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Preconceptual Engineering Services For The Next Generation Nuclear Plant (NGNP) With Hydrogen Production

Executive Summary Report - NGNP and Hydrogen Production Preconceptual Design Studies Report

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Acronyms

AC	Alternating Current
AGR	Advanced Gas Reactor
ALWR	Advanced Light Water Reactor
ASME	American Society of Mechanical Engineers
ATR	Advanced Test Reactor
AVR	Arbeitsgemeinschaft Versuchsreaktor (in Germany)
BAF	Bacon Anisotropy Factor
BCC	Base Construction Cost
BEA	Battelle Energy Alliance
BOP	Balance of Plant
C-C	Carbon-Fiber Reinforced Carbon
CB	Catcher Bearing
CDC	Capitalized Direct Costs
CEA	French Commissariat à l’Energie Atomique
CFC	Capitalized Financial Cost
CFR	Code of Federal Regulations
CIC	Capitalized Indirect Services Cost
COC	Capitalized Owner Cost
COLA	Combined Operating License Application
CPC	Capitalized Pre-construction Costs
CSC	Capitalized Supplementary Costs
DBE	Design Basis Event
DC	Direct Current
DCC	Direct Construction Costs
DDN	Design Data Need
DOE	US Department of Energy
DV&S	Design Verification and Support
EFPD	Effective Full Power Days
EM	Electromagnetic
EMB	Electromagnetic Bearing
EPRI	Electric Power Research Institute
FES	Fuji Electric Systems
FFF	Fuel Fabrication Facility
FIC	Field Indirect Costs
FIMA	Fissions per Initial Metal Atom
FMC	Field Management Cost
FSV	Fort St. Vrain Nuclear Power Station

GA	General Atomics
GCRA	Gas-Cooled Reactor Associates
GIF	Generation IV International Forum
GNEP	Global Nuclear Energy Partnership
GT	Gas Turbine
GT-MHR	Gas Turbine – Modular Helium Reactor
H2-MHR	Hydrogen Production – Modular Helium Reactor
HEU	High-Enriched Uranium
HFR	High Flux Reactor (Petten, Netherlands)
HI	Hydriodic Acid
HM	Heavy Metal
HPC	High Pressure Compressor
HPCC	High Pressure Conduction Cooldown
HPS	Helium Purification System
HSS	Helium Services System
HTGR	High Temperature Gas-Cooled Reactor
HTIV	High Temperature Isolation Valve
HTS	Heat Transport System
HTTR	High Temperature Engineering Test Reactor (Japan)
HVAC	Heating, Ventilation, and Air Conditioning
IAEA	International Atomic Energy Agency
IHX	Intermediate Heat Exchanger
ILS	Integrated Laboratory-Scale (experiment to demonstrate SI process)
INL	Idaho National Laboratory
I-NERI	International Nuclear Energy Research Initiative
IPS	Investment Protection System
IPyC	Inner Pyrocarbon
ISI	In-Service Inspection
ITRG	Independent Technology Review Group
IVM	In-Vessel Metalworks
JAEA	Japan Atomic Energy Agency
JSW	Japan Steel Works
KAERI	Korean Atomic Energy Research Institute
KIER	Korea Institute of Energy Research
KIST	Korea Institute of Science and Technology
LEU	Low Enriched Uranium
LMTD	Log Mean Temperature Difference
LPC	Low Pressure Compressor
LPCC	Low Pressure Conduction Cooldown
LWA	Limited Work Authorization

LWR	Light Water Reactor
MHR	Modular Helium Reactor
MMBtu	Million (thousand-thousand) Btu
MS	Molten Salt
MWe	Megawatt - electric
MWt	Megawatt - thermal
NERI	Nuclear Energy Research Initiative
NFI	Nuclear Fuel Industries (Japan)
NGNP	Next Generation Nuclear Plant
NHDD	Nuclear Hydrogen Development and Demonstration
NHI	Nuclear Hydrogen Initiative
NI	Nuclear Island
NNSA	National Nuclear Security Administration
NOAK	Nth of a Kind
NRC	US Nuclear Regulatory Commission
O&M	Operations & Maintenance
OC	Operations Center
OCC	Overnight Construction Costs
OKBM	OKB Mechanical Design (Russian Federation)
OPyC	Outer Pyrocarbon
ORNL	Oak Ridge National Laboratory
PBMR	Pebble Bed Modular Reactor
PBR	Pebble Bed Modular Reactor
PCDIS	Plant Control, Data, and Instrumentation System
PCDSR	Preconceptual Design Studies Report
PCHE	Printed-Circuit Heat Exchanger
PCS	Power Conversion System
PH	Process Heat
PHA	Preliminary Hazards Assessment
PHC	Primary Helium Circulator
PHX	Process Heat Exchanger
PIE	Post-irradiation Examination
PMR	Prismatic Modular Reactor
PPMP	Preliminary Project Management Plan
PRA	Probabilistic Risk Assessment
PSAR	Preliminary Safety Analysis Report
PSR	Permanent Side Reflector
PyC	Pyrocarbon
QA	Quality Assurance
R/B	Release Rate to Birth Rate

RB	Reactor Building
RCCS	Reactor Cavity Cooling System
RF	Russian Federation
RN	Radionuclide
ROK	Republic of Korea
RPS	Reactor Protection System
RV	Reactor Pressure Vessel
RSB	Reactor Service Building
RSC	Reserve Shutdown Control
SAR	Safety Analysis Report
SCS	Shutdown Cooling System
SDD	System Design Description
SFC	Static Frequency Converter
SFSS	Spent Fuel Storage System
SHC	Secondary Helium Circulator
SI	Sulfur-Iodine
SiC	Silicon Carbide
SMR	Steam Methane Reforming
SNF	Spent Nuclear Fuel
SNL	Sandia National Laboratories
SOE	Solid Oxide Electrolyzer
SOEC	Solid Oxide Electrolyzer Cell
SOFC	Solid Oxide Fuel Cells
SRM	System Requirements Manual
SSC	Systems, Structures, and Components
TBD	To Be Determined
TC	Turbocompressor
TDP	Technology Development Plan
THTR	Thorium High Temperature Reactor
TM	Turbomachine
TRISO	TRI-material, ISOtropic
TRU	Transuranic
UA	Product of Overall Heat Transfer Coefficient and Heat Transfer Area
UCO	Uranium Oxycarbide (a mixture of uranium oxide and uranium carbide)
VHTR	Very High Temperature Reactor (an MHR capable of reactor outlet temperatures >850°C)
VLPC	Vented Low Pressure Containment
w-Pu	Weapons Grade Plutonium
WBS	Work Breakdown Structure

1. INTRODUCTION

1.1 Scope

The Energy Policy Act of 2005 required the Secretary of the U.S. Department of Energy (DOE) to establish the Next Generation Nuclear Plant (NGNP) Project. In accordance with the Energy Policy Act, the NGNP Project consists of the research, development, design, construction, and operation of a prototype plant (to be referred to herein as the NGNP) that (1) includes a nuclear reactor based on the research and development activities supported by the Generation IV Nuclear Energy Systems initiative, and (2) shall be used to generate electricity, to produce hydrogen, or to both generate electricity and produce hydrogen. The NGNP Project supports both the national need to develop safe, clean, economical nuclear energy and the National Hydrogen Fuel Initiative (NHI), which has the goal of establishing greenhouse-gas-free technologies for the production of hydrogen. The DOE has selected the helium-cooled Very High Temperature Reactor (VHTR) as the reactor concept to be used for the NGNP because it is the only near-term Generation IV concept that has the capability to provide process heat at high-enough temperatures for highly efficient production of hydrogen. The DOE has also selected the Idaho National Laboratory (INL), the DOE's lead national laboratory for nuclear energy research, to lead the development of the NGNP under the direction of the DOE.

As part of the initial design phase of the NGNP Project, the Battelle Energy Alliance (BEA), operator of the INL, contracted with three modular helium reactor technology development teams, including a team led by General Atomics (GA), to provide preconceptual engineering services. The GA team consists of Washington Group International (Washington Group), Rolls-Royce in the United Kingdom, Toshiba Corporation and Fuji Electric Systems (FES) in Japan, the Korean Atomic Energy Research Institute (KAERI) and DOOSAN Heavy Industries and Construction (DOOSAN) in the Republic of Korea, and OKB Mechanical Design (OKBM) in the Russian Federation. A Work Plan was prepared by GA and approved by BEA to define the work scope to be performed by the GA team. The tasks defined in the Work Plan include:

- Prepare a system requirements manual (SRM) to identify the NGNP top-level requirements (i.e., mission needs and objectives) and to show how these top-level requirements flow down through subordinate requirements at the plant, system, and component level

- Perform four special studies:
  - A study to compare the two modular helium reactor variations being considered for the NGNP; namely, the pebble bed reactor (PBR) and the prismatic modular reactor (PMR)
  - A study to develop a recommendation with respect to the NGNP reactor power level and the size of the NGNP hydrogen production plant(s)

- A study to compare potential working fluids for the NGNP secondary heat transport system (HTS) and to recommend a working fluid
- A study to identify the end-products and by-products of the NGNP and to estimate the economic value or economic penalties associated with these products

• Prepare a Technology Development Plan (TDP) to define the design data needs (DDNs) for the NGNP and to help focus the NGNP and NHI R&D Programs to be responsive to these DDNs

• Develop a preconceptual plant design to the extent necessary to support preparation of an NGNP project cost estimate and schedule

• Perform a safety assessment for the proposed NGNP design

• Evaluate NGNP licensing options and recommend a licensing approach

• Develop capital cost and 30-year operating cost estimates for the NGNP

• Prepare an NGNP Project schedule

• Prepare a life cycle cost analysis and economic assessment for a follow-on commercial plant based on the NGNP

The Preconceptual Design Studies Report (PCDSR) [PCDSR 2007] covers all of the work performed by the GA Team as identified above. This Executive Summary Report has been prepared to satisfy a requirement of the INL Statement of Work. The approach used in preparing this Executive Summary Report has been to present essentially all of the information in the PCDSR in summary fashion and to maintain a one-to-one correspondence with the first- and second-level section numbers in the PCDSR such that a reader of this summary report can easily go to the PCDSR and find more detailed information on topics of interest.

Sections 2 and 3 provide preconceptual design information on the overall NGNP and on the various plant systems, respectively. Section 4 describes the NGNP buildings and structures. Section 5 presents plant assessments, including a safety assessment, a recommended NGNP licensing approach, and NGNP capital cost and operating cost estimates developed based on the preconceptual design information presented in Sections 2 through 4. Section 5 also presents a life cycle cost estimate and economic assessment for two variations of commercial hydrogen production plants that are based on the NGNP. Section 6 presents a resource-loaded NGNP project schedule based on the NGNP Project Work Breakdown Structure (WBS) provided by INL and the NGNP capital cost estimate. Section 7 addresses the R&D required to support design and construction of the NGNP, and presents GA's fuel acquisition strategy for the NGNP. Section 8 lists the references cited throughout document.

1.2 NGNP Mission

As defined in the NGNP Preliminary Project Management Plan (PPMP) [PPMP 2006], the NGNP Project objectives that support the NGNP mission and DOE's vision are as follows:

a. Develop and implement the technologies important to achieving the functional performance and design requirements determined through close collaboration with commercial industry end-users
b. 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.
c. Establish the basis for licensing the commercial version of the NGNP by the Nuclear Regulatory Commission (NRC). This will be achieved in major part through licensing of the prototype by NRC, and by initiating the process for certification of the nuclear system design.
d. Foster rebuilding of the U.S. nuclear industrial infrastructure and contributing to making the U.S. industry self-sufficient for its nuclear energy production needs

GA is in agreement with the above objectives, but believes that the NGNP Project as described in the PPMP is missing a program element that is essential to achieving objectives b and d. Specifically, GA believes that the NGNP Project should include a demonstration of the viability of commercial-scale coated-particle fuel fabrication by building, licensing, and operating an NGNP Fuel Fabrication Facility (FFF) in Idaho to supply UCO fuel for the NGNP and to demonstrate the fuel fabrication technology needed for a commercial fuel supply business, thereby greatly reducing the costs and risk associated with a first-of-a-kind commercial fuel fabrication facility.

GA also believes that the NGNP mission should be expanded to include demonstration of the "Deep Burn" concept that has been proposed by GA for destruction of weapons-grade Pu (w-Pu) and transuranics (TRU) in the spent nuclear fuel from LWRs. GA's economic analysis indicates that the extremely large burnup that can be achieved in MHRs with fully enriched Pu and TRU fuels opens entirely new possibilities with respect to the economics of fuel recycle and material disposition.

As shown in Figure 1-1, the flexible fuel cycle capability of the MHR, combined with its flexible energy output capability results in a design concept that is very well suited for a wide variety of energy-growth scenarios. It is a potential major advantage for gas-cooled reactors that MHR technology, which will be demonstrated by the NGNP as a major new reactor system for commercial application, is well suited to contribute to the disposition/recycling mission and is capable of destroying the heat generating actinides that limit repository capacity. MHRs are uniquely attractive for this reason, because they will require no subsidies to compete with LWRs (particularly in the large energy market segments where LWRs currently cannot penetrate at all) beyond support for first-of-a-kind engineering and commercialization costs.

Consequently, the NGNP Project could play an important role with respect to both Russian w-Pu destruction and DOE's Global Nuclear Energy Partnership (GNEP), and much could be gained by better coordinating the three activities. This is because the MHR core and fuel designs for these missions are expected to be substantially identical and easily testable in an NGNP PMR. At the very least, success of the NGNP would likely lead to commercial deployment of MHRs capable of operating with Pu and TRU-based fuel cycles. In other words, extensive commercial deployment of MHRs will benefit, not hinder, GNEP objectives, and the NGNP could make a major contribution to GNEP by demonstrating an MHR with the same engineering features as the reactor system planned for use in GNEP. Consequently, technology development (including fuel cycle issues) should be consistent with the objectives of all of the programs. Further, the NGNP could be used as a test bed for demonstrating the irradiation performance of advanced Deep Burn fuels that can significantly improve the economics and public acceptance of the closed fuel cycles being advocated as part of GNEP.

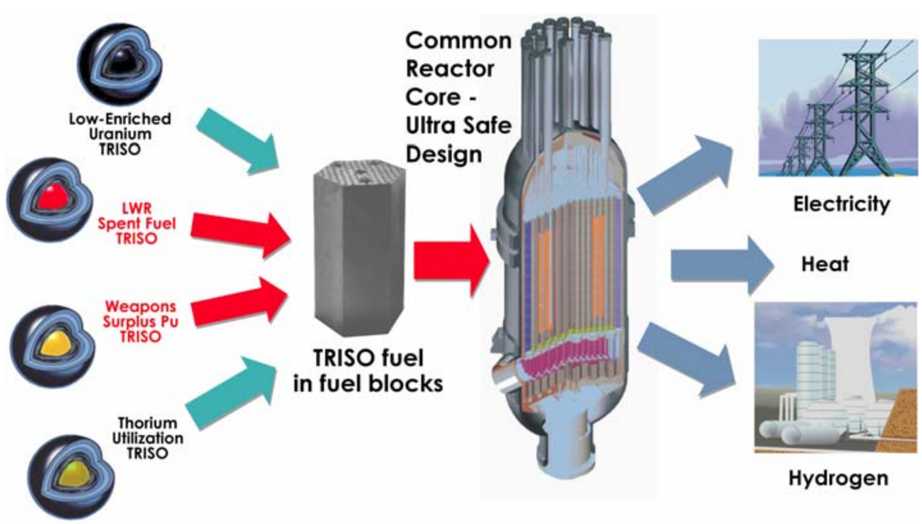

Figure 1-1. MHR Fuel Cycle and Energy Output Options

1.3 Modular Helium Reactor Technology Status

High-temperature gas-cooled reactor (HTGR) technology has been under development since the middle 1960s for electricity production and a variety of process-heat applications, including the production of hydrogen. Two HTGR concepts, the PMR and the PBR, were developed and demonstrated in commercial-size plants in the 1970s and 1980s. The PMR concept was demonstrated in the Fort St. Vrain (FSV) nuclear power station in the U.S. The PBR concept was demonstrated in the AVR and THTR in Germany.

Since the mid 1980s, General Atomics (GA) has been developing a passively safe, modular-sized design referred to as the Modular Helium Reactor (MHR). In 1986, a task force consisting of General Atomics (GA), Bechtel, Combustion Engineering, EG&G Idaho, Gas-Cooled Reactor Associates (GCRA), General Electric, Oak Ridge National Laboratory (ORNL), and Stone and Webster Engineering performed an evaluation of the PBR and PMR concepts to determine which of these concepts could best meet the requirements of potential commercial users in the United States. At that time, commercial interest in the MHR was focused on highly efficient production of electricity and cogeneration of electricity and process steam. The strategy was to develop a standard passively safe MHR design that was amenable to serial production and to design certification by the U.S. Nuclear Regulatory Commission (NRC). The ranking of the two concepts was close, but plant economics (i.e., the overall electricity generation busbar cost) favored the PMR, which resulted in selection of a 4 x 350 MWt PMR as the reference plant design to be developed by the U.S. MHR Program.

Both the PMR and PBR concepts have gone through considerable design evolution since the 1986 study. The motivation for this evolution has been to reach higher power levels within the constraint of passive safety, and to achieve greater thermal energy conversion efficiency in order to improve the economics of the reactors relative to other options for electricity production. For the PMR, the reactor core diameter was first enlarged to increase the power level from 350 to 450 MWt. The power level was the increased to 550 MWt by further enlarging the core diameter. Finally, the design power was increased to 600 MWt by increasing the core power density of the 550-MWt design. The core outer diameter that GA selected for the 450-MWt, 550-MWt, and 600-MWt PMR designs was based on the results of a GA vendor survey that was performed to determine the largest diameter reactor vessel (RV) that could be fabricated using available commercial vessel manufacturing capability. The initial MHR plants had a Rankine (steam) power conversion cycle for the generation of electricity. Starting with the 450 MWt design, the steam generator was replaced with a gas turbine to obtain the higher efficiency available from a Brayton power conversion cycle. This concept (Figure 1-2) is referred to as the Gas Turbine MHR (GT-MHR) and is described in [Shenoy 1996]. This concept operates with a thermal power level of 600 MW and an outlet helium temperature of 850°C to drive the direct

Brayton cycle power conversion system with a thermal-to-electrical conversion efficiency of about 48 percent.

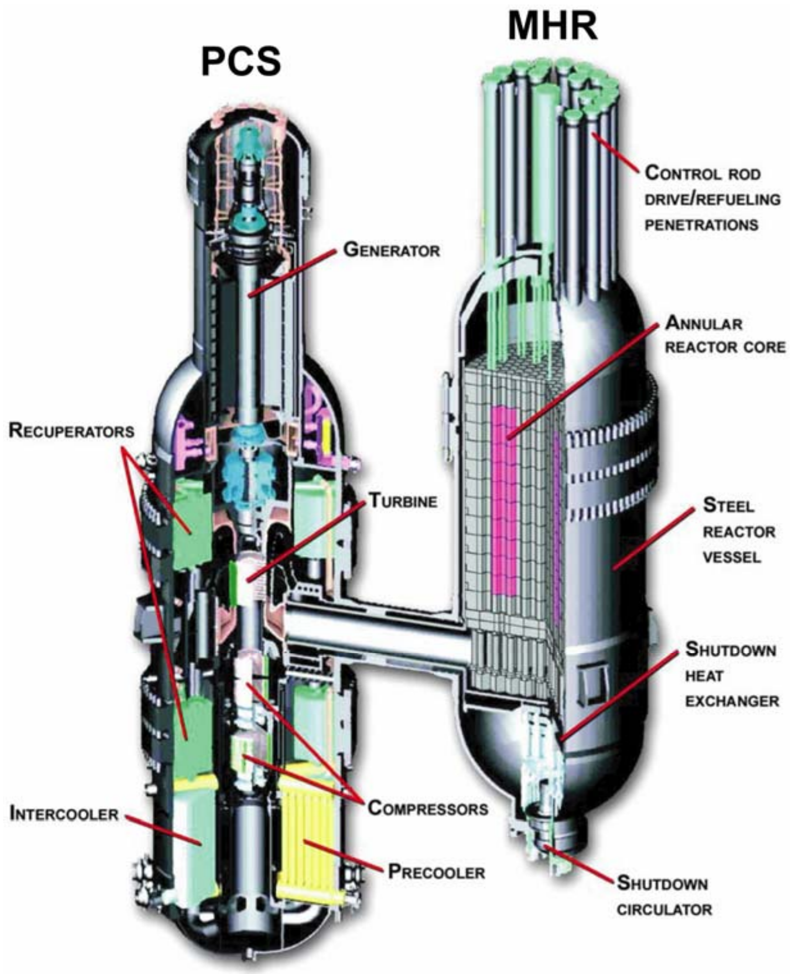

Figure 1-2. The Gas-Turbine Modular Helium Reactor

The DOE-sponsored HTGR Technology Development Program under which the GT-MHR conceptual design was developed was terminated in 1995, but development of the GT-MHR continued under the International GT-MHR Project, which was started in 1995 by GA and Minatom (currently Rosatom) of Russia for the disposition of surplus w-Pu. The project is currently being funded on a parity basis by the U.S. and Russian governments. The design work and technology development is being performed in Russia, but U.S. organizations, including GA and ORNL, have assisted the project through oversight and by sharing technology and experience related to design and operation of gas-cooled reactors.

The conceptual design for the w-Pu disposition GT-MHR was completed in 1997 and was independently reviewed by a panel of experts representing the U.S., Russia, Japan, Germany and France. The panel concluded that the GT-MHR is a viable design for deeply burning w-Pu in a once-through fuel cycle and that there were no insurmountable obstacles to prevent construction of a GT-MHR on a reasonable schedule. The preliminary design phase was completed in 2002. Work is currently focused on areas related to technical risks, including coated particle fuel development, demonstration of the Power Conversion System (PCS) with electromagnetic bearings, and verification/validation of computer codes for core design, including core physics, thermal hydraulics, fuel performance, and fission product transport. [LaBar 2003] provides additional information on the w-Pu disposition GT-MHR design and its technology background.

1.4 Hydrogen Production Technology Status

In principle, nuclear electricity can be used to split water using conventional low-temperature electrolyzers. However, even the high-efficiency electricity production available with a GT-MHR, economic evaluations of coupling nuclear energy to low-temperature electrolysis have generally not been favorable when compared to hydrogen production by steam-methane reforming (SMR). For this reason, two concepts that make direct use of the high-temperature process heat available from a MHR are being investigated to improve the efficiency and economics of hydrogen production. The first concept involves coupling the MHR to the Sulfur-Iodine (SI) thermochemical water splitting process. The second concept involves coupling the MHR to high-temperature electrolysis (HTE). Both processes have the potential to produce hydrogen with high efficiency and have been proven to work at the laboratory scale. A brief summary of the current status of these advanced hydrogen-production technologies is presented below.

1.4.1 Thermochemical Water Splitting Technology Status

Water thermally dissociates at significant rates into hydrogen and oxygen at temperatures approaching 4000°C. As part of an earlier study sponsored by the DOE Nuclear Energy Research Initiative (NERI), a team headed by GA and supported by Sandia National Laboratories (SNL) and the University of Kentucky evaluated 115 different thermochemical cycles that produce hydrogen [Brown 2003] at much lower temperatures. The sulfur-iodine (SI) cycle was determined to be the best cycle for coupling to the MHR because of its high efficiency and potential for further improvement.

The US DOE research and development effort on the SI process has been done primarily in collaboration with the French Commissariat à l'Energie Atomique (CEA) under an International NERI (I-NERI) agreement since 2003. As discussed in Section 3.8.2, the SI process consists of three primary chemical reactions. There is close coordination between the project participants in developing the three component reaction sections — the H_2SO_4 decomposition section, done by Sandia National Labs; the HI decomposition section, done by General Atomics (GA); and the Bunsen reaction equipment, provided by CEA. Each participant has designed and constructed their respective section, and is working to integrate them in an SI Integrated Laboratory-Scale Experiment (ILS) to be conducted at the GA site in San Diego, CA. This experiment is on track to begin integrated operations in late 2007.

