR pertains to maintenance requirements. A requirement is that provisions must be provided to inspect, maintain and, if necessary, replace equipment that is not intended for no access over the life of the plant. This requirement will be applied for the IHX and other SSCs during the next phases of design development. Significant maintenance activities on the IHXs will be performed from the top.	
8		Comment: For this design, what are the anticipated temperatures that control rods will experience during normal operation and during depressurized conduction cooldown?	AREVA action – bounding numbers will be provided in the PCDSR for this area under these conditions.	
9		Comment: How well is the vessel emissivity understood? Will there be a difference between the inside of the vessel (i.e., absorption) and the external?	There will be a difference in the inside and outside vessel wall emissivities. Characterizing these emissivities has been identified as a data need and will be addressed during the completion of design development.	

10		Comment: It appears from the creep rupture data that the properties of 617 are not degraded by the HTR helium environment; has there been a statistical analysis to analytically address this potential issue? Are there similar studies available to determine if HTR He degrades the creep rupture properties of Haynes 230?	Resolving these issues is a part of the R&D program requirements for examining the effects of the environment on material strength properties as identified in the PCDSR, Appendix C.
11		Question: Coatings on a compact heat exchanger? What composition, made by what method and how does one address the code issues of inspectability and repair?	Coatings would be aluminum based; the precise composition and methods of application are proprietary to AREVA. It will not be possible to inspect the coatings during operation. Extensive corrosion / erosion test programs combined with conservative design based on the results of this testing will be used to provide adequate confidence in the integrity of the coatings over the expected life of the heat exchanger module. It is also not possible to repair a module; instead a failed or end-of-life module will be replaced.
12		Required: In the PCS presentation, a plant efficiency number of 48.2% was given. Did this plant efficiency apply specifically to the PCS? Efficiency numbers for electricity and hydrogen generation from the overall plant should be provided instead because of the other losses that occur (thermal losses, hotel load losses, compressor energy costs, etc.). Otherwise, the numbers are biased upward in comparison to the standard practice of giving overall plant efficiencies for electricity and hydrogen production.	Section 3.5.3 states that the net plant efficiency is 45.8%. Section 8.6.3.2 states that the gross PCS efficiency for a FOAK plant is 48.2%.

13		Need to have more of a discussion in the PCDSR concerning the He test loop. Discuss the drivers (justification), requirements, design criteria and anticipated component testing (at a preconceptual Design Level).	Table 19-6 lists the required testing facilities. The specific requirements for the capabilities of these tests sites will be established as part of the design development. Critical needs will be addressed early in the Conceptual Design phase to ensure that facilities can be designed and constructed in the time frame required to support critical SSC qualification.
14		For Consideration: Is the schedule dependent or independent of the hydrogen production technologies that will be delivered by the DOE Nuclear Hydrogen Initiative?	The PCDSR schedule makes no assumption on the availability of the hydrogen production process. The 60MWt loop will be provided and will be isolated at initial operation if the hydrogen plant is not available or does not require process heat. The schedule does assume that the hydrogen process will require only He and/or electricity from the NHSS.
15		Risk is a qualitative measure at this level of design. Will this become quantitative or even a measure of $ risk as the process continues and becomes more refined and understood? Obviously, this team is going to use risk management tools that they are familiar with or have had success with in the past. Do these tools follow the design process and knowledge gained?	BEA Action.

16		The greatest programmatic risk associated with using 9Cr-1MoV forgings for the pressure vessel is obtaining large enough forgings. JSW, as of 4/06, had never made ingots of 9Cr larger than 140 tons. They appeared to be satisfied with the production process, materials properties, and technology basis for fabricating 9Cr-1MoV steel forgings up to this size, but were very concerned about the scale-up required for the larger forgings needed for an NGNP vessel. They indicated that all needed laboratory-scale work on the forging of 9Cr-1MoV steel has been completed but that production of very large ingots would be needed to develop and qualify the process for sizes needed for NGNP. This will be a time-consuming and expensive step. The estimates that AREVA has made have indicated that ingots up to 200 tons may be needed for the largest single vessel sections (the flange rings). There is a clear risk to schedule, cost, and product quality in assuming that JSW can make these rings in the time frame required for the NGNP and must be addressed for an RPV made of 9Cr to be included as part of the project.	Item P-002 Table 18-5 identifies this specific risk with the use of 9Cr-1Mo material for the RPV as in the category requiring mitigation. A future study to be conducted in FY08 has been formulated to address all risk issues associated with selection of the RPV material for NGNP. This specific risk will be addressed as part of this study when evaluating 9Cr-1Mo as a candidate material.
17		The basis for the choice of 500C for Tc is complex and needs to be better explained. Is it affected by the choice of prismatic as opposed to pebble bed fuel?	The factors affecting the selection of reactor inlet temperature are examined in detail in the Primary / Secondary Concept special study, Appendix B4. There are significant differences between the prismatic and pebble bed designs in the factors affecting the selection of reactor inlet temperature.