Through 2004 and 2005, experimental work in glass equipment was conducted to evaluate and choose appropriate methods for carrying out the reactions in each section. Design work in 2006 allowed for lab-scale devices to be constructed in 2007 from engineering materials that are expected to be used in a pilot-scale hydrogen production facility scheduled for operation beginning in 2013. These lab-scale devices make up the equipment of the ILS. Unlike previous demonstrations elsewhere, the ILS will operate at temperatures and pressures expected to be seen at larger scales. The ILS is expected to operate at least through the end of 2008.

The highly corrosive nature of chemical streams in the SI process has led to significant research work in the area of materials compatibility. Early screenings showed that alloys of tantalum appeared suitable, and current work is exploring long-term performance and corrosion resistance of materials stressed or machined in ways that materials of construction for larger scale plants will be subjected to. The ILS will be a test bed for corrosion resistance of engineering materials during its operation.

Other areas of research include membrane and catalyst development. High-temperature inorganic membranes are being developed for use in separation of SO_2 and O_2 from other chemical species in the high-temperature decomposition of H_2SO_4. This separation has the potential to shift the equilibrium of the reaction resulting in a potentially lower reaction temperature or increased process efficiency. The use of membranes for dewatering process streams is also being investigated. Most importantly, the removal of water from a mixture of water, elemental iodine, and hydriodic acid (HI) is being studied. Catalysts are also being developed that will be highly active and stable in the harsh acidic environments and high temperatures encountered in the SI cycle. Iron oxide catalysts for sulfuric acid decomposition are suitable at higher temperatures (above 870°C), and platinum-based catalysts can be used when the peak process temperature is below 870°C. Platinum-based catalysts are not suitable for use in HI decomposition reactors, but activated carbon catalysts have been shown to be effective and inexpensive.

The Japanese Atomic Energy Agency (JAEA) has also selected the SI process for further development and has successfully completed bench-scale demonstrations of the SI process at atmospheric pressure. JAEA also plans to proceed with pilot-scale demonstrations of the SI process and eventually plans to couple an SI demonstration plant to its High Temperature Test Reactor (HTTR). Development of the SI process is also being performed in South Korea by the Korea Institute of Energy Research (KIER) and the Korea Institute of Science and Technology (KIST). KAERI and DOOOSAN are also participating in the project, which is known as the Nuclear Hydrogen Production and Technology Development and Demonstration Project (NHDD).

1.4.2 High Temperature Electrolysis

The Solid Oxide Electrolyzer Cell (SOEC) is the key component of the HTE process. Two SOEC designs are under development internationally, a planar design and a tubular design. For comparison, a tubular type SOEC and a planar type SOEC are shown in Figure 1-3. A high-level description of each SOEC design is provided below.

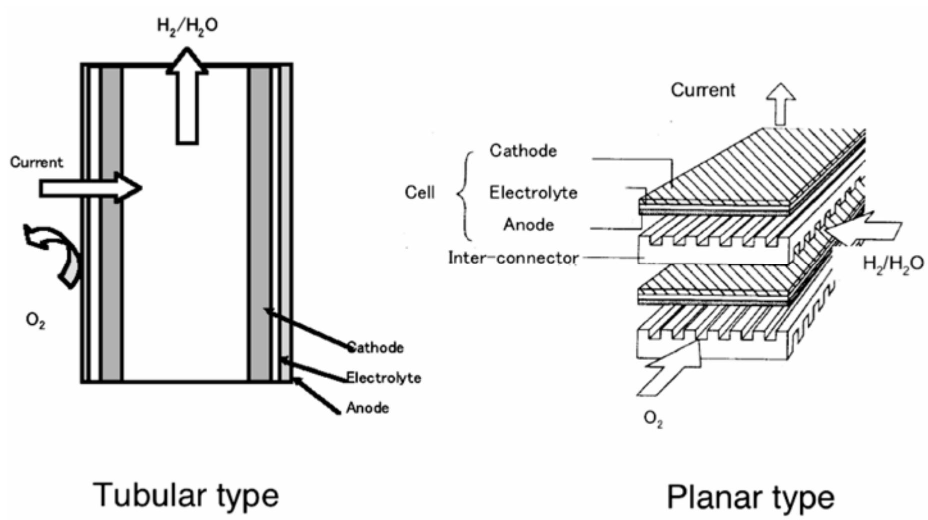

Figure 1-3. Comparison of Tubular-Type and Planar-Type HTE SOECs

1.4.2.1 Planar Cell Technology

SOE modules based on the planar cell technology are being developed under a collaborative project between Idaho National Laboratory (INL) and Ceramatec [Herring 2005]. Stacked assemblies of 100-mm × 100-mm cells have been tested successfully at INL. Figure 1-4 shows a schematic diagram of a unit cell. When operated at or near the thermal-neutral voltage (1.288 V at 850°C), the endothermic heat of reaction is balanced by ohmic heating in the electrolysis stack, such that no additional heat is required for the stack to maintain high temperature. The cell electrolyte is fabricated from either yttria- or scandia-stabilized zirconia. A 1.5 mm cathode plate made of nickel cermet material is bonded to one side of the electrolyte. A 0.05 mm anode plate is bonded to the other side of the electrolyte. The anode is composed of a mixed (i.e., both electronic and ionic) conducting perovskite, lanthanum manganate ($LaMnO_3$) material. Bipolar plates with a doped lanthanum chromite (e.g., $La_{0.8}Ca_{0.2}CrO_3$) are attached to the outside of the anode and cathode, and join the anode and cathode of adjacent units to form the stack. The bipolar plates also provide flow passages between each of the units in a stack for the steam-hydrogen mixture and separate passages for the steam/oxygen sweep gas. The relatively small active area of the individual cells is determined by the thermal expansion compatibility between the electrolyte and the electrodes.

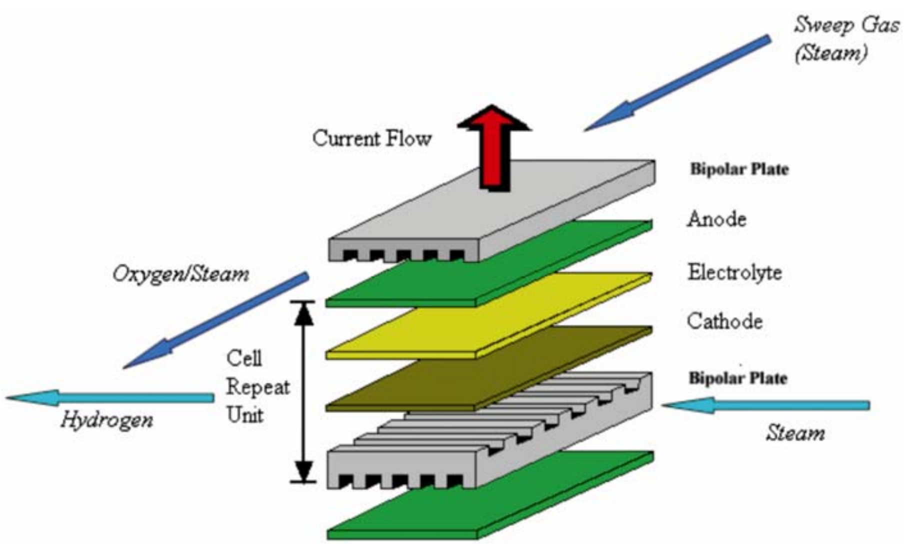

Figure 1-4. SOE Unit Cell Schematic

1.4.2.2 Tubular Cell Technology

Figure 1-5 shows the general configuration of a tubular SOEC module. It consists of an internally insulated pressure vessel housing electrolysis cells. Scale up of the SOEC hydrogen production process can be accomplished by having more pressure vessel modules or by using larger modules having larger pressure vessels containing more electrolysis cells. Figure 1-6 shows a tubular SOEC developed by Toshiba Corporation. The electrolyte is YSZ (Yttria-Stabilized Zirconia), the anode (oxygen electrode) is LSM (Strontium-doped Lanthanum Manganite), and the cathode (hydrogen electrode) is Ni-YSZ (a mixture of metallic Nickel and Yttria-Stabilized Zirconia).

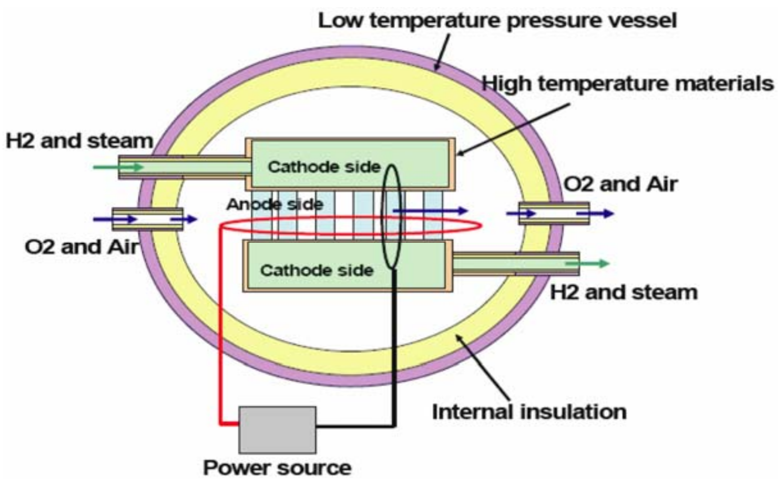

Figure 1-5. High Pressure SOEC Module Configuration

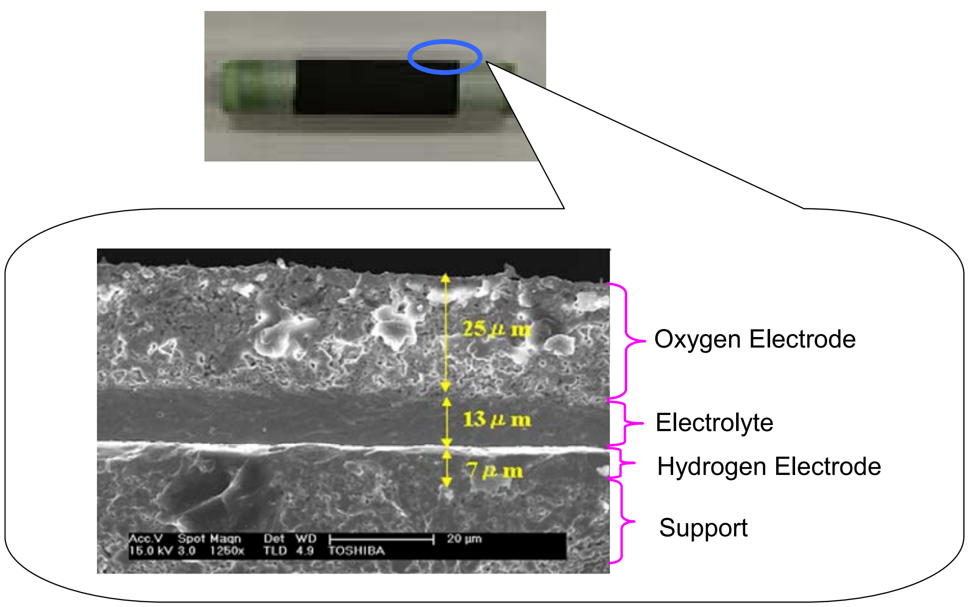

Figure 1-6. Tubular SOE Cell Developed by Toshiba Corporation

1.5 NGNP Trade Studies

The NGNP preconceptual design information presented herein is based on the requirements of the SRM [SRM 2007] and on the results of the reactor type comparison study [Baxter 2007], the reactor power level study [Labar 2007], and the heat transfer/transport system study [Bolin 2007]. An NGNP end-products study [Hanson 2007a] was also performed to develop the requisite data needed for the commercial plant economic assessment presented in Section 5.4. A brief summary of each study is presented below.

1.5.1 Reactor Type Comparison Study

The objective of the study was to identify the reactor type (i.e., either the PMR or PBR) that is best suited for the VHTR commercial mission of cogeneration of electricity and process heat for production of hydrogen using advanced, highly-efficient processes such as the SI and THE processes. It is important to note that the objective of the study was to identify the best choice for a commercial VHTR as opposed to identifying the design that best fits into the current schedule for the NGNP Project. This is because the best design for the VHTR should drive the selection of the NGNP design, and, hence, the NGNP project schedule, as opposed to the NGNP schedule driving selection of the NGNP design and, hence, the VHTR design.

A systematic comparison of the 600-MWt GT-MHR design and the 400-MWt PBMR-400 design (as described in the open literature) was performed against a set of evaluation criteria selected by GA based on the requirements for a commercial VHTR and the NGNP, and the perceived capability of the criteria to discriminate between the designs. The comparison revealed that the PMR has a clear advantage over the PBR as the MHR type best suited for commercial deployment. This is because the PMR inherently allows higher reactor power levels, which results in better plant economics. The PMR concept also has a clear advantage in that it involves fewer uncertainties (and therefore less risk) with respect to dust in the primary coolant circuit, core thermal/hydraulic performance, replacement of graphite reflector elements, and nuclear fuel accountability. The PMR also allows more flexibility with respect to the use of alternate fuel cycles, such as those fabricated from w-Pu or TRU from spent LWR fuel. Consequently, the PMR is also the best choice for the NGNP, and the NGNP preconceptual design presented herein is for a PMR.

1.5.2 Reactor Power Level Study

For the commercial VHTR plant, the reactor power level should be as high as possible for economy-of-scale reasons. The design of the 550/600 MWt metallic RV is at, or close to, the largest practical size that can be constructed based on current manufacturing capabilities. Selecting the commercial VHTR reactor power as 600 MWt with a reference commercial plant consisting of four reactor modules is projected to result in a commercial VHTR plant having a significant economic advantage relative to alternatives for electricity and/or process heat generation.

The study recommended that the NGNP be a full-size prototype of a commercial VHTR module having a power level of 600 MWt. This reactor size was judged to best satisfy the evaluation criteria that the NGNP should be designed such that construction, licensing, and operation of the NGNP would eliminate much of the uncertainty associated with utility/user costs to build, license, and operate a commercial VHTR. The elimination of such uncertainty was judged essential to demonstrate to potential utility/users that a VHTR would enjoy a significant cost advantage with respect to alternate means of electricity and/or process heat generation (without which there would be no incentive for a utility/user to build a VHTR.

The minimum sizes for the NGNP hydrogen production facilities, in terms of the thermal energy required from the reactor, were recommended to be ~4 MWt for an HTE-based plant and 60 MWt for an SI-based plant. These sizes were recommended to allow for ten SOEC modules in the HTE-based plant and three process trains of 20 MWt each in the SI-plant, which was deemed desirable to demonstrate process reliability and process control methodologies.

1.5.3 Heat Transfer/Transport System Study

Helium and the molten salt (MS) FLiNak, were evaluated and compared as potential working fluids for the NGNP secondary HTS. The evaluation focused on economics and technical risk. The capital cost of a He HTS was estimated to be about $16 million higher than a MS HTS primarily due to the cost of the helium circulator. However, the operating cost for a MS system was estimated to be substantially higher due to the higher replacement cost of a He-MS intermediate heat exchanger (IHX). Overall, the difference in cost was relatively insignificant compared to the total NGNP cost. However, the technical risk associated with use of molten salt was judged to be much greater than that associated with use of helium. For these reasons, it was concluded that there is no compelling reason to choose molten salt over high-pressure helium, particularly in view of the high-level NGNP Project requirement to use the lowest-risk technology consistent with satisfying the NGNP objectives. Thus, the NGNP preconceptual design presented herein uses high-pressure helium as the working fluid for the secondary HTS.

1.5.4 NGNP End-Products Study

The purpose of this study was to identify the NGNP commercial end-products (e.g., electricity, hydrogen, and oxygen) and by-products such as radioactive and chemical wastes, to evaluate potential management options for these products, and to assess the market value of the commercial end-products and the economic penalties associated with the by-products. This study was a prerequisite for the commercial plant economic assessment in that the end-product values and by-product economic penalties were inputs to the assessment. Table 1-1 lists the commercial value (in 2007$) established for each of the NGNP end-products in the 2020 – 2060 time frame.

Table 1-1. NGNP End-Product Commercial Value Predictions (in 2007$)

End-Product	NGNP Venue* 2020 – 2060	H2-MHR Venue** 2020 - 2060	Comments
Electricity (mil/kWh)	55	106	EIA forecast
Hydrogen ($/kg)	2.5	2.5	Set by natural gas price
Oxygen ($/tonne)	27	27	EPRI forecast

* In Idaho

** U.S. Gulf Coast location

2. OVERALL NGNP PLANT DESCRIPTION

2.1 Summary of NGNP Plant Design

The nuclear heat source for the NGNP consists of a single 600-MW prismatic MHR module with two primary coolant loops for transport of the high-temperature helium exiting the reactor core to a direct cycle PCS and to an IHX (Figure 2-1). The reactor design is essentially the same as for the GT-MHR, but includes the additional primary coolant loop to transport heat to the IHX and other modifications to allow operation with a coolant-outlet temperature of 950°C (vs. 850°C for the GT-MHR). The IHX transfers a nominal 65 MW of thermal energy to the secondary heat transport loop, which transports the heat energy to both an SI-based hydrogen production facility (60 MW) and an HTE-based hydrogen production facility (~4 MW). Figure 2-2 shows a schematic process flow diagram of the NGNP preconceptual plant design. Table 2-1 summarizes the key design features.

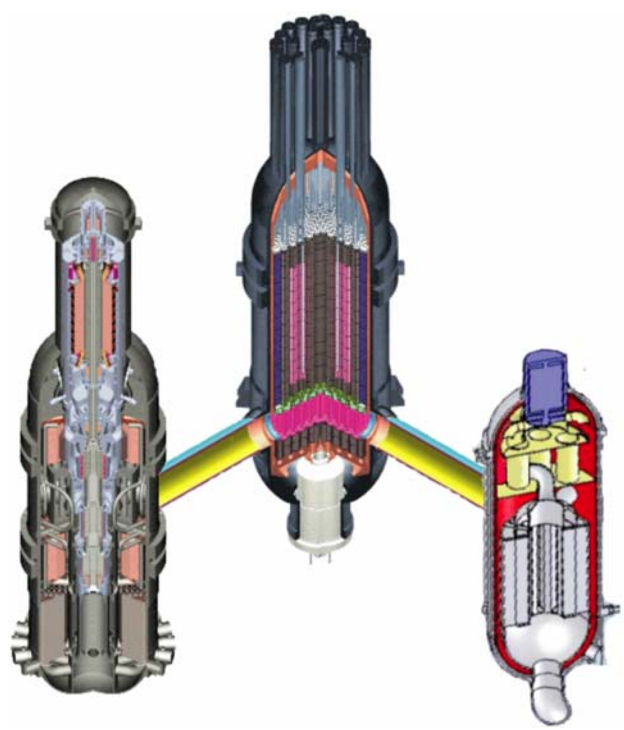

Figure 2-1. The MHR Module is Connected to a Direct Cycle PCS and an IHX

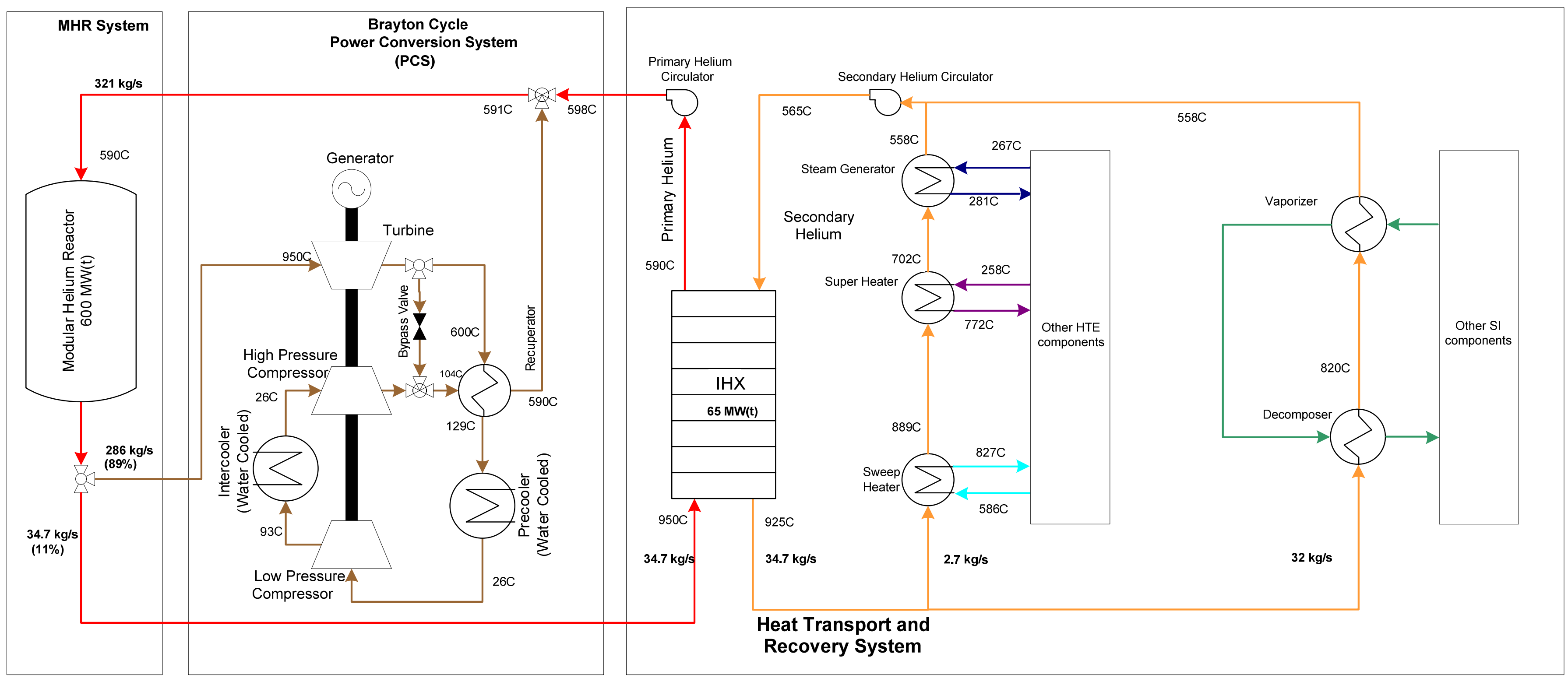

Figure 2-2. Schematic Process Flow Diagram of NGNP Plant

Table 2-1. Key Features of NGNP Design

Design Attribute	Design Selection
Reactor type	Prismatic block
Reactor power level	600 MWt
Fuel	Initial Core: TRISO-coated 500-µm UO_2 (~9.9% enriched) Reloads: TRISO-coated UCO (likely with two or more U-235 enrichments)
Power conversion cycle	Reference: Direct Brayton cycle with GA/OKBM vertical integrated PCS Alternate: Direct combined cycle (steam cycle with a GT topping cycle) as proposed by Rolls-Royce
Core outlet/inlet coolant temperatures	Reference: 950ºC/590ºC Alternate: 950ºC/510ºC
Vessel materials	Reactor: 2¼Cr – 1Mo PCS: SA508 IHX: 2¼Cr – 1Mo Crossduct: 2¼Cr – 1Mo Hot duct: Allow 617 or alloy 800H
Primary loop inlet/outlet pressure	7.07MPa/7.0 MPa (Electricity mode)
Number of loops	3 (PCS loop, primary heat transport loop, and secondary heat transport loop)
Primary coolant	Helium
Secondary loop working fluid	Helium
Heat transferred to secondary loop	65 MWt
Intermediate heat exchanger type (and LMTD)	Reference: Printed circuit (25°C) Backup: Helical coil (91°C)
Hydrogen production process	SI requiring 60 MWt thermal energy HTE requiring ~4 MWt thermal energy
Heat rejection	Dry cooling tower

2.2 Plant Level Functions and Performance Requirements

The topmost requirements for the NGNP include the project mission as defined in the Energy Policy Act of 2005 and the NGNP Project objectives as defined by DOE/INL in the NGNP PPMP. At the next level are the high-level functions and requirements defined by INL [Functions & Requirements 2003], as modified based on the recommendations of the Independent Technology Review Group (ITRG) [ITRG 2004]. These high-level functions and requirements will be accomplished through implementation of plant-level requirements derived from the INL high-level requirements and other institutional sources such as utility/user requirements for commercial reactors, or that are developed though plant-level functional analysis including trade studies, plant performance analyses, engineering decisions, etc.

The high-level functions for the NGNP as defined in [Functions & Requirements 2003] are as follows:

- Develop and demonstrate a commercial-scale prototype VHTR
- Develop and demonstrate the production of electricity at high efficiencies
- Obtain licenses and permits to construct/operate the NGNP
- Develop and demonstrate the capability for efficient production of hydrogen
- Enable the demonstration of energy products and processes
- Provide capability for future testing to enhance plant safety and operational performance

Document DOE-GT-MHR-100248 [Utility/User Requirements 1995] provides extensive Utility/user requirements for a commercial GT-MHR. These requirements were developed from Utility requirements for advanced light water reactors (ALWRs) [ALWR Requirements 1991], input provided by constituents of the GT-MHR Program, and pertinent information from IAEA-TECDOC-801, “Development of Safety Principles for the Design of Future Nuclear Power Plants”. GA had discussions with members of GA’s Utility Advisory Board and Academic Advisory Group, who made the following recommendations with respect to the mission of the NGNP.

a. The NGNP should be a full-size prototype of a commercial VHTR module.
b. The initial power level for the NGNP could be somewhat lower than the power level for a commercial MHR module, but the NGNP should be designed for up-rating to the full commercial MHR module power level
c. The mission of the NGNP should include demonstration of process-heat applications, including steam methane reforming for hydrogen production. From a utility/users viewpoint, process heat applications are a more important near-term mission than demonstration of hydrogen production.
d. The NGNP should be capable of demonstrating use of alternate fuels, including Pu-based and actinide-based fuel (i.e., “deep-burn” fuel) from re-processed LWR spent fuel.
e. The NGNP should be designed to demonstrate a commercial MHR that meets the key Utility/User design requirements for a commercial MHR

Based on the above institutional requirements, GA has defined preliminary plant and system-level requirements for the NGNP in the SRM. As defined by INL, the primary purpose of the

SRM at this early stage of the project is "to define the design independent high-level requirements that establish the framework within which subsequent work will be performed to establish the specific design attributes of the NGNP (e.g., type of reactor, direct versus indirect power conversion, hydrogen production processes, etc.)". However, in recognition of past DOE-sponsored work by GA that has resulted in a relatively-mature definition of the GT-MHR concept[1] and in preconceptual designs for both SI-based and HTE-based commercial H2-MHR plants, GA expanded the scope of the initial version of the SRM to include lower-level, design-specific requirements for the NGNP based on the GT-MHR and the H2-MHR designs. Although the systems, the functions of the systems, and the design-specific requirements for these systems defined in the initial version of the SRM are preliminary in nature, GA included this information to provide guidance to the NGNP pre-conceptual design effort and to establish a methodology and framework for further development of the requirements for the NGNP during conceptual design.

2.3 Overall Plant Arrangement

A layout for the NGNP plant is shown in Figure 2-3. The plant layout consists of the Reactor Building (RB), the two hydrogen generation plants and several support buildings and facilities. Systems containing radionuclides and safety-related systems are located in the Nuclear Island, which is separated physically and functionally from the remainder of the plant. A key consideration for safety and licensing of the NGNP is co-location of the MHR module with the hydrogen production plants. It is proposed to locate the two hydrogen production facilities at a distance of 90 meters from the MHR in order to limit the distance over which high-temperature heat is transferred. This separation distance is consistent an INL engineering evaluation that concluded that separation distances in the range of 60 m to 120 m should be adequate in terms of safety [INL 2006]. No earthen berm or blast suppression barrier is considered necessary between the hydrogen production facilities and the reactor with a separation distance of 90 meters because the reactor is below grade. However, the hydrogen production facilities are surrounded by a low berm, which serves as a chemical spill retention barrier. The plant also includes a below-grade hydrogen storage tank for on-site storage of up to 100 kg of hydrogen, which is the limit suggested in [INL 2006], and space for a large dry-cooling tower.

Because of uncertainties associated with potential uses of the NGNP, the preconceptual design of the plant includes features that allow some flexibility in adjusting the mission of the NGNP. As shown in Figure 4-3 in Section 4, the below-grade concrete portion of the reactor building is

[1] The previous work on the GT-MHR included essentially the same concept selection studies that are being performed currently as part of the scope of NGNP preconceptual design.

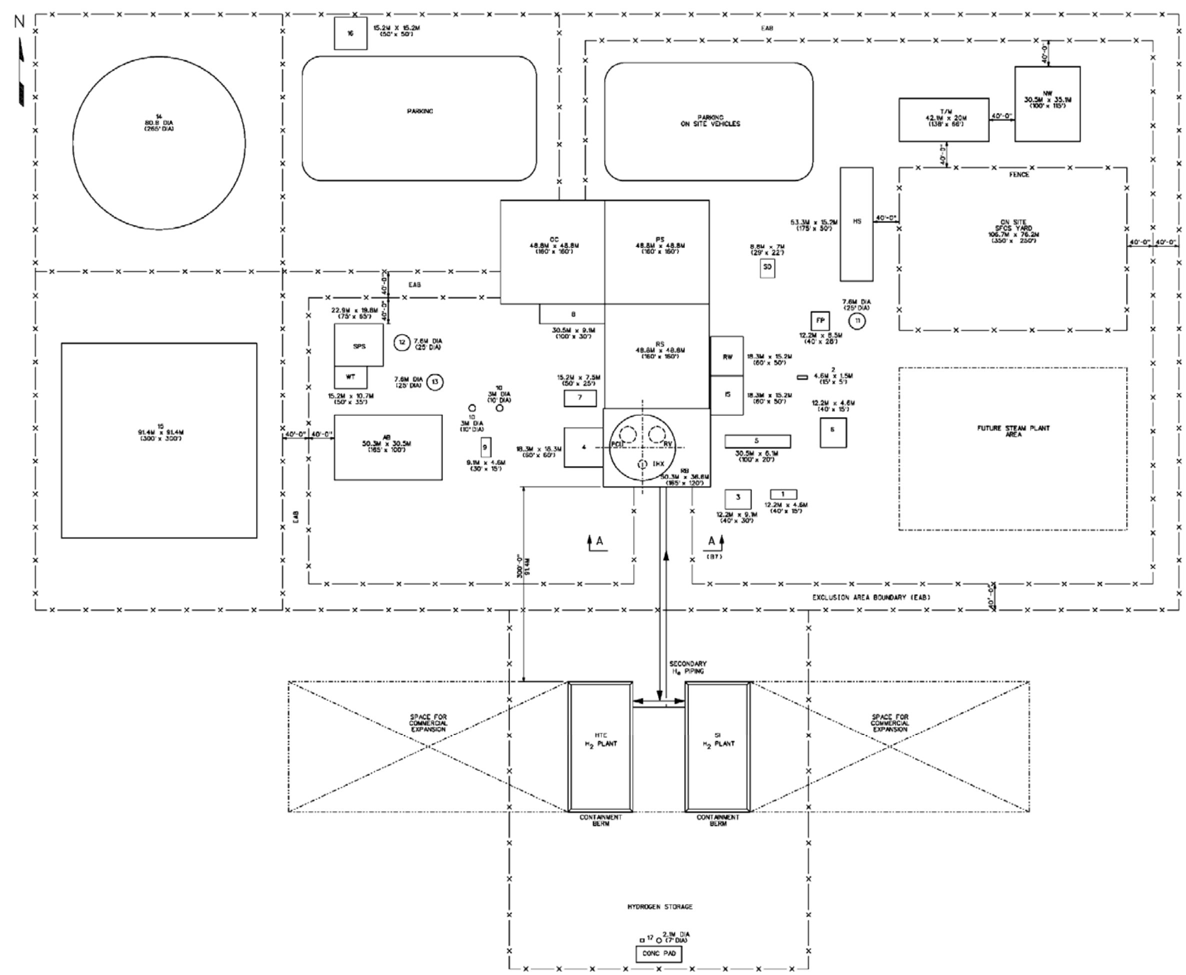

Figure 2-3. NGNP Plant Layout

designed to include a cavity for an IHX that is large enough to accommodate a printed-circuit type heat exchanger of much greater size than needed to transfer 65 MW of heat energy. Also, the cross vessel from the RV to the IHX vessel and the nozzle on the RV side are the same diameter as for the RV-to-PCS cross vessel. For initial NGNP operation with 65 MW of heat energy being transferred to the IHX, the RV-to-IHX vessel cross vessel will have a conical area reducing section to reduce the cross vessel diameter to the size needed for the small IHX. However, in the event that the small IHX is replaced with a larger IHX, the area reducing section and smaller diameter portion of the cross vessel adjoining the IHX vessel will be replaced with a larger-diameter cross vessel. Also, the plot plan includes extra space to allow for addition of a facility for some yet-to-be-determined process heat or process steam application, and for expansion of the hydrogen facilities to increase hydrogen production capacity.

2.4 Nominal Plant Design Parameters

The NGNP must be designed for both electricity-only production and for cogeneration of electricity and process heat to satisfy the requirement that the NGNP be capable of generating electricity, hydrogen, or both electricity and hydrogen. When the NGNP is operating in cogeneration mode with 65 MWt of the 600 MWt reactor output diverted to the IHX, the maximum system pressure is reduced from 7 MPa to 6.23 MPa, which reduces the electrical output of the PCS while maintaining high efficiency. The reduced pressure and density increases the pressure drop in the reactor core and PSR coolant holes from 80 kPa to 90 kPa. Tables 2-2 and 2-3 give the nominal plant design parameters for both modes for operation of the reactor with outlet gas temperatures of 850°C and 950°C, respectively.

Two of the major design decisions for the NGNP include selection of the reactor outlet helium temperature and the choice of an indirect or direct power conversion cycle. Both of these decisions have a major impact on the overall NGNP plant design and on the level of programmatic risk. These key issues are discussed below.

2.4.1 Reactor Outlet Gas Temperature

The original goal with respect to the reactor outlet coolant temperature for the NGNP was 1000°C [Functions & Requirements 2003]. However, based on its review of the NGNP Program, the ITRG concluded that there are materials development risks at a reactor outlet temperature of 1000°C that make it impossible to achieve an operational date of 2020 (let alone 2018) for the NGNP. The ITRG identified the IHX, hot duct, turbine inlet components, and in-core metallic materials as being the high-risk components. Consequently, the ITRG recommended that the NGNP start operation with a reactor outlet gas temperature of 900°C to 950°C in order to maintain maximum metal temperatures at or below 900°C. The ITRG also

Table 2-2. NGNP Nominal Plant Design Parameters
850°C Reactor Outlet Gas Temperature

	Electricity Only	Cogeneration
MHR System		
Power rating	600 MWt	600 MWt
Core inlet/outlet temperatures	490°C / 850°C	490°C / 850°C
Peak fuel temperature – normal operation	<1250°C	<1250°C
Peak fuel temperature – accident conditions	< 1600°C	< 1600°C
Helium mass flow rate	321 kg/s	321 kg/s
MHR System pressure	7.0 MPa	6.4 MPa
Power Conversion System		
Mass flow rate	321 kg/s	286 kg/s
Heat supplied from MHR System	600 MWt	535 MWt
Turbine inlet/outlet temperatures	848°C / 510°C	848°C / 510°C
Turbine inlet/outlet pressures	7.0 MPa / 2.6 MPa	6.2 MPa / 2.3 MPa
Electricity generation efficiency*	47.5%	47.5%
Heat Transport System		
Primary helium flow rate	N/A	35 kg/s
Secondary helium flow rate	N/A	35 kg/s
IHX heat duty	N/A	65 MWt
IHX primary side inlet/outlet temperatures	N/A	850°C / 490°C
IHX secondary side inlet/outlet temperatures	N/A	825°C / 465°C
HTE-based Hydrogen Production System		
Peak SOE temperature	N/A	TBD°C
Peak SOE pressure	N/A	TBD MPa
Product hydrogen pressure	N/A	TBD MPa
Annual hydrogen production	N/A	TBD
Plant hydrogen production efficiency**	N/A	TBD%
SI-based Hydrogen Production System		
Peak process temperature	N/A	~825°C
Peak process pressure	N/A	TBD MPa
Product hydrogen pressure	N/A	TBD MPa
Annual hydrogen production	N/A	TBD
Plant hydrogen production efficiency**	N/A	~42%

*Neglects parasitic heat losses from the RCCS and SCS
**Based on the higher heating value of hydrogen (141.9 MJ/kg)

Table 2-3. NGNP Nominal Plant Design Parameters
950°C Reactor Outlet Gas Temperature

	Electricity Only	Cogeneration
MHR System		
Power rating	600 MWt	600 MWt
Core inlet/outlet temperatures	590°C / 950°C	590°C / 950°C
Peak fuel temperature – normal operation	1250°C - 1350°C	1250°C - 1350°C
Peak fuel temperature – accident conditions	< 1600°C	< 1600°C
Helium mass flow rate	321 kg/s	321 kg/s
MHR System pressure	7.0 MPa	6.4 MPa
Power Conversion System		
Mass flow rate	321 kg/s	286 kg/s
Heat supplied from MHR System	600 MWt	535 MWt
Turbine inlet/outlet temperatures	948°C / 617°C	948°C / 617°C
Turbine inlet/outlet pressures	7.0 MPa / 3.0 MPa	6.2 MPa / 2.6 MPa
Electricity generation efficiency*	50.5%	50.5%
Heat Transport System		
Primary helium flow rate	N/A	35 kg/s
Secondary helium flow rate	N/A	35 kg/s
IHX heat duty	N/A	65 MWt
IHX primary side inlet/outlet temperatures	N/A	950°C / 590°C
IHX secondary side inlet/outlet temperatures	N/A	925°C / 565°C
HTE-based Hydrogen Production System		
Peak SOE temperature	N/A	862°C
Peak SOE pressure	N/A	5.0 MPa
Product hydrogen pressure	N/A	4.95 MPa
Hydrogen production rate	N/A	6,000 Nm^3/h
Plant hydrogen production efficiency**	N/A	~53%
SI-based Hydrogen Production System		
Peak process temperature	N/A	900°C
Peak process pressure	N/A	6.0 MPa
Product hydrogen pressure	N/A	4.0 MPa
Hydrogen production rate	N/A	9,000 Nm^3/h
Plant hydrogen production efficiency**	N/A	~45%

*Neglects parasitic heat losses from the RCCS and SCS; neglects reduction in efficiency due to turbine blade cooling
**Based on the higher heating value of hydrogen (141.9 MJ/kg)

outlined a potential approach for up-rating the NGNP to operate with a reactor outlet gas temperature of 1000°C. This approach calls for the NGNP to be designed to allow for replacement of the critical metal components with higher-temperature materials developed and qualified in parallel with construction and early operation of the NGNP.

In addition to discussing the technical issues associated with a reactor outlet gas temperature greater than 950°C, the ITRG questioned the practicality of attempting to achieve a gas temperature of 1000°C in the NGNP. The ITRG also questioned the significance of any economic benefit to be gained by increasing the reactor outlet gas temperature to 1000°C and recommended against embarking on the extensive and costly research and development program that would be necessary to achieve such a goal unless the need to do so could be justified on an economic basis.

The GA Team agrees with the ITRG's assessment and suggests that the same logic could be applied in questioning the need for a reactor outlet gas temperature of 950°C versus 900°C, or even 850°C given the relatively small difference in efficiency of either the SI or HTE hydrogen production processes at temperatures of 825°C and 925°C. Consequently, the GA Team recommends an approach for NGNP operation that is similar to that recommended by the ITRG, except that initial operation of the NGNP should be with a reactor outlet gas temperature of 850°C. However, all of the NGNP plant systems should be designed to operate with a reactor outlet gas temperature of up to 950°C. After an initial operating period with a reactor outlet gas temperature of 850°C, the plant could be up-rated to operate with a reactor outlet gas temperature of 900°C to 950°C depending on the results of economic evaluations performed to justify such operation and on the success of the R&D programs to develop materials and/or qualify designs for higher-temperature operation.

2.4.2 Direct vs. Indirect Power Conversion Cycle

The ITRG reviewed the design, fabrication, and operation of the various power conversion systems proposed by the proponents of the NGNP, including the vertical integrated PCS design developed by GA/OKBM, with the objective of identifying the major risks associated with these systems. Based on this review, the ITRG concluded that given the "large number and formidable nature of the risks associated with the direct cycle and their potential impact on the NGNP schedule," the "NGNP should proceed on the basis of a lower-risk indirect cycle". The ITRG acknowledged that the indirect cycle necessitates use of a large IHX, which is itself a component having significant developmental risks, but the ITRG considered these risks to be more manageable than the aggregate of the risks associated with the direct cycle concepts.

However, the ITRG also concluded that it is highly unlikely that a metallic material will become available that will not require replacement of the IHX at least once, and probably more than once, during the plant life, even for an operating temperature limited to 900°C. The ITRG further concluded that it is not clear that a metallic material will become available for IHX operation at 1000°C on any reasonable time scale, or at all, and that it is unlikely that an IHX could be fabricated from a ceramic material. Thus, the ITRG concluded that from a material standpoint, operation at 1000°C or higher favors the use of the direct cycle.

GA and its team members are well aware of the technical challenges associated with the GT-MHR vertical integrated PCS design and appreciate the ITRG's concerns. Nevertheless, the GA Team remains convinced that a direct power conversion cycle is a better choice for the NGNP than an indirect power conversion cycle, and the NGNP preconceptual design presented herein reflects this choice. The reasons for this design selection are as follows:

- Past GA studies of direct vs. indirect cycle (albeit for an MHR for electricity generation), have concluded that a direct cycle is the clear choice over an indirect cycle for electricity generation at low cost.

- The GA Team was not tasked to perform the PCS trade study that was included in the statement of work for the original NGNP preconceptual engineering services solicitation. Consequently, the GA team has not developed any indirect cycle concepts nor systematically evaluated the pros and cons of a direct cycle versus an indirect cycle for a MHR whose primary purpose is to provide process heat (as opposed to electricity generation). Consequently, the GA team has no basis for replacing its direct cycle GT-MHR design with an indirect cycle design.

- GA is not convinced that the technical risks associated with a 600-MWt IHX and a 600-MWt helium circulator are more manageable than the risks associated with the vertical integrated PCS design. Furthermore, GA suspects that the cost associated with a 600-MWt IHX capable of operating at 950°C may be prohibitive.

- Construction of the NGNP with an indirect cycle would likely preclude operation of the NGNP with a gas outlet temperature of 1000°C because, as concluded by the ITRG, it is unlikely that an IHX can be fabricated for operation at 1000°C in any reasonable time frame, if at all. Although it is doubtful that operation of the NGNP at 1000°C reactor outlet temperature can be justified on an economic basis (Section 2.4.1), it would be desirable if the NGNP design does not preclude such operation.

- An extensive effort is in progress under the U.S./Russian International GT-MHR Program to develop and demonstrate the vertical integrated PCS design. If this program proceeds according to schedule, this PCS design will be fully demonstrated in time for deployment in the NGNP.

- Rolls-Royce has reviewed the GA/OKBM vertical integrated PCS design and has made a number of recommendations for design modifications that could improve the design and reduce the cost and/or risk associated with the design[2] (Section 3.6.2).

- Rolls-Royce has developed a pre-conceptual design for a direct combined cycle PCS as a backup to the reference vertical integrated PCS design. This design eliminates or reduces some of the more significant risks in the reference design, but would add to the complexity and cost of the plant. This design requires further evaluation and development during conceptual design, but appears to be a viable backup for the reference design should the need for a fallback become apparent based on results from the OKBM design demonstration program.

2.5 Plant Operation

The NGNP must be designed for both electricity-only production and for cogeneration of electricity and process heat to satisfy the requirement that the NGNP be capable of generating electricity, hydrogen, or both electricity and hydrogen. Tables 2-2 and 2-3 give the nominal plant design parameters for both operating modes for operation of the reactor with reactor outlet helium temperatures of 850°C and 950°C, respectively. As discussed in Section 2.4.1, GA recommends that initial operation of the NGNP be with a reactor outlet helium temperature of 850°C.

The two primary coolant loops and the requirement to operate in either an electricity-only mode or a cogeneration mode introduce some complexity into the plant design and operation that will have to be addressed during subsequent design phases. However, GA's preliminary evaluation of plant operation as discussed in Section 3.10 provides reasonable confidence that the plant can be operated in either mode. The plant will require a helium inventory control system for the secondary heat transport loop and a plant control system that is designed to include the necessary instrumentation and controls to make the necessary simultaneous adjustments to the primary and secondary helium inventories to maintain the pressure difference across the IHX within acceptable limits.

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[2] The Rolls-Royce recommendations need to be jointly evaluated in more detail by Rolls-Royce, OKBM, and GA during conceptual design.

3. PLANT TECHNICAL DESCRIPTION

This Section provides a technical description of the entire NGNP plant, including the nuclear systems, the PCS, the HTS, the hydrogen production facilities, the Helium Services System, the Plant Operation and Control System, and the Balance of Plant (BOP). The nuclear systems include the Reactor System, the Vessel System, the Shutdown Cooling System (SCS), the Fuel Handling System, and the Reactor Cavity Cooling System (RCCS).

3.1 Reactor System

3.1.1 System Configuration

The NGNP nuclear heat source will be a single MHR module based on the GT-MHR design with some modifications to permit operation with a reactor outlet helium temperature of 950°C (vs. 850°C for the GT-MHR). Figure 3-1 shows a cross-sectional view of an MHR. Figure 3-2 shows a cross section of the GT-MHR core at vessel midplane. The reactor power level is 600 MWt; the core thermal power density is 6.6 MWt/m^3. The MHR active core consists of 102 fuel columns in three annular rings with 10 fuel elements per fuel column, for a total of 1020 fuel elements in the active core. The effective inner diameter and outer diameter of the active core are 2.96 m and 4.83 m, respectively. The active core height is 7.93 m.

In addition to the fuel elements, other graphite reactor internal components include the side, central, top, and bottom graphite reflector elements and the graphite core support assembly. Metallic reactor internal components include the metallic core support, the upper core restraint, and the upper plenum shroud. These metallic components are manufactured from high-temperature alloys (e.g., Incoloy 800H, Hastelloy-X, or Inconel 617).

From top to bottom, the graphite core support assembly consists of two layers of hexagonal elements, support pedestals for the fuel and reflector columns that form the lower plenum, and the lower plenum floor, which consists of a layer of graphite elements and two layers of ceramic elements that insulate the metallic core support from the hot helium in the lower plenum. The upper core restraint elements have the same hexagonal cross sections as the graphite elements below them and are one-half the height of a standard fuel element. Dowel/socket connections are used to align the core-restraint elements with the graphite blocks. The core restraint elements are also keyed to each other and to the core barrel. The upper core restraint blocks provide stability during refueling and maintain relatively uniform and small gaps between columns during operation. The metallic core support surrounds the core and includes a floor section and a core barrel that are welded together. The metallic core support is supported both vertically and laterally by the RV. The upper plenum shroud is a welded, continuous dome that rests on top of the core barrel to form the upper plenum. The upper plenum shroud includes penetrations for inserting control rods and reserve shutdown material, for refueling, and for core component replacement.

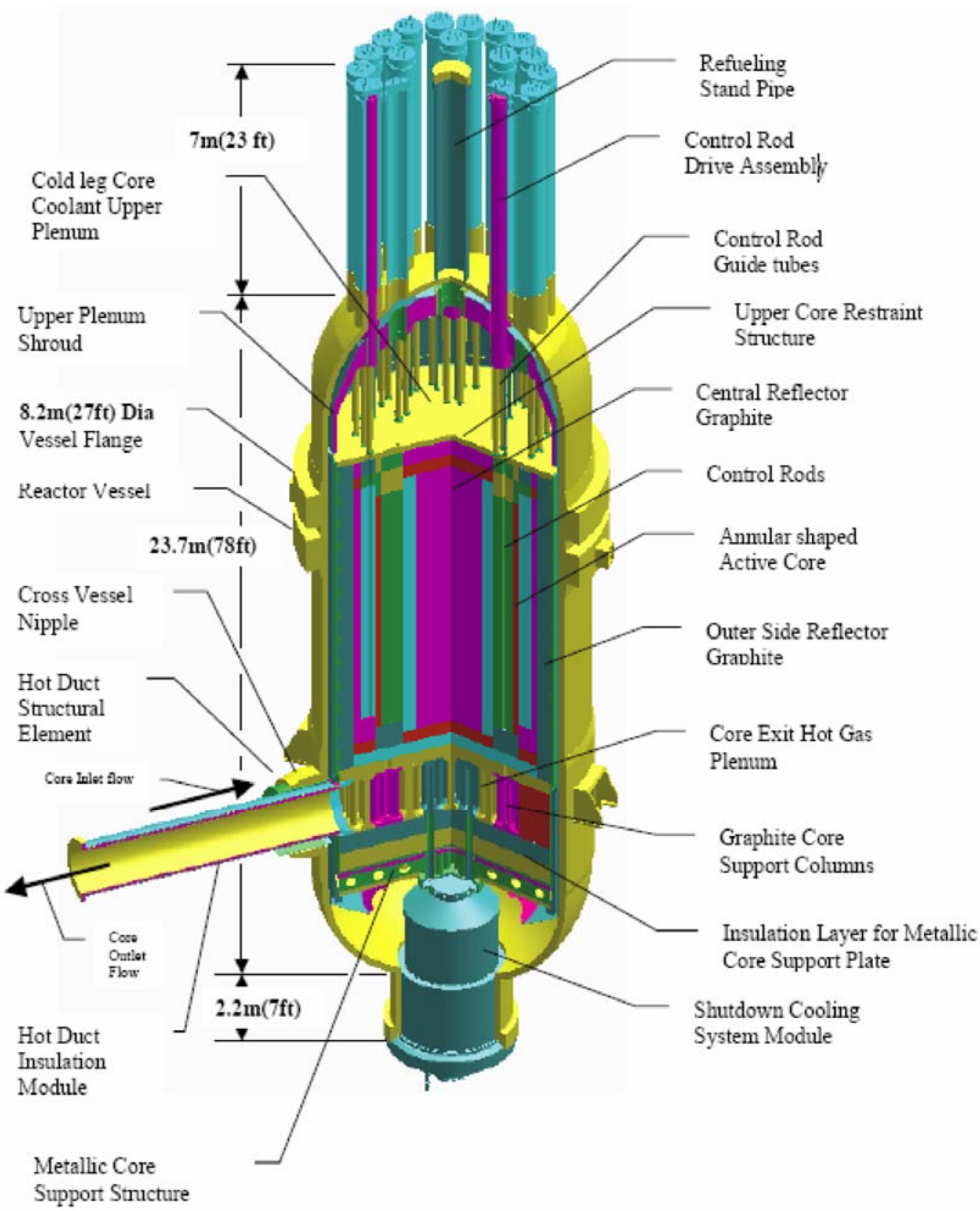


Figure 3-1. Cross-Sectional View of MHR

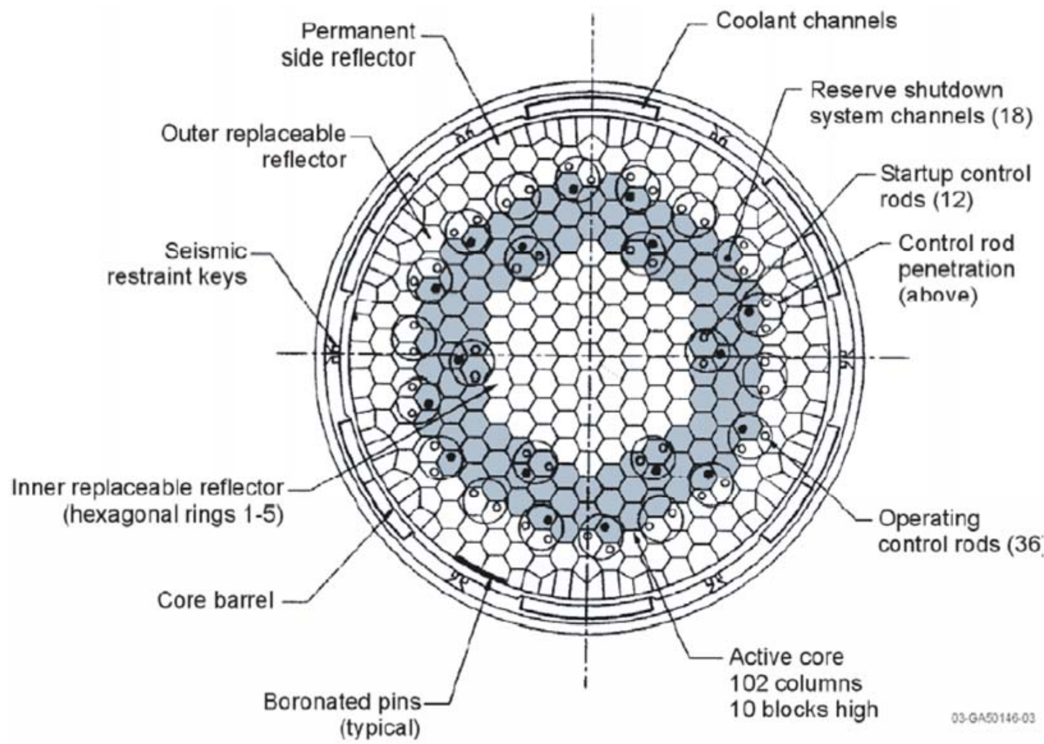


Figure 3-2. MHR Core Cross Section at Vessel Midplane

3.1.2 GT-MHR Fuel Design

Figure 3-3 shows the NGNP MHR fuel element and its components. The fuel for the GT-MHR consists of microspheres of uranium oxycarbide (UCO) that are coated with multiple layers of pyrolytic carbon (pyrocarbon) and silicon carbide. The buffer, inner pyrolytic carbon (IPyC), silicon carbide (SiC), and outer pyrolytic carbon (OPyC) layers are referred to collectively as a TRISO coating. The coating system has been engineered to serve as a miniature pressure vessel that provides containment of radionuclides and gases during normal operation and all design basis events (DBEs). This coating system is also an excellent engineered barrier for long-term retention of radionuclides in a repository environment.

The UCO kernel composition was selected for the GT-MHR because of its ability to perform well at relatively high burnup. The carbide component of the kernel undergoes oxidation to getter excess oxygen released during fission. If the carbide component were not present, excess oxygen would react with carbon in the buffer to form carbon monoxide. High levels of carbon monoxide can lead to failure of the coating system by overpressurization or by the amoeba effect (i.e., kernel migration). The oxide component of the kernel is highly effective at retaining many radionuclides that can chemically attack or diffuse through the coating layers.

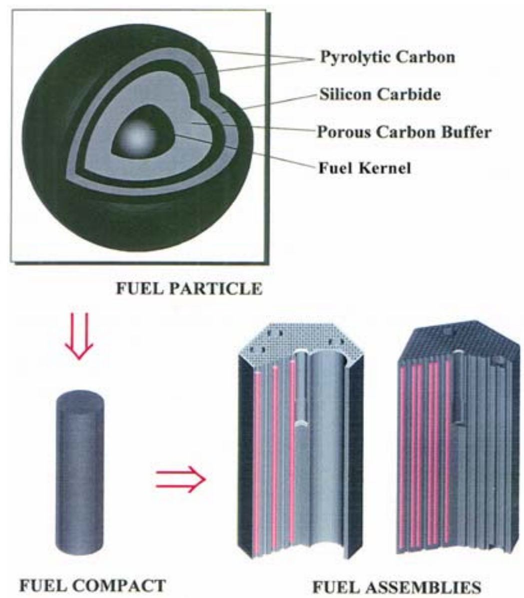


Figure 3-3. GT-MHR Fuel Element Components

The GT-MHR core is designed to use a blend of two different particle types; a fissile particle that is enriched to 19.8% U-235 and fertile particle with natural uranium (NU, enrichment of 0.7% U-235). The fissile/fertile loading ratio is varied with location in the core, in order to optimize reactivity control, minimize power peaking, and maximize fuel cycle length. The GT-MHR coated particle design parameters are given in Table 3-1. The fissile and fertile particle designs are somewhat different, with the fertile particle having a larger kernel and a thinner buffer coating layer. Preliminary core physics calculations performed by INL for an NGNP prismatic MHR suggest that the reactor may be able to utilize a single fuel particle design, with the fuel particles potentially having different U-235 enrichments. However, more detailed calculations are needed to confirm that a single fuel particle design provides adequate core design flexibility.

The TRISO fuel particles are bonded together in a carbonaceous matrix to form cylindrical compacts having nominal dimensions 12.45 mm (0.49 in.) in diameter and 49.3 mm (1.94 in.). The fuel compacts are stacked in the blind fuel holes of the graphite fuel element. Graphite plugs are cemented into the tops of the fuel holes to enclose the stacked compacts. The stacks under each of the four dowels per graphite fuel element block contain 14 fuel compacts; all other stacks contain 15 fuel compacts. Because of sorption mechanisms, the fuel compacts can provide an additional barrier to the release of metallic fission products.

Table 3-1. GT-MHR Coated Particle Design Parameters

	Fissile Particle	**Fertile Particle**
Composition	$UC_{0.5}O_{1.5}$	$UC_{0.5}O_{1.5}$
Uranium enrichment, %	19.8	0.7 (Natural Uranium)
Dimensions (μm)		
Kernel Diameter	350	500
Buffer thickness	100	65
IPyC thickness	35	35
SiC thickness	35	35
OPyC thickness	40	40
Particle diameter	770	850
Material Densities (g/cm^3)		
Kernel	10.5	10.5
Buffer	1.0	1.0
IPyC	1.87	1.87
SiC	3.2	3.2
OPyC	1.83	1.83
Elemental Content Per Particle (μg)		
Carbon	305.7	379.9
Oxygen	25.7	61.6
Silicon	104.5	133.2
Uranium	254.1	610.2
Total particle mass (μg)	690.0	1184.9
Design burnup (% FIMA) [a]	26	7
[a]Fissions per Initial Metal Atom (FIMA)		

The graphite blocks are fabricated from high-purity, nuclear-grade graphite. Each block is a right hexagonal prism with dimensions 794 mm (31.2 in.) in length and 360 mm (14.2 in.) across the flats of the hexagonal cross section. Fuel and coolant holes run parallel through the length of the block in a regular triangular pattern of nominally two fuel holes per coolant hole. The pitch of the coolant and fuel-hole array is 18.8 mm (0.74 in.). The minimum web thickness between a coolant hole and fuel hole is 4.5 mm (0.18 in.). This web provides an additional barrier to release of metallic fission products. A standard fuel element has 210 blind fuel holes, 108 coolant holes, and contains 3126 fuel compacts. The GT-MHR active core also contains some fuel elements having a single 4.0 in. diameter channel to allow for insertion of control rods and two 3.75 in. diameter channels for insertion of reserve shutdown control (RSC) material. A control/RSC element has 186 blind fuel holes, 95 coolant holes, and contains 2766 fuel compacts.

3.1.3 Use of Japanese Fuel in NGNP

As stated above, the reference fuel type for the NGNP preconceptual design is UCO. However, there is no current supplier of UCO fuel that can support startup of the NGNP in the required time frame (i.e., by 2018). Furthermore, there are no current domestic U.S. sources for fabrication of large quantities of coated-particle fuel of any type, such as would be required for the NGNP. Consequently, GA has formulated a fuel acquisition strategy for the NGNP based on obtaining TRISO-coated UO_2 for the first core fuel load (and possibly one or more reload segments) from Nuclear Fuel Industries (NFI) in Japan, which has a large-scale, coated-particle fuel manufacturing capability for the HTTR.

Under the envisioned plan, NFI would fabricate the kernels, coated particles, and fuel compacts for the initial core of the NGNP and send the fuel compacts to the U.S. for loading into the graphite fuel blocks. The NFI extended burnup fuel particle design was selected rather than the reference HTTR fuel particle design because this fuel particle is designed for irradiation to higher burnup and is more consistent with the reference German fuel particle design. Table 3-2 summarizes the physical properties of the two NFI fuel particle types and compares them to the reference German particle and to the reference fuel particle for the DOE AGR Fuel Development and Qualification Program (AGR Fuel Program) as defined in the preliminary AGR Fuel Product Specification [AGR Fuel Spec. 2004]. The as-manufactured quality of the NFI fuel would be consistent with the desired quality level for NGNP as specified in the preliminary AGR fuel product specification.

The primary implications of this approach are that the kernel will be UO_2 (rather than UCO), the U-235 enrichment will be limited to 10%, the fuel compacts will be made using the HTTR matrix material, and the particle packing fraction in the fuel compacts will be limited to about 35%. GA has evaluated two different reload strategies for a 10% LEU core and determined both to be feasible: (1) operate initial cycle through 425 EFPD and then reload the entire core with U.S. made fuel, and (2) reload at ~300 EFPD intervals with 10% LEU fuel fabricated by NFI. If a U.S. fuel source is available soon after NGNP startup, NFI strategy 1 would be implemented. Otherwise, NFI strategy 2 would be implemented continuously until the U.S. fuel source is available. The results also indicated that the NGNP loaded with NFI made fuel, with some further optimization of fuel and burnable poison loadings/zonings, can meet a satisfactory core physics design with respect to power peaking and fast fluxes.

Table 3-2. Physical Properties of NFI Fuel, AGR Reference Fuel, & German Fuel

Property	NFI HTTR	NFI Extended Burnup (HRB-22)	AGR Spec.	German
Kernel Diameter (µm)	600	544	350	508
U-235 Enrichment (%)	~ 3 – 9.9	4.1	19.8	10.6
Buffer Thickness (µm)	60	97	100	100
IPyC Thickness (µm)	30	33	40	39
SiC Thickness (µm)	25	34	35	35
OPyC Thickness (µm)	45	39	40	40
Buffer density (g/cm^3)	1.10	1.1	0.95	1.02
IPyC & OPyC density (g/cm^3)	1.85	1.85	1.90	1.91
SiC density (g/cm^3)	3.20	3.20	>3.19	3.20
Max. Burnup (% FIMA)	3.6	10	25	~10

GA's overall fuel acquisition strategy for the NGNP, including use of NFI fuel for the initial core fuel load, is discussed in Section 7.3

3.1.4 Design Modifications for Higher Temperature Operation

The GT-MHR was designed to operate with core inlet and core outlet helium temperatures of 490°C and 850°C, respectively. For the GT-MHR, the inlet coolant flow is routed through riser channel boxes between the core barrel and vessel as indicated in Figure 3-2. With this configuration, the design of the RV (including wall thickness and materials selection) is driven in large measure by the design point selected for the reactor inlet coolant temperature. The design point of 490°C ensures acceptable operating conditions for a RV manufactured from steels that do not experience creep damage at higher temperatures (e.g., 2¼Cr-1Mo or 9Cr-1Mo-V).

For the NGNP design, the core outlet helium temperature has been increased from 850°C to 950°C, in part to compensate for temperature drops through the IHX and maintain high thermal efficiency for hydrogen production and other process-heat applications. The coolant inlet temperature was also increased by 100°C to 590°C to provide a sufficiently high coolant flow and convective heat-transfer rate within the MHR core that ensures acceptable fuel performance and to limit release of Ag-110m and other noble-metal fission products that can diffuse through intact SiC coatings at high temperatures. However, this higher coolant inlet temperature could result in RV temperatures that exceed the limits for Cr-Mo steels if the current GT-MHR flow configuration were used. For this reason, one of the design modifications for NGNP is to route the inlet flow through holes in the permanent side reflector (PSR), which places additional thermal resistance between the inlet flow path and RV and lowers vessel temperatures. Thermal analyses show this design modification can reduce vessel temperatures by approximately 100°C.

Other design modifications that have been investigated as part of the preconceptual engineering studies include modifications to the reactor internal design to reduce bypass flow and modifications to the fuel-element design to enhance heat transfer. In addition, fuel shuffling strategies have been investigated that can reduce power peaking factors. These modifications can provide additional margin for fuel temperatures during normal operation, and may allow additional reduction of the coolant inlet temperature, such that SA-533/SA-508 steel (used for LWR RVs) could be used for the NGNP RV.

Bypass Flow Reduction. Fuel temperatures can be reduced by reducing bypass flow. Bypass flow is defined as any flow that bypasses the coolant holes of the fuel elements. Bypass flow channels include gaps between fuel columns and leakage between/from PSR blocks. Bypass flow can be reduced by using graphite sealing keys below the active core to provide additional flow resistance for bypass flow occurring between fuel columns. Lateral restraint devices and sealing tubes in the PSR riser channels can reduce the leakage flow between/from the PSR blocks.

FES has analyzed the flow distribution in the RV using a 3-D, 120°-sector ANSYS model. For the reference GT-MHR design, the bypass flow fraction is approximately 0.20. Routing the inlet flow through the PSR increases the bypass flow fraction to 0.37, primarily because of the relatively large lateral pressure gradients between the inlet flow path and reactor core. Adding sealing sleeves and lateral restraints reduces the bypass flow fraction to 0.14. Adding sealing keys at the bottom of the core further reduces the bypass flow fraction to 0.10. Reducing the bypass flow fraction from 0.20 to 0.10 reduces peak fuel temperatures by approximately 50°C.

Fuel-Element Modifications. The thermal performance of the graphite fuel element can be improved by reducing the temperature rise from the bulk coolant to the fuel compact centerline. This can be accomplished by reducing the diameters of the coolant holes and fuel compacts. This modified design is referred to as a 12-row block because the number of rows of fuel holes across the flats of the hexagonal block was increased from 10 to 12 (excluding boundary rows). For the 12-row block design, the minimum web thickness between the fuel and coolant holes was kept the same as the 10-row block for structural/strength considerations. As shown in Figure 3-4, the 12-row block design can reduce peak fuel temperatures by 30°C to 40°C, which can allow for reduction of the coolant inlet temperature. The higher flow resistance for the 12-row block is compensated for by the lower flow rate associated with a lower inlet temperature.

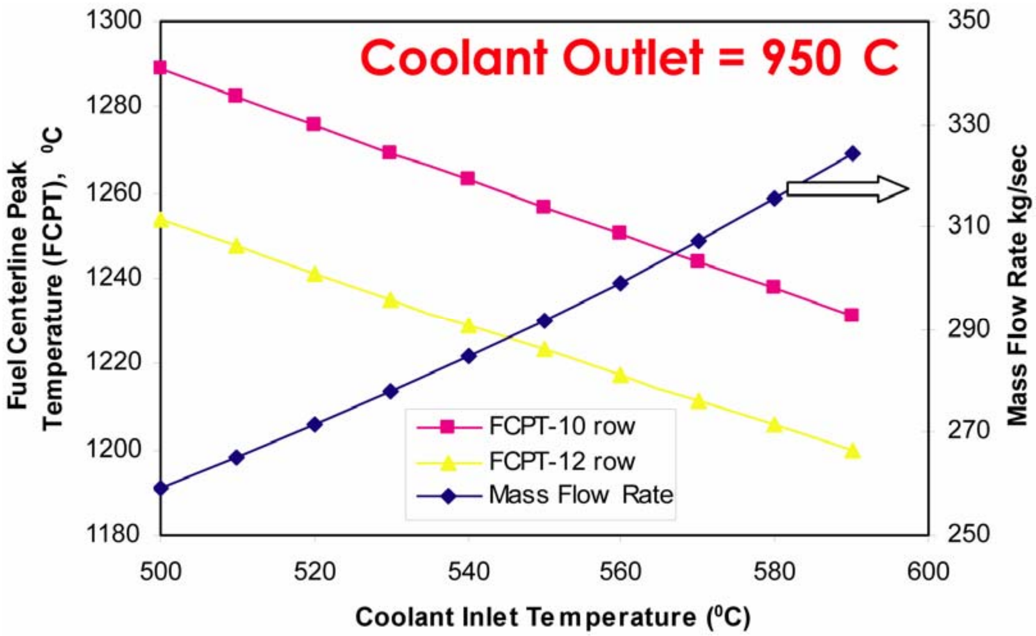


Figure 3-4. Comparison of 10-Row and 12-Row Block Thermal Performance

Fuel Management Strategies. As part of their work with GA on nuclear hydrogen development, KAERI has been investigating a block-shuffling refueling scheme for a 3-batch core. The KAERI concept uses 9 fuel elements (slightly longer than standard) per column to facilitate a 3-batch shuffling scheme, and adds 6 additional columns (108 fuel columns) to reduce the average power density by 5.6%. KAERI has performed 3-dimensional physics calculations to evaluate this concept, using 12% enriched fissile fuel only and zoning the particle packing fraction to reduce radial peaking factors. For these calculations, the bypass flow fraction was assumed to be 0.10 for each column. Figure 3-5 shows the calculated core temperature distributions for the 10-row and 12-row block designs with a coolant outlet temperature of 950°C and the coolant inlet temperature reduced to 490°C. Because of the relatively flat power and flow distributions, the calculated peak fuel temperature is below 1250°C, even with the reduced inlet temperature and coolant flow rate. Only about 20% to 30% of the fuel is predicted to be above 1000°C, which helps limit release of Ag-110m and other noble metallic fission products.

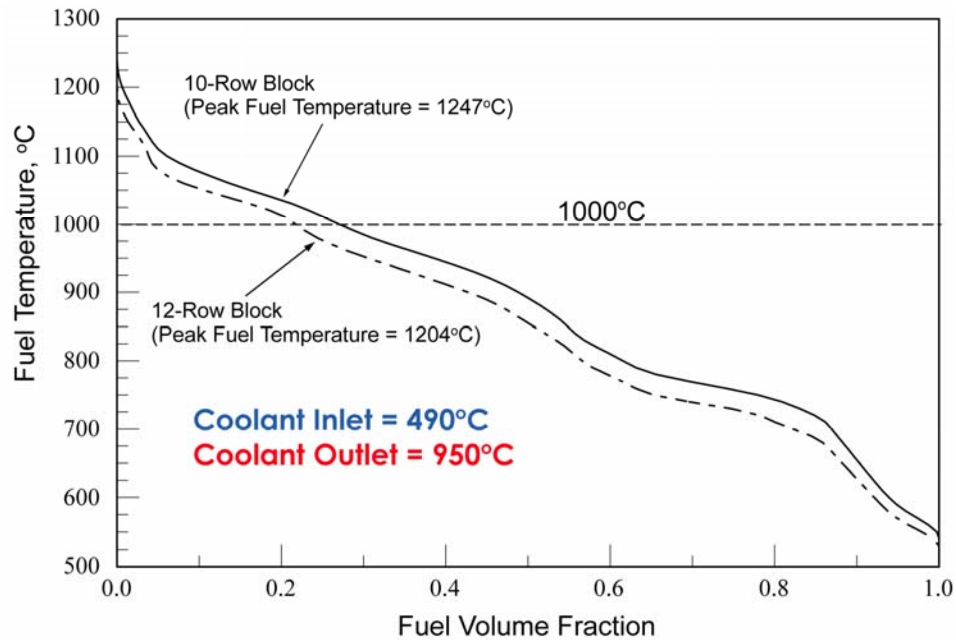


Figure 3-5. Calculated Fuel Temperature Distribution with KAERI Fuel Shuffling Scheme

Using this core configuration and power distributions, FES has performed normal-operation and safety assessments to determine under what conditions SA533 (SA508) steel could be used for the RV without using direct vessel cooling. Results are summarized in Table 3-3. The minimum bypass flow fraction was approximately 0.13 and was predicted to occur near the bottom of the core (where fuel temperatures are the highest). The peak fuel temperatures ranged from 1212°C to 1267°C for the four cases. Lowering the inlet temperature from 590°C to 490°C increased peak fuel temperatures by about 40°C, and raising the power level from 550 MWt to 600 MWt increased peak fuel temperatures by about 10°C.

Table 3-3. Peak Fuel Temperatures and Bypass Flow Fractions

Thermal Power (MW)	Inlet Temp. (°C)	Flow rate (kg/s)	Peak Fuel Temp. (°C)	Bypass Flow (%)		
				min.	max.	avg.
600	490	251	1267	13.2	21.5	18.2
600	590	321	1222	12.7	20.0	17.8
550	490	230	1257	13.1	21.3	18.1
550	590	294	1212	12.7	19.9	17.7

A 30-deg. sector ANSYS model was used to analyze both low-pressure conduction cooldown (LPCC) and high-pressure conduction cooldown (HPCC) events. In order to reduce vessel temperatures during these accidents, the reactor internal design was modified to include a 100-mm layer of carbon insulation on the outer radial boundary of the PSR. This carbon insulation was also assumed to contain B_4C to reduce neutron fluence to the RV. In the axial direction, the carbon insulation is applied over a length corresponding to the 3rd through 7th layers of fuel blocks.

A key parameter for these calculations is the graphite thermal conductivity, which decreases with damage caused by neutron irradiation. For these studies, calculations were performed using both irradiated and unirradiated graphite properties. Calculations were also performed assuming annealing of irradiation damage as the graphite temperature increases according to the GA model for H-451 graphite. Full recovery from irradiation damage is assumed to occur at temperatures greater than 1300°C. Other key parameters that affect heat transfer to the RCCS are the emissivities of the PSR, core barrel, RV, and RCCS panels. The PSR, RV outer face, and RCCS panels were assumed to have emissivities of 0.8. The core barrel inner and outer faces and inner face of the RV were assumed to have emissivities of 0.6. RCCS heat removal was assumed to occur by natural convection of air with an inlet temperature of 40°C and a stack height of 30 m. The decay heat rate was assumed to be 15% higher than the nominal rate in order to account for uncertainties in thermal properties, i.e., uncertainties in decay heat, thermal conductivities, emissivities, etc. were approximately accounted for by lumping them into a higher decay heat rate. Results for peak fuel and vessel temperatures for seven cases are summarized in Table 3-4. In addition to the different assumptions regarding graphite thermal conductivity discussed above, thermal power levels of 500 MW, 550 MW, and 600 MW were also analyzed. These cases all correspond to coolant inlet/outlet temperatures of 490°C/950°C.

Based on these results, the following conclusions are made:

- The reduction in graphite thermal conductivity with irradiation results in a peak fuel temperature increase of approximately 100°C. Accounting for thermal annealing of the irradiation damage reduces peak fuel temperatures by approximately 30°C. However, the effect of irradiation on graphite thermal conductivity has little impact on peak vessel temperatures.

- With the current reactor configuration and RCCS design, the reactor power level would have to be reduced to approximately 500 MWt in order for peak vessel temperatures to remain below the 538°C limit for SA533 (SA508) steel.

Table 3-4. Peak Fuel and Vessel Temperatures During HPCC and LPCC Events

Case	Thermal Power MW	Fuel Block Graphite Thermal Conductivity	Peak Fuel Temperature (°C)		Peak Vessel Temperature (°C)	
			HPCC	LPCC	HPCC	LPCC
1	600	Unirradiated	*	1595	*	582
2	550	Unirradiated	1416	1504	543	559
3	500	Unirradiated	1336	1415	521	536
4	550	Irradiated	1489	1602	539	561
5	500	Irradiated	1407	1516	515	537
6	550	Annealed	*	1570	*	564
7	500	Annealed	1398	1487	519	537
*Calculations not performed.						

Conclusions and Recommendations

1. Routing the inlet flow through the PSR appears to be a feasible concept that can significantly reduce vessel temperatures. Some additional trade studies should be performed to evaluate benefits/drawbacks of the OKBM design that uses both PSR and central reflector coolant risers.

2. Design modifications to reduce bypass flow include sealing sleeves in the PSR risers, lateral restraints, and sealing keys below the core. Independent analyses by FES and OKBM indicate that bypass flow can be reduced to about 10%. The reactor internals design should be developed in more detail and include these modifications.

3. The block-core design provides great flexibility to optimize power distributions using fuel shuffling schemes. Scoping studies show fuel shuffling can significantly reduce power peaking factors and flatten flow distributions. More detailed assessments of fuel shuffling should be performed, including coupled physics/thermal analyses and assessing the impact of control-rod movement.

4. An additional 30°C to 40°C margin for peak fuel temperatures can be obtained using a modified, 12-row block design, which could allow for further reduction in the coolant inlet temperature. More detailed assessments of this concept include manufacturability, structural/stress analyses, and impacts on fuel costs.

5. Direct vessel cooling using a 140°C slipstream flow of helium can reduce vessel temperatures to levels acceptable for use of SA-533/SA-508 steel for the RV. More detailed accident analyses are needed to further evaluate this concept, and the impacts on passive safety and investment risk should be evaluated in detail.

6. Additional optimization studies should be performed to determine if the NGNP reactor system can be designed to use SA-533/SA-508 steel for the RV without requiring direct vessel cooling. Preliminary results show coolant inlet temperatures of 490°C or lower would be required and enhanced RCCS heat removal would also be required for operation with reactor power levels above about 500 MWt.

3.1.5 Neutron Control System

The neutron control system design is the same as that for the GT-MHR. The system components consist of inner and outer neutron control assemblies, neutron source, source-range detector assemblies, ex-vessel neutron detector assemblies, and the in-core flux mapping system. Figure 3-2 above shows the locations of the neutron control assemblies and channels.

The neutron control assemblies are located in the top head of the RV. The structural equipment consists of an upper structural frame, gamma shielding, neutron shielding, thermal barrier, upper and lower guide tubes, and seals. The control rod guide tubes extend from the gamma shielding downward through the top head of the RV and upper plenum shroud to the upper core restraint elements. The guide tubes provide a clear passage for the control rods as they are inserted into and withdrawn from the core. All neutron control assemblies are equipped with two independent control rod drive units. The control rod drive equipment is located in the upper part of the neutron control assembly. The control rod is lowered and raised with a flexible high-nickel alloy cable.

The neutron absorber material consists of B_4C granules uniformly dispersed in a graphite matrix and formed into annular compacts. The compacts are enclosed in Incoloy 800H canisters for structural support. Alternatively, carbon-fiber reinforced carbon (C-C) composite canisters may be used for structural support. The control rod consists of a string of 18 canisters with sufficient mechanical flexibility to accommodate any postulated offset between elements, even during a seismic event.

The reserve shutdown control material is of the same composition as that for the control rods, except the B_4C granules and graphite matrix are formed into cylindrical pellets with rounded ends. The B_4C granules are coated with dense PyC to prevent oxidation during off-normal events. The pellets are stored in hoppers located above the reactor core in both the both the inner and outer neutron control assemblies.

During normal operation, the neutron flux levels are monitored by 6 symmetrically-spaced ex-vessel fission chamber thermal neutron detectors. The signals from these detectors interface with the automatic control and protection systems to operate the control rod drives or the reserve shutdown control equipment. Three fission chamber source-range detectors are used to monitor neutron flux during startup and shutdown. These detectors are symmetrically spaced in reentrant penetrations located in the bottom head of the RV. These penetrations extend into vertical channels in the reflector elements near the bottom of the core. The in-core flux mapping system consists of movable detectors in the central column of the inner reflector and in the outer permanent reflectors. The system enters from a housing located above the RV and vertically traverses down through the core to the bottom reflectors. The system contains two independent fission chambers and a single thermocouple.

3.1.6 Fuel Quality and Performance Requirements

As discussed in Section 3.1.2.2, optimization of the NGNP core nuclear and thermal hydraulic design should result in fuel service conditions that are not significantly different from those for the GT-MHR. As a result, the fuel quality and performance requirements for the NGNP are expected to be the same as for the GT-MHR. The expected service conditions, as-manufactured quality requirements, and in-service performance requirements for NGNP fuel are given in Tables 3-5 through 3-7. The requirements for in-service performance are specified on a core-average basis. The allowable release fractions for Cs-137 and Ag-110m are included in Table 3-7 because these nuclides are expected to be the strongest contributors to worker dose.

The Germans have manufactured high-quality, TRISO-coated fuel that have exhibited irradiation performance consistent with the expected fuel-performance requirements listed in Table 3-7 for the NGNP. The Japanese have achieved very high as-manufactured fuel quality and excellent irradiation performance with their low-enriched UO_2 fuel. The U.S. is developing UCO coated-particle fuel with similar requirements for as-manufactured quality and performance during normal operation and accident conditions [AGR Plan 2005].

Table 3-5. Service Conditions for Fissile and Fertile Fuel

	Fissile Fuel		Fertile Fuel	
Parameter	Peak	Core Average	Peak	Core Average
Fuel temperature (normal operation), °C	1250	[850] [(a)]	1250	[850]
Fuel temperature (accident conditions), °C	1600	—	1600	—
Fuel burnup, % FIMA	26	[15]	7	[4]
Fast fluence, 10^{25} n/m^2 (E > 0.18 MeV)	5	[3]	5	[3]
[(a)]Quantities in brackets indicate preliminary values.				

Table 3-6. As-Manufactured Quality Requirements for Fissile and Fertile Fuel

	Fissile Fuel		Fertile Fuel	
Parameter	Maximum Expected	Design	Maximum Expected	Design
Missing or defective buffer	1.0×10^{-5}	2.0×10^{-5}	[1.0×10^{-5}][(a)]	[2.0×10^{-5}]
Defective SiC	5.0×10^{-5}	1.0×10^{-4}	[5.0×10^{-5}]	[1.0×10^{-4}]
HM contamination	1.0×10^{-5}	2.0×10^{-5}	[1.0×10^{-5}]	[5.0×10^{-5}]
HM contamination outside intact SiC	6.0×10^{-5}	1.2×10^{-4}	[6.0×10^{-5}]	[1.2×10^{-4}]
[(a)]Quantities in brackets indicate preliminary values.				

Table 3-7. In-Service Performance Requirements for Fissile and Fertile Fuel

	Fissile Fuel		Fertile Fuel	
Parameter	Maximum Expected	Design	Maximum Expected	Design
Allowable fuel failure fraction (normal operation)	5.0×10^{-5}	2.0×10^{-4}	[5.0×10^{-5}][(a)]	[2.0×10^{-4}]
Allowable fuel failure fraction (accident conditions)	[1.5×10^{-4}]	[6.0×10^{-4}]	[1.5×10^{-4}]	[6.0×10^{-4}]
Allowable Cs-137 release fraction (normal operation)	1.0×10^{-5}	1.0×10^{-4}	[1.0×10^{-5}]	[1.0×10^{-4}]
Allowable Cs-137 release fraction (accident conditions)	1.0×10^{-4}	[1.0×10^{-3}]	[1.0×10^{-4}]	[1.0×10^{-3}]
Allowable Ag-110m release fraction (normal operation)	2.0×10^{-4}	2.0×10^{-3}	[2.0×10^{-4}]	[2.0×10^{-3}]
Allowable Ag-110m release fraction (accident conditions)	[2.0×10^{-3}]	[2.0×10^{-2}]	[2.0×10^{-3}]	[2.0×10^{-2}]
[(a)]Quantities in brackets indicate preliminary values.				

Two advanced coated particle designs are being considered to provide additional performance margins at higher temperatures. These particle designs incorporate ZrC either as a replacement for the SiC layer or as an oxygen getter within the particle. These particle designs are discussed in more detail in [Richards 2006a] and have been included as part of the development plan prepared by GA for advanced coated-particle fuel [Hanson 2004b].

3.2 Vessel System

The NGNP Vessel System includes the RV, the PCS vessel, the IHX vessel, and the cross vessels that connect these vessels. Because of the large size of the RV and PCS vessel, transportation of the vessels to the Idaho NGNP site may be problematic. Some on-site assembly of these vessels may therefore be necessary. A study is needed to assess the feasibility of vessel transportation to the NGNP site and on-site assembly operations.

3.2.1 Reactor Vessel

As shown in Figure 3-6, the RV is composed of a main cylindrical section with hemispherical upper and lower heads. In the GT-MHR design, the upper head is bolted to the cylindrical section. An alternative to bolting the upper head to the vessel is to use a weld joint since the head is not intended to be removed during the lifetime of the reactor. The upper head includes penetration housings for the neutron control assemblies and the in-vessel flux monitoring unit. These housings are sealed with a blind flange. The lower head is welded to the cylindrical section and includes penetrations for the SCS, in-service inspection access, and source-range neutron detectors. The upper portion of the lower head incorporates a ring forging that provides support to the core through the core support structure. The cylindrical section includes a nozzle forging for attachment of the cross vessel, RV support lugs, and lateral restraint keys. Lateral seismic restraint is provided to the core by lugs welded to the interior surface of the vessel, near the top of the cylindrical section.

The reference GT-MHR design selected 9Cr-1Mo-V steel for the RV. However, GA material specialists have recommended against using 9Cr-1Mo-V steel for the NGNP, primarily due to expected welding difficulties and lack of manufacturing and operating experience. Although the primary coolant temperature for the NGNP is higher than that for the GT-MHR, the results of core optimization studies indicate that RV temperatures can be maintained within limits that allow selection of a vessel material having temperature limits lower than 9Cr-1Mo-V steel. Accordingly, the material selected for the RV for the NGNP preconceptual design is 2¼Cr-1Mo steel. However, design alternatives are being considered that could potentially lower RV temperatures to a level that would allow use of proven LWR vessel materials (e.g., SA508/SA533 steel).

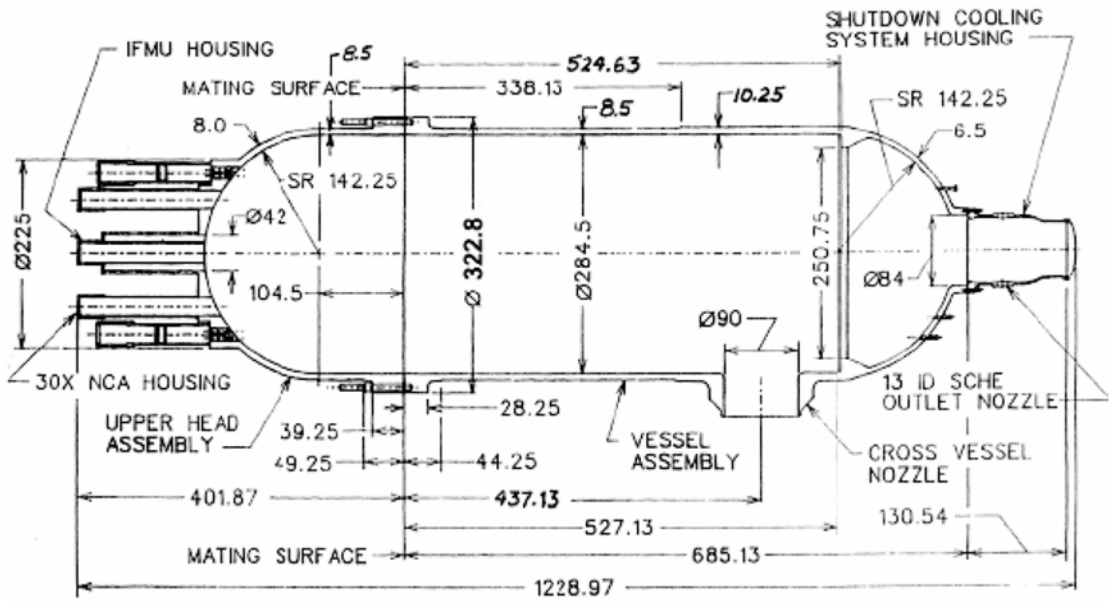


Figure 3-6. RV (dimensions are in inches)

GA has had discussions with two RV manufactures concerning NGNP vessel fabrication, specifically Japan Steel Works (JSW) and DOOSAN Heavy Industries and Construction (DOOSAN). The current maximum cylindrical forging size is limited to 8.2 m diameter. As an alternative approach to forgings, GA material experts suggest manufacturing the RV from rolled plate, or a combination of rolled plant and forgings. DOOSAN has provided GA with vessel manufacturing schemes for both design approaches.

3.2.2 Cross Vessel and Hot Duct Assembly

The NGNP design includes cross vessels that connect the RV to the PCS vessel and the RV to the IHX vessel. The hot ducts that provide the hot-leg primary coolant flow path from the RV to the PCS and IHX vessels are concentrically located within the cross vessels. The annular space between the hot duct and cross vessel provides the cold-leg primary coolant flow path from the PCS and IHX vessels to the RV. The hot duct assembly includes a ceramic fiber insulation layer to minimize heat transfer between the hot-leg and cold-leg flow paths. A similar insulation layer may also be included on the inside diameter of the cross vessel.

As shown in Figure 4-3 in Section 4, the IHX cross vessel does not have a constant diameter. The RV nozzle side is sized to accommodate a full-size (600 MWt) IHX. An area reducer is provided in the cross vessel to accommodate the 65 MWt printed-circuit type heat exchanger that is planned for initial installation in the NGNP.

The NGNP hot duct material will be a high-temperature alloy (e.g., Incoloy 800H, Hastelloy-XR, or Inconel 617). The material selected for the cross vessels for the NGNP preconceptual design is 2¼Cr-1Mo steel; however, it may be feasible to use LWR vessel materials for the cross vessels as well as for the RV.

3.2.3 Power Conversion System Vessel

The PCS vessel consists of a cylindrical shell with a lower head that is welded to the shell and an upper cylindrical closure head that is bolted to the lower vessel shell. The lower cylindrical shell contains the penetrations for the cooling water inlet and outlet of the precooler and intercooler modules. The upper closure head contains the turbomachine penetration. The outer diameter of the vessel (at flange) is 8.5 m. The material selected for the PCS vessel is SA508/SA533 (LWR steel).

3.2.4 IHX Vessel

The IHX vessel is a pressure boundary for the primary helium coolant and will be designed according to Section III of the ASME Code. The material selected for the IHX vessel for the NGNP preconceptual design is 2¼Cr-1Mo steel. The IHX vessel may include a ceramic fiber insulation layer, such as Kaowool, on inside surfaces to maintain operating temperatures within the material temperature limits. The vessel has an inner diameter of 3.81 m and is approximately 16 m in height.

3.2.5 Vessel System Support Arrangement

Vertical vessel support is provided at the same building elevation for the reactor, PCS, and IHX vessels. This feature minimizes differential vertical thermal expansion between connecting vessels at the cross vessel elevation, thus minimizing shear and bending moments on either cross vessel. The vertical support is provided through sliding pads which allow unrestrained thermal and pressure expansions of the vessel system in the horizontal plane, minimizing axial loads on the cross vessels. The vessel support design limits relative motions between the vessels and RB during a seismic event.

3.3 Fuel Handling System

The Fuel Handling System for the NGNP reactor will be similar to that designed for the GT-MHR. The GT-MHR system includes a fuel handling machine, two fuel transfer casks, an auxiliary transfer cask, a fuel handling equipment positioner, a fuel handling equipment support structure, and local spent fuel storage and handling facilities. Two or three large, portable, isolation gate valves are also included in the fuel handling system equipment inventory. These valves are placed over the RV refueling penetrations, spent fuel storage wells, or spent fuel sealing and inspection facility whenever elements are moved in or out of these locations. All operations and movements of the machines and the associated fuel and reflector elements are automatically monitored and recorded to maintain full accountability. Each fuel and reflector element is uniquely identified as necessary to support this accountability requirement. The fuel sealing and inspection facility is included in the system to provide for receipt and inspection of new fuel, and for packaging of spent fuel that is to be transported for storage or disposition either within the plant area or off-site.

The refueling procedure for the NGNP will also be essentially the same as that developed for the GT-MHR. Refueling takes place on a specific schedule, and involves the entire 1020 fuel element inventory in the reactor core, plus certain replaceable reflector elements as may be required. The arrangement of fuel handling equipment is shown in Figure 3-7. A routine refueling commences with depressurization of the vessel system and installation of the fuel handling support structure above the RV. This support structure is moved and handled using the fuel handling equipment positioner. Using the auxiliary service cask, the nuclear instrumentation equipment is removed from the RV centerline penetration. A fuel element guide sleeve and support plate assembly is then inserted into this penetration, also using the auxiliary service cask. Under controlled conditions, a neutron control assembly (control rod drive) is removed from one of the vessel top head inner penetrations using the auxiliary service cask. Using the fuel handling equipment positioner, the fuel handling machine is installed over that same penetration. Also using the fuel handling equipment positioner, a fuel transfer cask is mounted over the RV centerline penetration, immediately adjacent to the fuel handling machine. Both machines are anchored to the support structure to assure seismic integrity.

Fuel and reflector elements are removed from the reactor in a specific order and placed in the fuel transfer cask, one by one. When full, this cask is moved to the spent fuel storage area where all or a portion of these elements are placed in a helium-filled spent fuel storage well for interim cooling. New fuel elements are then loaded into the fuel transfer cask and moved to the reactor where they are placed into the core, also in a specific order.

Replacement of certain fuel and reflector elements near the outer edges of the core requires that the control rods (and guide tubes) and reserve shutdown guide tubes associated with the neutron control assemblies in the outer penetrations to be withdrawn to allow access into this area by the fuel handling machine. All such control rod withdrawals must be fully approved prior to withdrawal, and carefully controlled and monitored during the actual withdrawal. When all element moves are completed in this outer area, the rods and guide tubes are re-inserted.

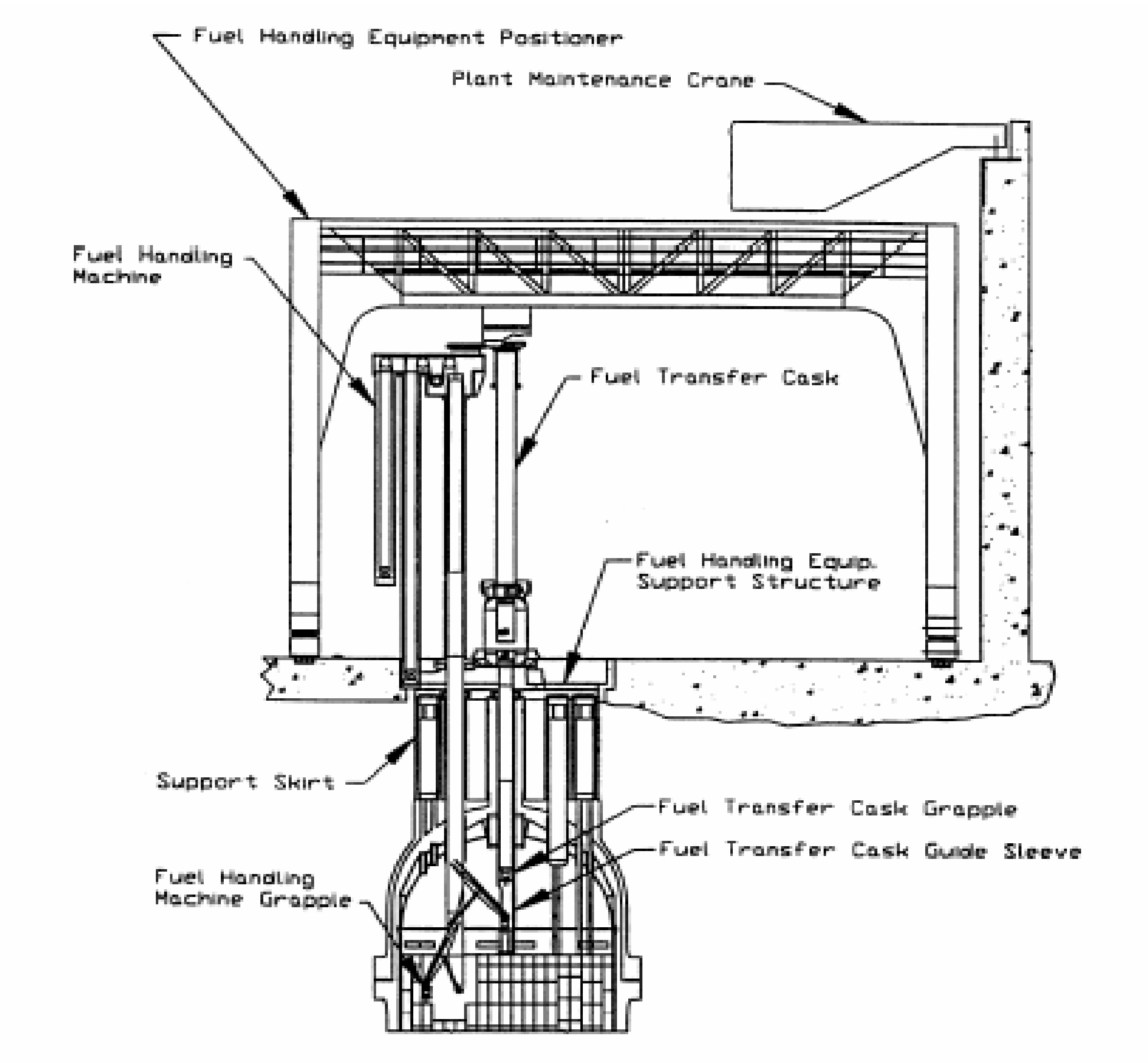


Figure 3-7. GT-MHR Fuel Handling Equipment

When the current one-sixth region of the reactor has been refueled, the fuel handling machine is removed and the neutron control assembly is replaced in that penetration. The fuel handling support structure is then rotated as needed to obtain access to the next inner top head penetration for continuation of the refueling procedure. This process continues until the entire reactor core has been refueled in accordance with a predetermined sequence of fuel and reflector movements, after which the fuel handling support structure is removed and the reactor is recovered and prepared for resumption of operations.

3.4 Shutdown Cooling System

The NGNP design has three diverse active heat-removal systems, the PCS, the HTS, and the SCS, all of which can be used for removal of decay heat from the reactor. The SCS is designed to provide decay heat removal when the primary heat removal systems are off line. The SCS consists of a circulator with shutoff valve, a heat exchanger, a control system, a shutdown cooling water system, and equipment for servicing the circulator and heat exchanger.

The SCS consists of a single loop with the heat exchanger in series with the circulator and loop shutoff valve assembly. These components are located at the bottom of the RV. Hot helium from the core outlet plenum flows through multiple parallel openings (pipes) in the center of the core support structure and into the heat exchanger. Once cooled, the helium continues downward through the loop shutoff valve to the circulator where it is compressed and discharged into the RV bottom head cavity. The cool helium then flows through the internal passage formed by the core support structure, up through the flow channels in the PSR, and into the core inlet plenum. The loop is completed as the helium flows down through the reactor core. The heat is transferred to a cooling water system that rejects the heat to the atmosphere through an air-cooled heat exchanger.

Because of the pressure drop associated with the IHX and PCS, there will be some back flow of helium through the IHX and PCS vessels. This backflow is factored into the SCS design in order to prevent local flow reversals and ensure adequate core cooling.

The SCS is sized to remove decay heat under both pressurized and depressurized conditions. Under pressurized conditions the SCS is sized to remove up to 40 MWt. When the reactor system is shutdown and depressurized for maintenance or refueling, the SCS is sized to remove up to 14.1 MWt. To ensure high reliability, the SCS can draw electrical power from either normal or standby systems.

During normal operation of the reactor system, the SCS operates in a standby mode. During this mode, a small amount of cold leg helium leaks (back flows) through the closed shutdown valve and flows opposite the normal flow direction through the SCS circulator and over the SCS heat exchanger tubes. In this mode the circulator is not operating, but the SCS cooling water system supplies a small amount of water flow to the heat exchanger. This water flow prevents thermal shock when the SCS switches to an active cooling mode, but also results in a parasitic heat loss of up to 1.3 MWt during normal operation. During standby mode, the primary coolant helium pressure is higher than the SCS water pressure to prevent water ingress into the reactor system during normal operation. The SCS is manually switched from standby mode to an active cooling mode at the discretion of an operator.

3.5 Reactor Cavity Cooling System

The Reactor Cavity Cooling System (RCCS) is a safety-related system that provides a passive means of removing core decay heat when the PCS, the HTS, and the SCS are unavailable for decay heat removal. Shown in Figure 3-8, the RCCS is a completely passive design that has no pumps, circulators, valves, or other active components. The RCCS receives heat transferred from the RV by thermal radiation and natural convection. RCCS components include cooling panels that surround the RV (as shown in Figure 4-3 in Section 4), inlet/outlet structures that are located above grade, and a concentric duct system with the annular, outer flow path acting as the cold leg and the inner flow path acting as the hot leg. Through a balance of buoyancy and gravitational forces, natural convection airflow is established through the RCCS circuit.

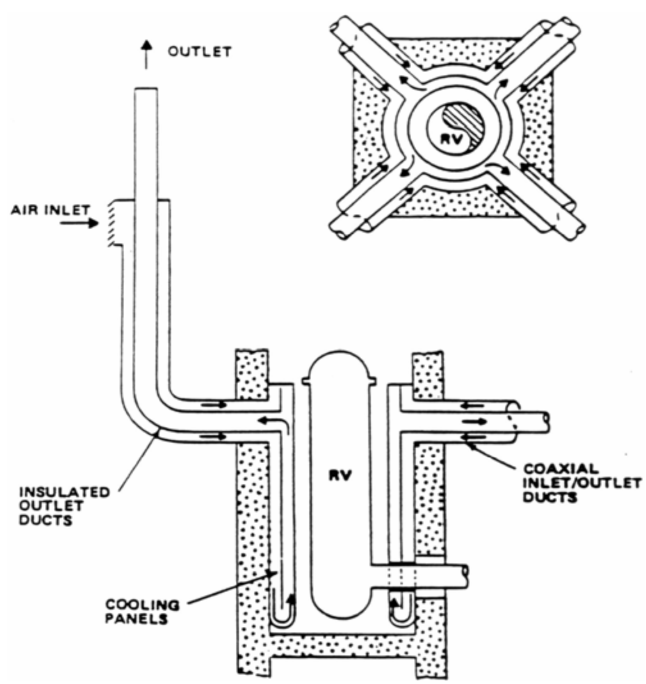


Figure 3-8. Passive Air-Cooled RCCS

For passive removal of decay heat, the core power density and annular core configuration have been designed such that the core decay heat can be removed by conduction to the pressure vessel and transferred by radiation from the vessel to the natural circulation RCCS without exceeding the fuel particle temperature limit. The RCCS has multiple inlet/outlet ports and interconnected parallel flow paths to ensure cooling in the event of blockage of any single duct or opening, and is robustly designed to survive all credible accidents scenarios. However, even if the RCS is assumed to fail, passive heat conduction from the core, thermal radiation from the vessel, and conduction into the silo walls and surrounding earth are sufficient to maintain peak fuel temperatures below the 1600°C design limit.

The RCCS is designed to passively remove ~4 MWt when the primary cooling circuit is either pressurized or depressurized. During normal operation, the RCCS provides cooling to the reactor cavity concrete structure. Also, during normal power operation, there is some parasitic heat loss to the RCCS.

3.6 Power Conversion System

A vertical integrated PCS design was selected for the GT-MHR from trade studies performed as part of the GT-MHR preconceptual design developed under a joint initiative of the DOE and U.S. Utilities over the period 1991 - 1994. The PCS design concept was developed by GA, General Electric, and Allied Signal. In 1994, the GT-MHR was selected as the basis for a joint effort by the U.S. and Russia to design a MHR to be used for disposition of w-Pu. OKBM was given responsibility for the GT-MHR design development. Starting with the U.S. GT-MHR PCS design, OKBM has further developed the PCS through preliminary design and has made several design improvements. Figure 3-9 illustrates the vertical integrated PCS design.

The PCS consists of four major components: a turbomachine (TM), a recuperator, a precooler and intercooler, and the in-vessel metalwork (IVM). The TM speed is 4400 rpm; a frequency converter is used to connect the generator with the outside grid with standard current frequency of 60 Hz. The attractive features of this PCD design include: (1) a direct Brayton cycle that provides high efficiency and superior economics, (2) a vertical shaft that minimizes blade/stator clearances to reduce bypass flows, reduces plant footprint and associated capital costs, allows vertical lifts for maintenance, and the use of gravity to offset turbine thrust, (3) electromagnetic bearings (EMBs) that reduce energy losses and eliminate the possibility of lubricant ingress into the primary circuit, (4) a single stage of intercooling that improves thermal efficiency by about 2% over a non-intercooled cycle, and (5) a submerged generator that eliminates a rotating seal in the primary pressure boundary and reduces leakage of primary helium coolant.

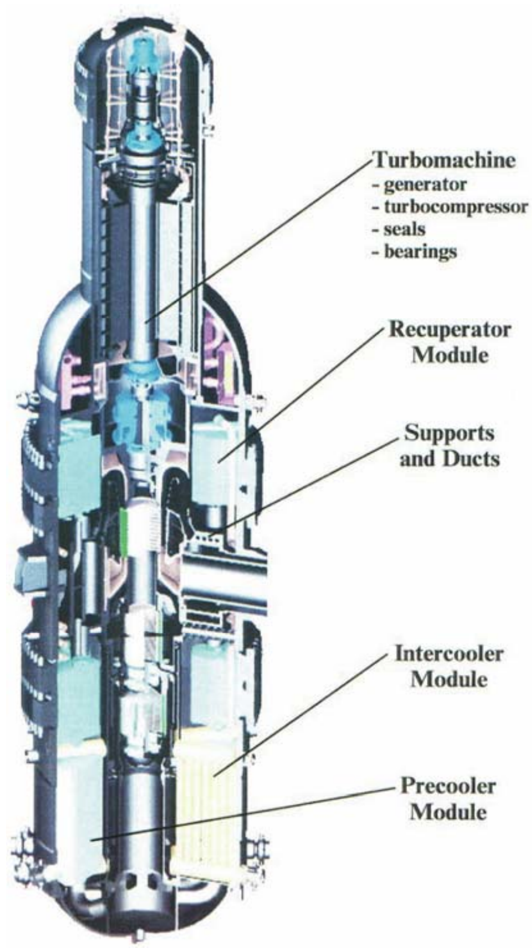


Figure 3-9. GT-MHR Power Conversion System Design Concept

Figure 3-10 shows how the helium circulates within the PCS. High-pressure helium from the reactor core outlet plenum flows through the hot duct inside the PCS cross vessel to the turbine where it expands. The mechanical energy generated in the turbine is used to drive the generator, the low-pressure compressor (LPC), and the high-pressure compressor (HPC), which are arranged on a common shaft. Downstream of the turbine, the helium flows through the low-pressure side of a recuperator where heat is transferred to the helium flowing back to the reactor through the high-pressure side of the recuperator. Upon exiting the low-pressure side of the recuperator, the helium passes through a precooler, where it is cooled to about 25°C, before passing through the LPC. Following the LPC, the helium passes through an intercooler where it is again cooled to about 25°C before entering the HPC. After exiting the HPC, the helium flows through the recuperator high-pressure side, where it is heated to the reactor inlet temperature and flows back to the reactor through the annular gap between the PCS cross vessel and hot duct.

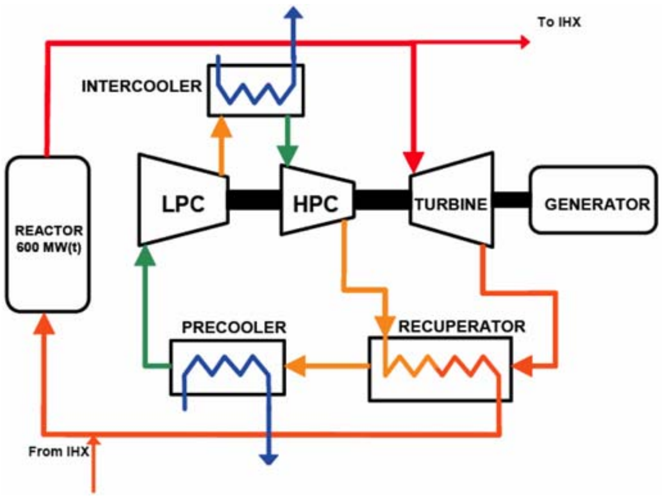


Figure 3-10. PCS Flow Diagram

The PCS operating conditions depend upon its operating modes, which include plant startup and shutdown, electric-only mode at any power level within the required NGNP operating range, and cogeneration mode during which ~90% of the reactor outlet helium flow is transferred from the reactor to the PCS. When the reactor power level is changed, helium is supplied or removed from the primary circuit to increase (or decrease) the mass flow rate while maintaining constant volumetric flow rate into the turbine and the same efficiency of thermal-to-electric power conversion. If a quick power decrease is needed, the TM bypass control valve is used to bypass helium flow from the HPC outlet to the turbine outlet.

3.6.1.1 Turbomachine Design

Turbocompressor (TC). The TC consists of the turbine, the LPC, the HPC, and the TC electromagnetic bearing (EMB) support system. Figure 3-11 illustrates the design of the TC. The turbine and compressor stators constitute a single load-bearing structure, which protects other PCS components and the PCS vessel against TC breakdowns (de-blading, etc.). The turbine and compressors are multistage and axial. The TC design provides rotor seals at the upper end (buffer and repair seals) and sliding seals for the TC stators. The buffer and repair seals are designed to prevent helium egress of the primary helium coolant into the PCS vessel upper section that contains the generator. Procedures and equipment designs have been developed for TC replacement if the design lifetime has been reached or in case of failure.

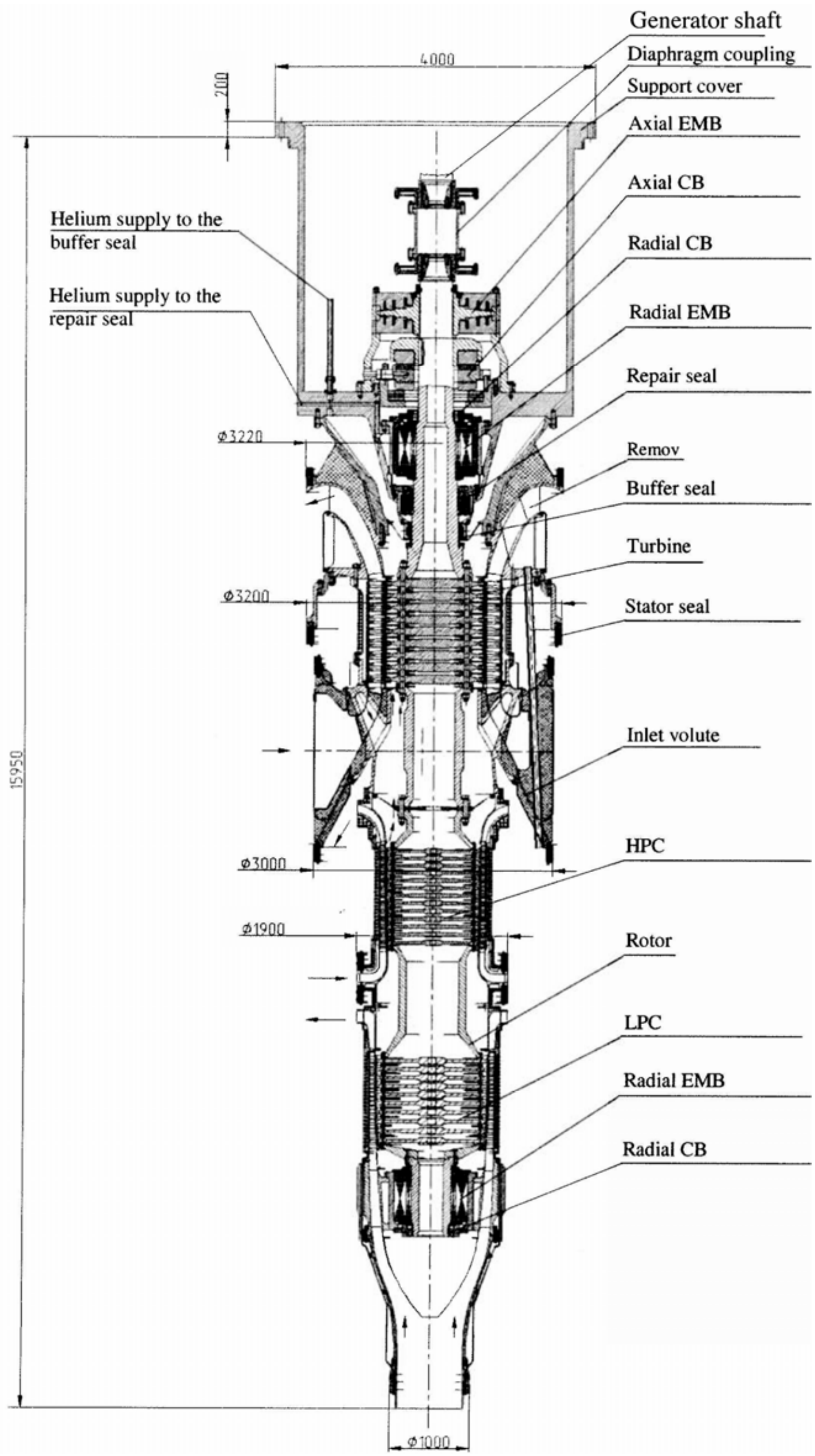


Figure 3-11. Turbocompressor (dimensions shown in millimeters)

<u>Generator</u>. The generator is a vertical rotating asynchronous machine consisting of a main generator, exciter, and a generator support system. The generator stator is enclosed in a casing, which is a load bearing structure for the rotor and the TC stator. The generator stator and rotor, exciter, and electric terminals are cooled with helium, which is circulated in a closed circuit by a fan installed on the generator rotor; heat is removed via gas coolers using a water cooling system. Helium flow from the TC cavity to the generator cavity is controlled by the buffer seal located between the two cavities. A higher pressure is maintained in the generator cavity than in the TC cavity. A supply of clean helium is provided to the helium cavity in the seal to prevent flow from the TC cavity to the generator cavity.

<u>Electromagnetic Bearings Support System Design</u>. The TC rotor and generator are connected by a flexible diaphragm coupling, which allows for separation of the TM EMB support system into two systems: one system for the TC rotor and another system for the generator rotor. The TC and generator EMB support systems consist of two radial and one axial EMB, catcher bearings for each of these EMBs, and an EMB control system (EMB CS). The CBs, which take the rotor load during TM mounting/dismounting and long-term outage, EMB failure, and external impacts exceeding EMB load-bearing capacity, are provided as a part of the EMB support system.

3.6.1.2 Recuperator Design

The recuperator is a gas-to-gas modular heat exchanger. It consists of twenty vertical modules, half arranged above and half below the hot gas duct. Each recuperator module contains approximately 200 individual heat transfer elements that are based upon a plate-type heat exchange surface design. The recuperator layout and design is shown in Figures 3-12 and 3-13, respectively. The recuperator is designed for the life of the plant, but removal and replacement of recuperator modules would be necessary if the modules started to leak. Replacement of recuperator modules, if necessary, would be accomplished through the use of remotely controlled devices because the recuperator modules are expected to be highly radioactive.

3.6.1.3 Precooler, Intercooler, and Generator Gas Cooler Design

The precooler, intercooler and generator gas cooler are shell-and-tube heat exchangers with helium on the shell side and water on the tube side. Reduced bending radius coils were selected as heat exchange surface.

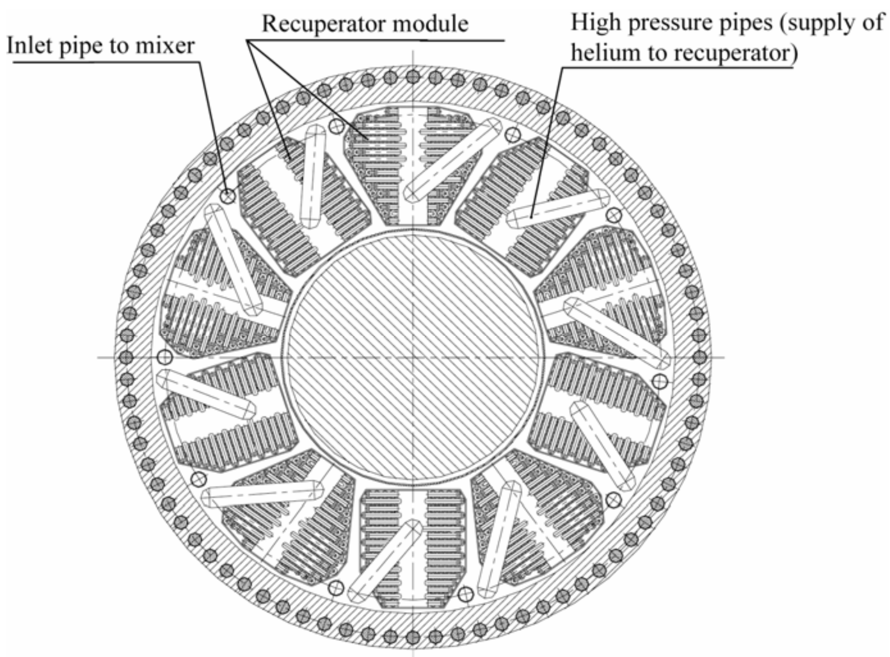


Figure 3-12. Recuperator Layout

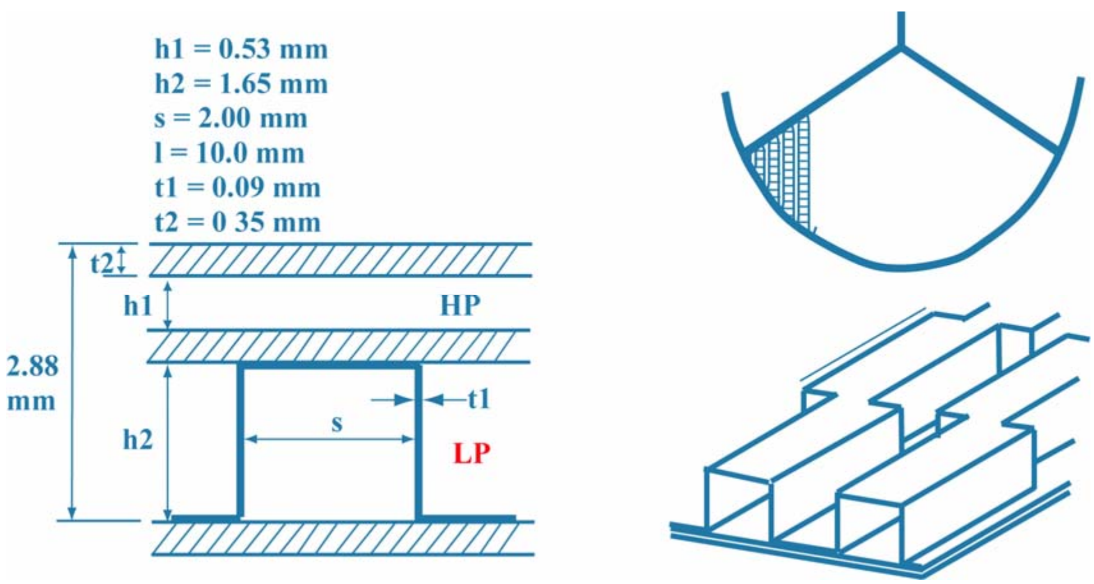


Figure 3-13. Recuperator Design (based on plate-type heat exchanger)

3.6.1.4 Power Conversion Unit In-Vessel Metalwork (IVM) Design

The PCS IVM forms the PCS helium circulation path, limits inter-circuit leaks, limits heat exchange between helium flows with different temperatures, and ensures components are fastened to the PCS vessel. The PCS IVM consists of component supports, gas ducts, thermal expansion compensators, and a mixer. The mixer is designed to mix helium at the turbine exhaust with helium from the HPC outlet when a portion of the helium flow is bypassed in power control modes and when protecting the turbine against acceleration.

3.6.2 Rolls-Royce Assessment of GT-MHR PCS Design

As part of the NGNP preconceptual design engineering studies, Rolls-Royce was tasked to perform a technical assessment of the OKBM design described in Section 3.6.1. The key results of the assessment are summarized below. [Rolls-Royce 2007] provides a detailed discussion concerning the scope, methodology, and results of the technical assessment.

- More work is needed to resolve key areas of uncertainty with respect to off-design and transient performance. The performance of the cycle under fault and accident conditions will be very important for the preparation of safety cases for the nuclear plant. Of particular interest is the loss of grid event and the management of it using bypass flow.

- The recuperator design is of particular concern. The OKBM design should work as intended, but it is expected to be quite expensive and difficult to manufacture given that the design requires an estimated 50 km of welds in the heat transfer elements alone. To mitigate the risk associated with the recuperator, an alternate cross-corrugated design (shown in Figure 3-14) that would be more compact, much lighter in weight, and likely less expensive is recommended.

- Although the temperatures experienced in the recuperator are not too challenging (and don't require special high temperature capable materials), the pressure differences between the two sides are relatively large. Making a delicate structure to survive a 60-year lifetime in this environment is very challenging. Consequently, both the OKBM reference design and the alternative cross-corrugated design should be considered to have a moderately high risk of not achieving the required life.

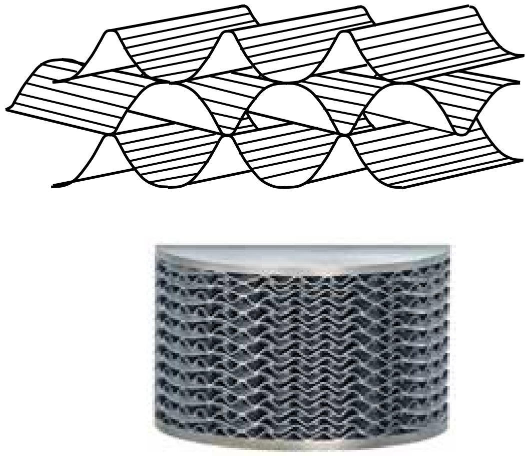


Figure 3-14. Cross-Corrugated Heat Exchanger

During the PCS study, it became apparent that certain areas require further study. Some of the more significant are:

1) Transient performance. The start-up and transient behavior of the PCS needs to be better understood so that the transient requirements for the components can be properly assessed. This will require a transient performance model to be constructed for the cycles and is a significant undertaking.
2) A study of the control system for the PCS needs to be made. This system's behavior is intimately bound up with the PCS's transient requirements.
3) Further refinement of turbine designs to increase confidence of achieving 60,000 hours creep life with uncooled turbine blades at 850°C.
4) Further exploration of the implications of 950°C operation, particularly cost, complexity, and performance trade-offs (including blade cooling and thermal barrier coatings).
5) A more thorough investigation into EMB capabilities and alternative technologies.

These studies should be performed during the next design phase.

3.6.3 Rolls-Royce Evaluation of Alternate PCS Designs

The vertical integrated PCS concept has long been recognized to pose several technical challenges with respect to individual equipment design and equipment arrangement within a single PCS vessel. Given these challenges, PCS design development was carefully monitored by the U.S./Russian International GT-MHR Program through a series of design reviews, both by internal experts and by independent third party experts. The results of these technical reviews have been thoroughly reviewed and evaluated to identify the uncertainties and assumptions (i.e., technical issues) in the science or engineering on which the design is based. (In general, the key risks identified by Rolls-Royce are consistent with the results of these earlier reviews.) A series of design data needs were prepared to define the data needed to resolve these uncertainties and assumptions, technology development plans were prepared for the various system components, and a comprehensive PCS technology demonstration plan was prepared to describe the overall technology development and demonstration program.

OKBM, in collaboration with GA and ORNL, is currently conducting this comprehensive technology development and demonstration program under the U.S./Russian International GT-MHR Program to qualify the OKBM PCS design, which GA has identified to be the reference design for the NGNP at this time. GA believes that this PCS technology demonstration program has a high probability of establishing the viability of the design before the end of NGNP preliminary design. Nevertheless, the GA Team believes that it would be prudent to develop an alternate backup PCS design to mitigate the risk associated with development and demonstration of the OKBM design.

Accordingly, Rolls-Royce was also tasked as part of the NGNP preconceptual design studies to explore options for, and recommend, a potential alternate PCS design for the NGNP. The results of this evaluation are summarized below. [Rolls-Royce 2007] provides detailed information concerning the scope and results of the evaluation.

Several alternate PCS concepts were evaluated. The most promising of the alternative concepts appears to be a direct combined cycle consisting of a 66 MWt gas turbine generator with the remainder of the thermal power taken by a conventional steam cycle (Figure 3-15). This cycle was worked up to a pre-concept level to understand feasibility and to make comparisons with the reference cycle. The conclusion of this work is that the combined cycle option looks feasible and may be slightly more efficient than the reference cycle. Its costs should be similar to that of the reference design and the concept mitigates some of the key risks identified with the reference design. The key features of this concept are:

- The recuperator is no longer required. A steam generator would be required, but this is considered much lower risk.
- EMB risks are reduced by reducing generator weight from 35 tons to around 10 tons, and the TC shaft weight from 32 tons to around 10 tons.
- Frequency converter Power electronics costs are reduced (since the generator is reduced from ~300 MW to ~66 MW in the gas turbine part).

- Plant efficiency is increased, compared with the GT-MHR Brayton cycle.
- Steam turbines and steam cycle electrical generators are commercial off-the-shelf items - low cost and low risk.

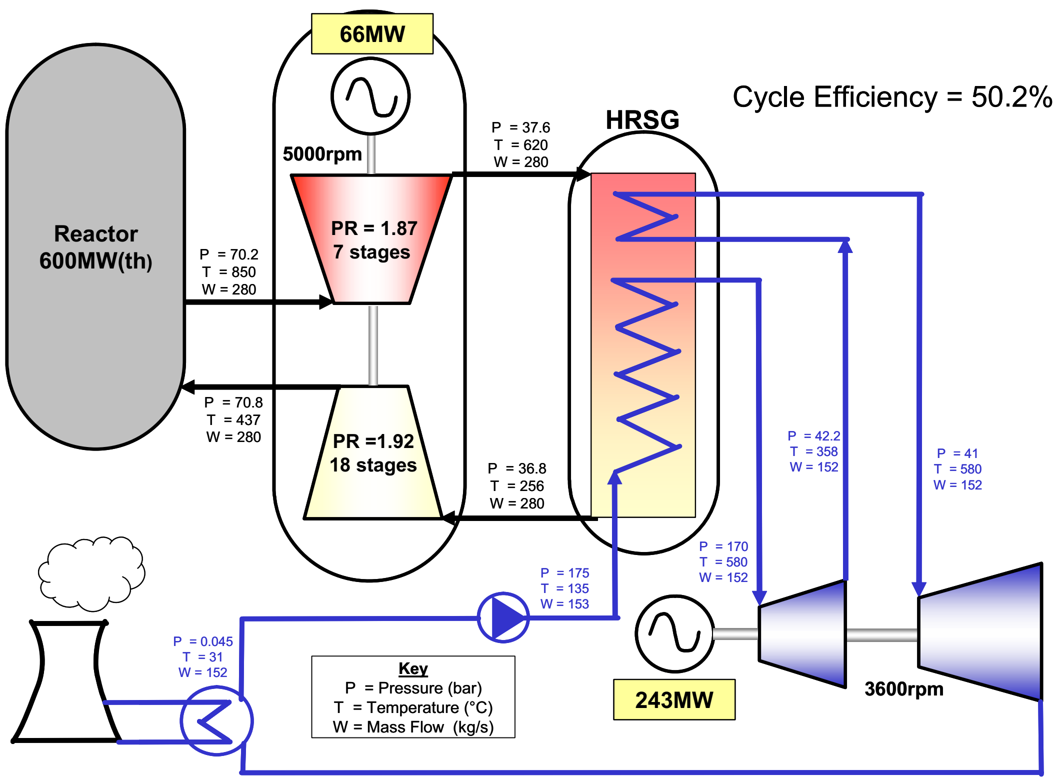


Figure 3-15. Combined Cycle Alternative Proposed by Rolls-Royce

The combined cycle alternative could have lower plant costs because much of the steam machinery is commercial off the shelf, but this saving would be offset by requiring a bigger RB and the extra maintenance burden of the steam cycle equipment.

Figure 3-16 shows the proposed layout for the combined cycle includes two separate pressure vessels, one containing the TC and generator and another containing the steam generator. The rest of the plant is considered to be low risk commercial off-the-shelf equipment and as such was not analyzed in any detail. In fact, the bottoming steam cycle is very similar to the steam cycles employed in U.K. nuclear plants, in particular the Advanced Gas Reactor (AGR) stations. The similarities to the AGR plant lend credibility to the choice of a combined cycle because of the generally successful performance of the AGR plant over the last 30 years.

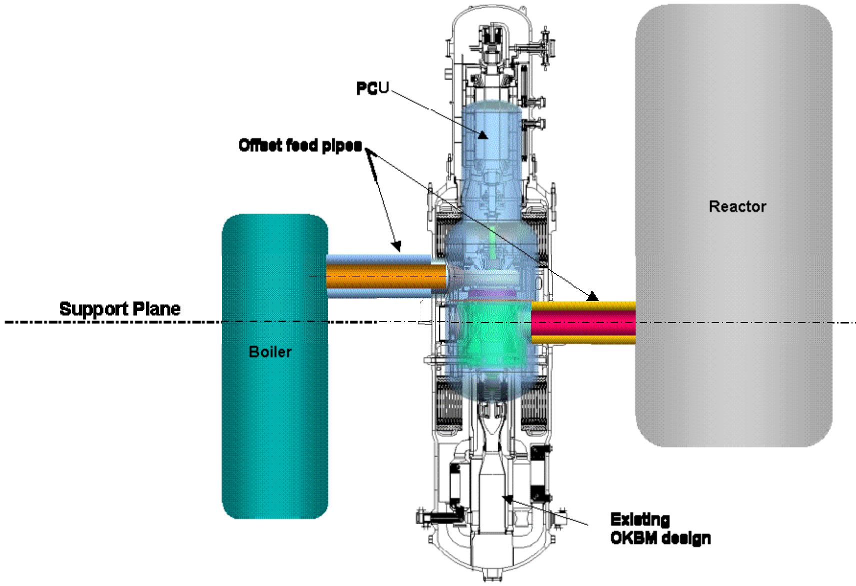


Figure 3-16. Layout of Combined Cycle Alternative in 3 Pressure Vessels

Advantages of the combined cycle include:

- Reduced EMB risk.
  - Generator from ~35 tons to ~12 tons.
  - TC from ~32 tons to 12 tons (due to shorter shaft).
- Elimination of recuperator risk.
- Commercial, off-the-shelf steam equipment (excluding steam generator). Total equipment costs should be lower.
- Flexibility to provide process steam instead of electricity from steam plant

Disadvantages of the combined cycle are:

- Increased equipment footprint, both inside RB and outside
- Increased complexity (but mainly commercial off-the-shelf steam plant)

For the combined cycle option to progress further, further study is needed with respect to the following design issues:

- Because there is no intercooler the number of times helium must be taken into and out of the TC is reduced. It may be possible to reduce costs and simplify the sealing arrangement between turbine inlet and the hot gas duct.

- Any helium leaking from the high pressure section of the gas turbine or exhaust will combine with flow entering the compressor and could cause hot streaks. It may be beneficial to move the economizer section of the boiler close to the GT inlet to mitigate this issue.

- The surfaces of gas turbine and steam generator outer casing and the inner surface of the pressure boundary will need to be insulated. The integrity and reliability of the fixing system needs to be demonstrated due to the potential to damage the compressor caused by debris ingestion.

- Power leads and instrumentation connections to the TC could be taken out to another interconnection at the bottom of the pressure vessel easing the problem of running such connections through the high pressure/temperature section. However this may lead to issues where such leads pass through the pressure vessel.

- The ambient temperature of the EMBs may become an issue if suitable thermal insulation and cooling strategy cannot be adequately demonstrated. The plate out of silver could also cause reliability issues for the EMB coils.

3.6.4 PCS Cooling Water System

This system is a closed loop piping arrangement that absorbs heat from the pre-cooler and intercooler heat exchangers in the PCS. This system also provides cooling water to the various coolers within the main plant electricity generating system, such as the generator cavity coolers, the stator windings, and the magnetic bearing system. The absorbed heat is rejected through a series of heat exchangers in the auxiliary building located outside the RB.

3.6.5 Power Generation Facility

Three-phase electric power delivered by the main facility generator is routed to the main power transformer for voltage upgrading (e.g. to 240 kV) as required for compatibility with the off-site power distribution and transmission system (the grid). A unit auxiliary transformer is connected to the high-voltage side of the main power transformer to supply power to the facility at reduced voltages (typically as 4.16 kV input to the various in-plant system transformers). This arrangement allows either the main facility generator or the outside power transmission system to provide house power to the plant. A reserve auxiliary transformer may also be included in the overall power generation and distribution system as may be needed if the unit auxiliary transformer is not available. The reserve auxiliary transformer takes power directly from the grid and feeds directly into the plant electrical system. A number of circuit breakers are included in the power distribution system to control the flow of power to and from the various sources and users.

3.7 Heat Transport System

The Heat Transport System (HTS) includes the systems, subsystems and components necessary to transport 65 MWt of high-temperature heat from the primary system of the NGNP to the process heat exchangers in the hydrogen production plants. It consists of both the Primary HTS and the Secondary HTS.

3.7.1 Primary Heat Transport System

The Primary HTS consists of the IHX and primary helium circulator (PHC). Both of these components, along with associated internal ductwork, are contained within the IHX vessel. The IHX vessel is connected to the RV by the IHX cross vessel and associated IHX hot duct. The Primary HTS diverts a portion of the primary coolant flow from the hot plenum of the reactor and sends it through the IHX in order to transfer the heat to the Secondary HTS. The Primary HTS helium circulator located on top of the IHX vessel returns the diverted primary coolant to the RV where it rejoins with the helium returning from the PCS on its way back to the inlet plenum at the top of the reactor core.

3.7.1.1 Intermediate Heat Exchanger

Two alternate IHX designs were developed based on the printed circuit heat exchanger (PCHE) concept developed by Heatric Corporation (Figure 3-17). This design consists of metal plates that are diffusion bonded to restore the properties of the base metal. Fluid-flow channels are chemically milled into the plates using a technique that is similar to that used for etching printed electrical circuits. The PCHE concept allows for simultaneous high-temperature and high-pressure operation with relatively thin wall thicknesses between the primary and secondary coolants. PCHEs are typically four to six times smaller than conventional shell-and-tube heat exchangers of equivalent heat duty, and designs have been developed with thermal effectiveness greater than 98%.

In the first PCHE-type IHX design, the arrangement of the flow paths through the PCHE modules is slightly different from that depicted in Figure 3-17. In order to bathe the exterior of the IHX in cold primary coolant helium, the flow arrangement depicted in Figure 3-18 was devised. The cross-sectional view shows the primary flow path on the lower half and the secondary flow path on the upper half which would represent the PCHE plate just above or below the plate used by the primary flow. Primary and secondary PCHE plates are stacked one on top of the other to form each PCHE module. The primary coolant inlet and secondary coolant inlets and exits would be at the top of the IHX. The primary coolant exits would be all along the north and south sides of the IHX as depicted in Figure 3-18.

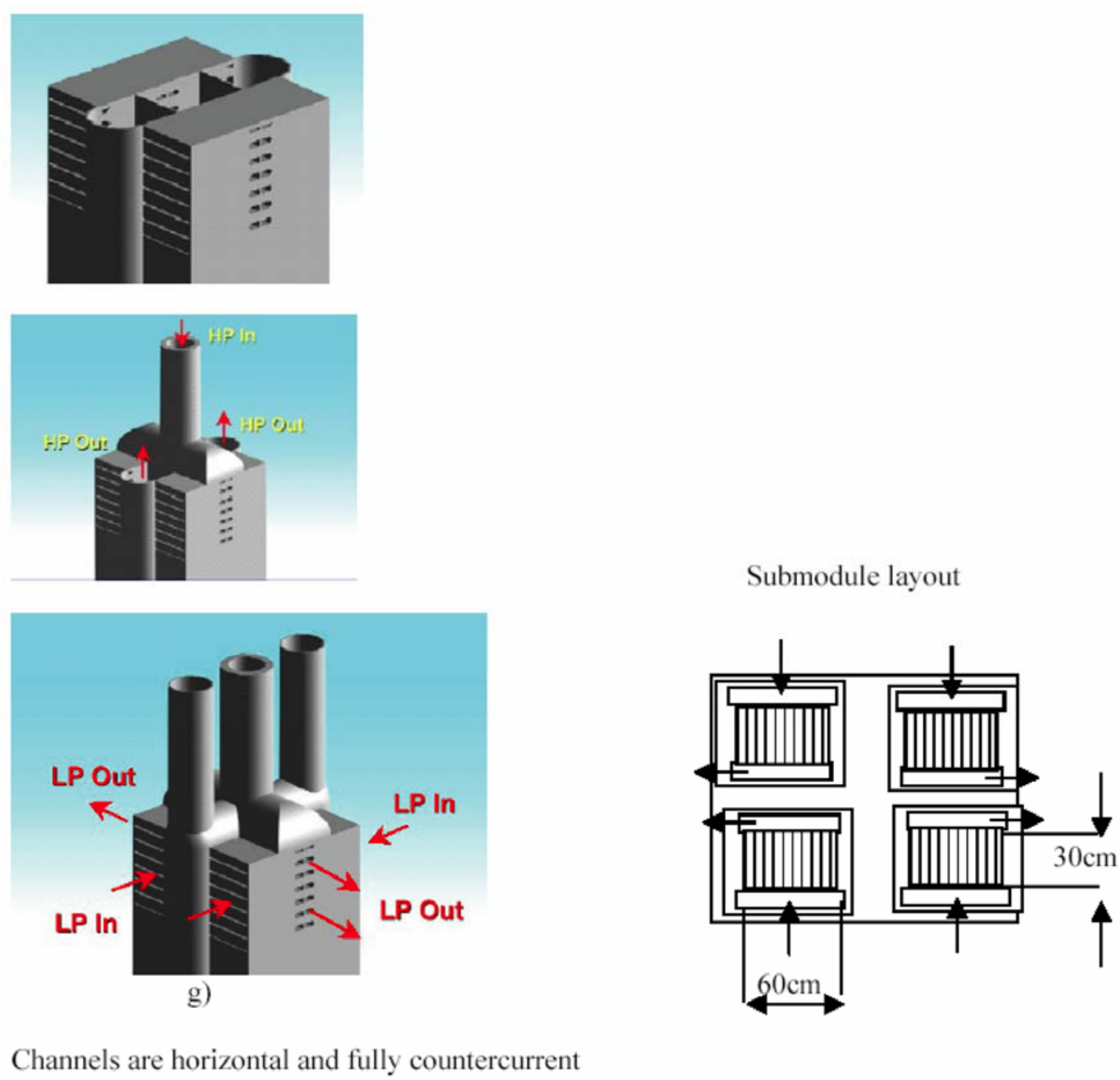


Figure 3-17. Counter Flow Heatric[R] Heat Exchanger (courtesy of HEATRIC Corp.)

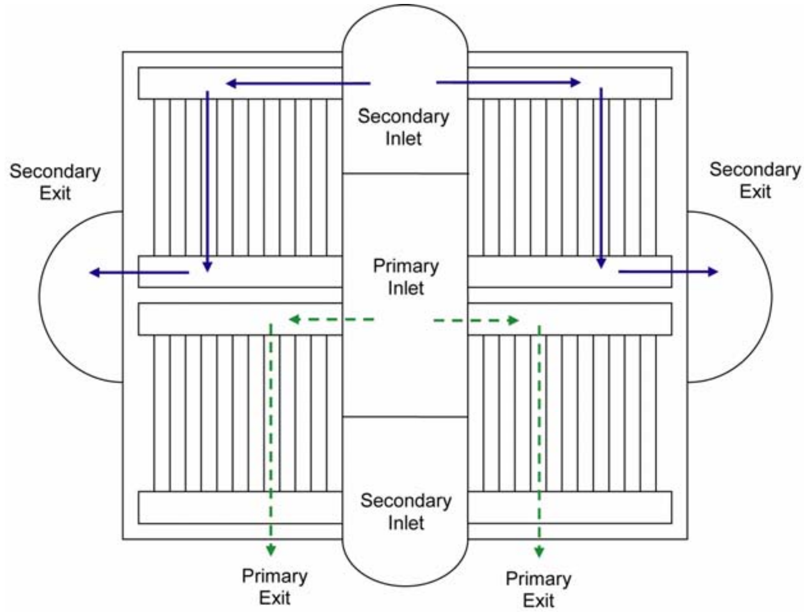


Figure 3-18. Primary and Secondary Flow Through IHX

The basic geometric parameters of this IHX design are presented in Table 3-8. Both the primary side and secondary side IHX flow channels are the same size. The log-mean temperature difference (LMTD) of the heat exchanger is 25°C and its effectiveness is 93.5%. The flow rates and pressures on the primary and secondary sides of the IHX are essentially the same. When the NGNP is operating in cogeneration mode to produce both electricity and hydrogen, the pressure in the primary system is reduced from 7 MPa to 6.23 MPa in order to maintain high efficiency in the PCS which is receiving 535 MWt of heat from the reactor. The pressure drop on the primary side is 31.2 kPa (4.53 psi). The pressure drop on the secondary side is essentially the same at 30.3 kPa (4.40 psi).

The NGNP IHX requires a nickel-based alloy, such as Alloy 617 due to the requirement to operate at temperatures of up to 950°C. Although the ASME Code does not presently support the use of Alloy 617 for stand-alone pressure containment, there appears to be adequate data to support design of PCHE modules as internals of the IHX pressure vessel. The external IHX pressure vessel will be designed and fabricated from existing ASME Code material. The IHX vessel is a pressure boundary for the primary helium coolant and will be designed according to the ASME Code, Section III.

Table 3-8. Basic Geometric Parameters of He-He PCHE IHX

Parameter	Value
Number of Modules	4
Module Height	1.82 m
Total Module Width (includes edges)	1.026 m
Edge Distance	13 mm
Total Module Length	0.90 m
Radius of Helium Channels	1.5 mm
Channel Center to Center Spacing	3.9 mm
Channel Offset Pitch	12.7 mm
Height of Offset	2.29 mm
Layer Thickness	2.4 mm
Flow Area per Module	0.3272 m^2
Heat Transfer Area per Module	680 m^2

The second PCHE-type IHX design was prepared by Toshiba by Toshiba. The design conditions for this IHX are presented in Table 3-9. The basic geometric parameters of the PCHE are shown in Table 3-10. The metal volume before etching the flow channels is 8.05 m^3 and has a mass of 67.3 tons. The large mass of the PCHE is due to the small log mean temperature difference (LMTD) which requires the PCHE to be quite long. Consequently, it would be desirable to increase the LMTD relative to the point design value of 25°C used in the NGNP preconceptual design.

Table 3-9. Toshiba PCHE-Type IHX Design Conditions

Parameter	Design Conditions
Heat Load, MWt	65
LMTD*, ^oC	25
Primary Side Flow Rate, kg/s	34.72
Primary Side Inlet / Outlet Temperature, ^oC	950 / 590
Primary Side Inlet / Outlet Pressure, MPa	6.2 / (6.15)
Secondary Side Flow Rate, kg/s	34.72
Secondary Side Inlet / Outlet Temperature, ^oC	565 / 925
Secondary Side Inlet / Outlet Pressure, MPa	6.1 / 6.05
Allowable Pressure Loss**, MPa	0.05
*LMTD = log mean temperature difference. **Tentative condition.	

Table 3-10. Basic Geometric Parameters of Toshiba PCHE IHX

Parameter	Value
Number of Modules	48
Total Module Height, m	0.453
Total Module Width, m	0.400
Total Module Length, m	0.960
Radius of Helium Channels, mm	1.5
Channel Center to Center Spacing, mm	3.9
Channel Offset Pitch, mm	12.7
Height of Offset, mm	2.286
Plate Thickness, mm	2.4

Eight PCHE modules are assembled by tungsten inert gas (TIG) welding to form a PCHE unit. The plenums are attached to both sides of the PCHE modules by TIG welding to form the flow passage for the secondary helium coolant. The material used for the plenum is Alloy 617, the same material as used for the PCHE. The cross section of the PCHE units is shown in Figure 3-19.

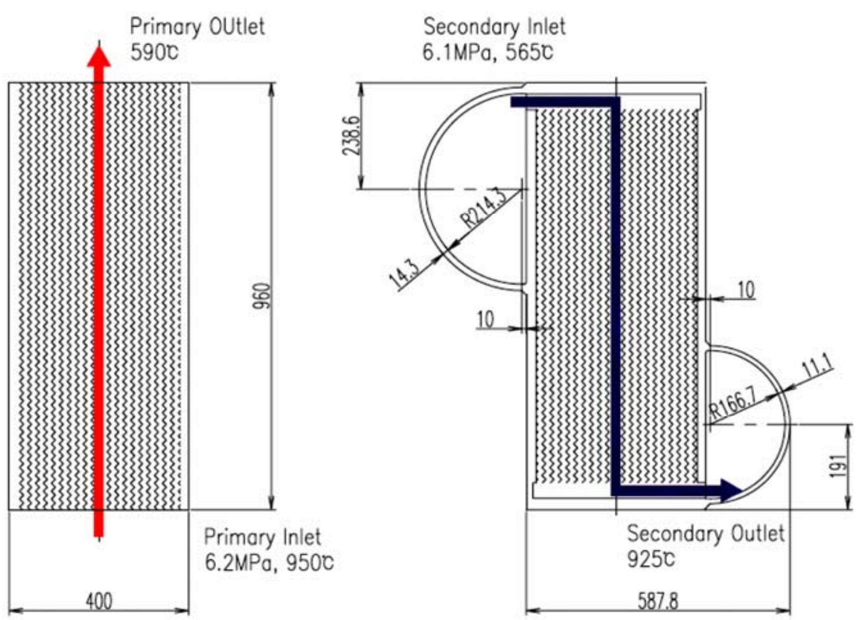


Figure 3-19. Horizontal Cross Section of PCHE Unit

The general arrangement of the IHX is shown in Figures 3-20 and 3-21. The primary coolant flows through the RV – IHX hot duct and upward into the bottom of the PCHE unit shell. The primary coolant at 950°C and 6 MPa flows upward through the PCHE units. Heat is transferred from the primary coolant to the secondary HTS helium in the PCHE units. The temperature of the primary coolant drops to 590°C and the coolant descends downward in the IHX and is carried to the helium circulator. The primary coolant is pressurized by the helium circulator, goes upward between the pressure vessel and shroud, and is transported through the annulus between the hot duct and cross-vessel to the RV. On the secondary side, the secondary HTS helium enters at 565°C and 6 MPa, flows into the inside of the IHX through the 8 inlet nozzles installed at the top spherical shell, goes through the piping, and is transported up to the plenum of the PCHE units. The helium flows through the PCHE unit where it heats up to 925°C, goes through the internally insulated piping, collects in the secondary outlet header, and is transported to the hydrogen production plants by the secondary HTS circulator. Kaowool is used as the thermal insulation for the piping.

Development of the PCHE IHX concepts for the NGNP preconceptual design have identified a number of design issues that need to be addressed by more detailed analyses as the design progresses. The following issues need to be addressed in conceptual design.

- The method for installation of the thermal insulation into the small pipe
- The size of cross-vessel and the helium circulator
- Maintenance considerations with respect to working space, fabrication methods, etc.
- Confirmation of secondary piping and PCHE support design feasibility
- IHX is lifetime; the IHX vessel is designed for 60 years, but the internals may need to be replaced in
- Implications of ISI on the PCHE as required by ASME Code. (Because of this, the primary coolant pressure boundary might be considered to extend to the isolation valves in the secondary HTS.)
- Methodology for monitoring the IHX for leakage
- Determination of the precise pressure drop through the PCHE by experiment
- Structural analysis is needed to confirm the feasibility of the design selections
- Many slide joints are used to enable maintenance of the IHX. An evaluation of leak rates due to the many slide joints used to accommodate maintenance

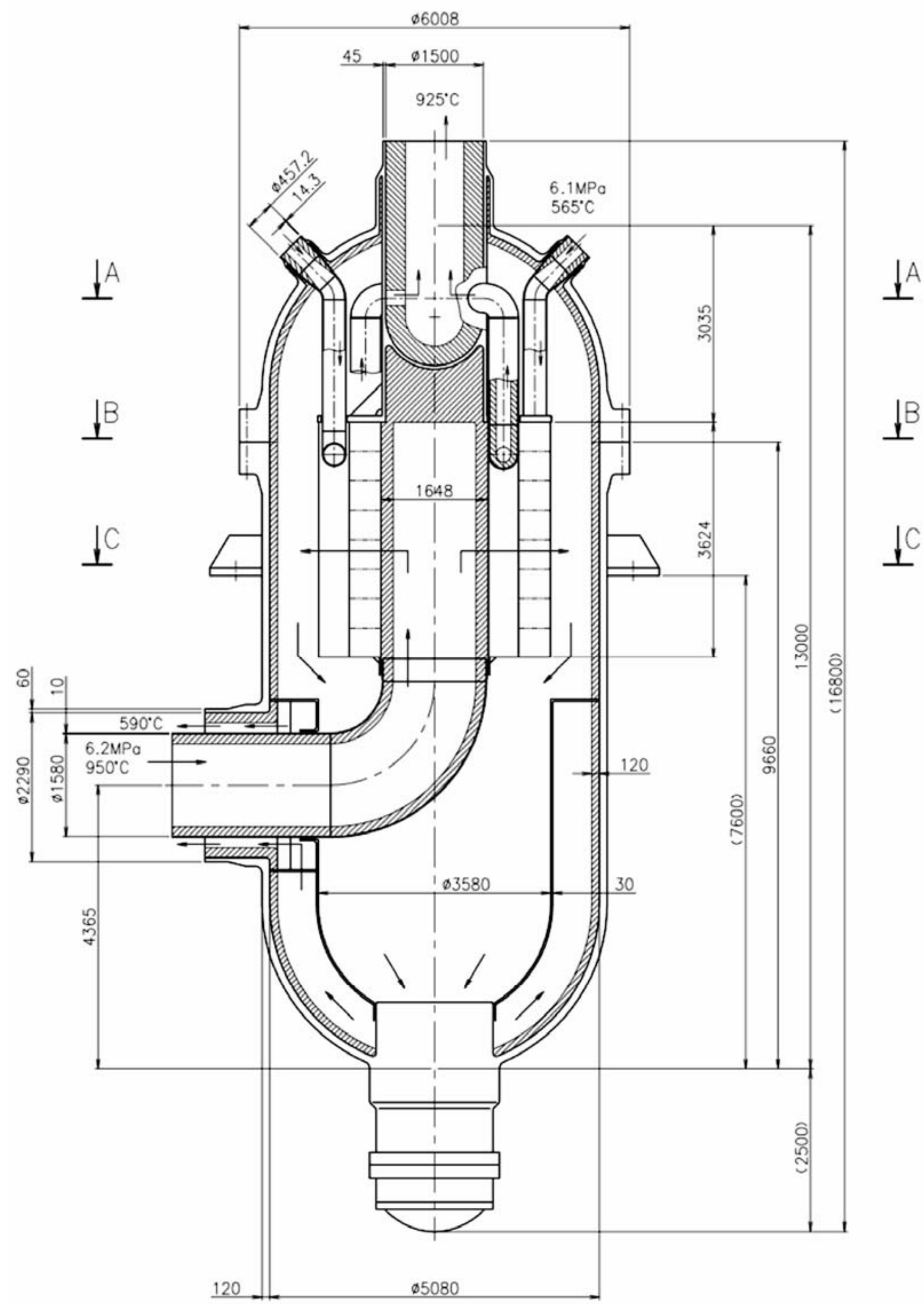


Figure 3-20. General Arrangement of PCHE IHX

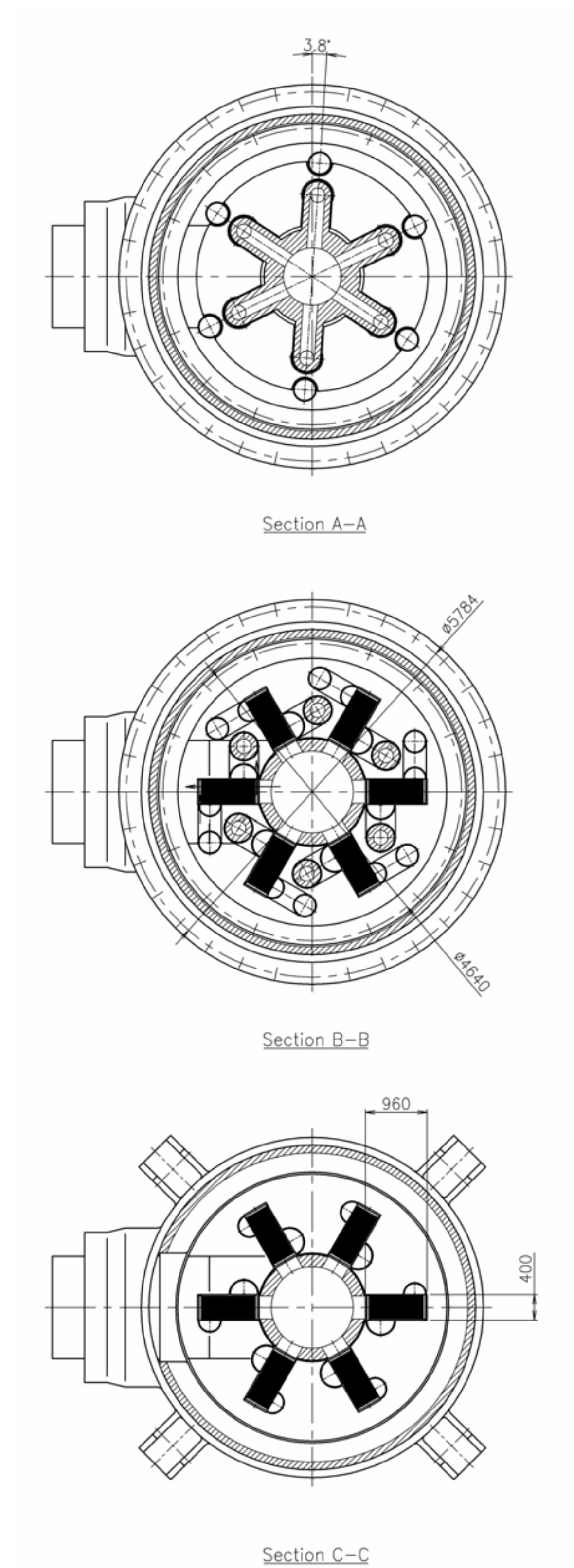


Figure 3-21. Horizontal Cross Section of PCHE IHX

The alternate Toshiba IHX design is a shell and tube, counter-flow heat exchanger, using a helically coiled tube. For an equivalent heat duty and LMTD, this type of heat exchanger is considerably larger than a PCHE. However, this design allows for in-service inspection (ISI) of the heat transfer tubes, and an IHX design of this type is connected to the HTTR and has successfully operated at 950°C for extended periods. To reduce the size of the helical-coil IHX, the LMTD would be increased to 91°C, which would alter the overall system heat balance and results in somewhat lower-temperature heat being transferred to the hydrogen-production processes. The design conditions for this IHX design are given in Table 3-11

The helical-coil type IHX concept is shown in Figure 3-22. In this figure, "1ry" stands for primary and "2ry" stands for secondary. Primary helium gas enters the center of the inlet nozzle, flows up through the region of tube bundles, returns at the upper end of the vessel, flows down through annulus path between inner shell and outer shell into the circulator, and from the circulator to the reactor through the annulus between the hot duct and the cross vessel. Secondary helium enters into four tube sheets at the head of the IHX, flows down through the helically-coiled tubes to a hot manifold header at the bottom of the center pipe, flows up through the center pipe and exits from the outlet nozzle at the top head of the IHX.

Table 3-11. Helical-Coil IHX Design Conditions

Parameter	Design Conditions
Heat Load, MWt	65
LMTD*, °C	91
Primary Side Fluid	Helium
Primary Side Inlet / Outlet Temperature, °C	950 / 590
Primary Side Inlet / Outlet Pressure, MPa	6.2 / 6.15
Secondary Side Fluid	Helium
Secondary Side Inlet / Outlet Temperature, °C	440 / 900
Secondary Side Inlet / Outlet Pressure, MPa	6.1 / 6.05
Allowable pressure drop, MPa**	0.05
*LMTD = log mean temperature difference. **Tentative condition.	

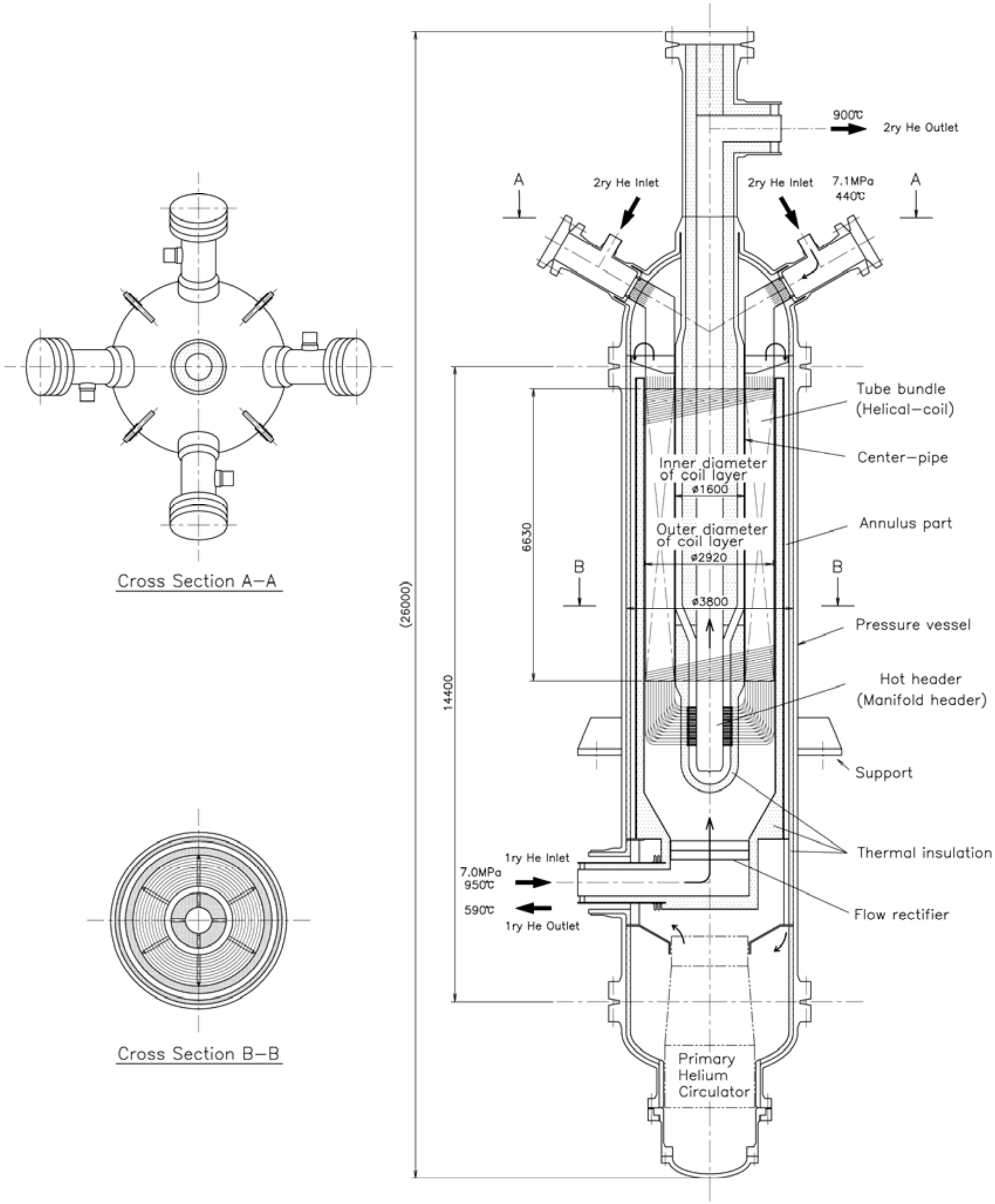


Figure 3-22. General Arrangement of Helical-Coil IHX

3.7.1.2 Primary Helium Circulator

The primary HTS helium circulator is mounted vertically at the top of the IHX vessel closure head and is part of the pressure boundary for the primary coolant. The helium flow rate can be adjusted by varying the speed of the motor. The circulator includes a loop shutoff valve for shutting off primary coolant flow through the circulator (and the primary HTS) when either the SCS circulator is operating or the NGNP is operating only to produce electricity.

The PHC is configured to accommodate the following items: (1) a variable speed electric motor, (2) an axial flow impeller and diffuser, (3) a loop shutoff valve (LSV), (4) an electric motor control and power subsystem (EMCPS), (5) a magnetic bearing control and power subsystem, (6) a labyrinth seal, (7) an internal circulator cooler, and (8) a barrier plate and motor outer sleeve. Figure 3-23 shows the general arrangement of the PHC.

The PHC is mounted vertically at the top of the IHX vessel (IHXV) closure head and is part of the pressure boundary for the primary coolant. Helium flow rate can be adjusted by varying the speed of the motor. The axial flow impeller is mounted to the bottom of the motor shaft. The cold return helium enters the circulator inlet, flows downwards through the impeller and the LSV, and is discharged into the circulator outlet plenums of the IHXV. The helium collected in the IHX vessel's outlet plenum is then returned to the RV via the IHX cross vessel. The LSV assembly shuts off primary coolant flow through the PHC when either the SCS circulator is operating or the NGNP is only operating to only produce electricity. Using a conservative value of 80% for the efficiency of the PHC, an initial estimated power requirement for the PHC motor is 1.5 MWe.

3.7.2 Secondary Heat Transport System

The Secondary HTS uses helium to remove heat from the IHX and transport it out of the RB to the hydrogen production plants. At the hydrogen production plants, the secondary coolant is divided into two flow paths in order to supply 60 MWt to the SI hydrogen production process and ~4 MWt to the HTE hydrogen production process. The secondary HTS circulators return the helium from the process heat exchangers back to the IHX. The secondary HTS consists of the secondary helium circulators, piping, and isolation valves. The preconceptual design of the Secondary HTS is based on the Heat Transfer/Transport study performed by GA [Bolin 2007].

Parallel hot leg and cold leg piping is used to transfer the process heat from the IHX to the hydrogen production plants. The piping is assumed to run 90 m in length between the IHX and process heat exchangers (PHXs) of either the SI or the HTE hydrogen production plant. The parallel pipe configuration is a simpler design compared to a concentric pipe configuration and can more easily accommodate the design features necessary to address thermal expansion and isolation valves. To reduce the pipe wall temperature and to meet the requirement for <1% heat loss to the environment, internal insulation is used for both the hot leg and cold leg piping. The internal insulation would be made of Kaowool with cover plates holding it in place. The same design approach was used in the FSV HTGR and is proposed for the hot ducts and cross vessels of the NGNP primary system. External insulation is also used to further reduce the heat loss to the environment. The addition of external insulation raises the pipe wall temperature.

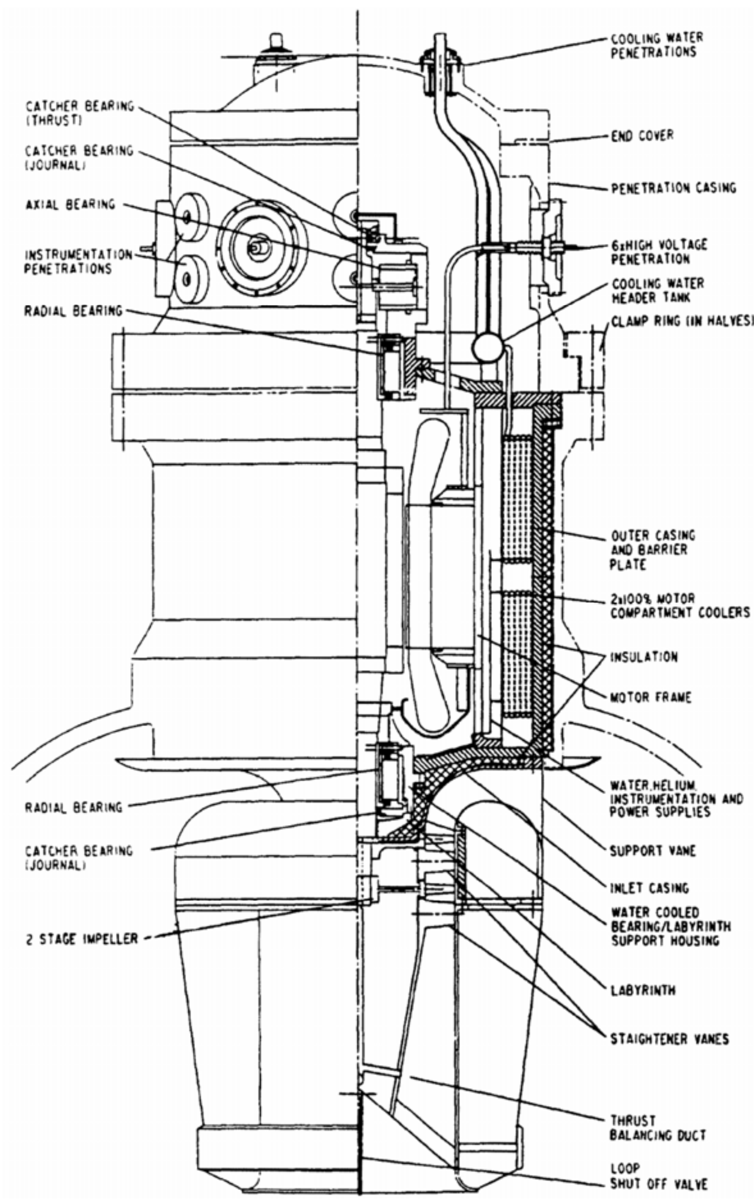


Figure 3-23. General Arrangement of Primary Helium Circulator

The secondary HTS is assumed to have three isolation valves on each leg – two near the IHX and one near the PHX. Isolation valves are necessary to prevent the propagation of events in either the NGNP reactor or hydrogen production plant from affecting the other. Double isolation valves on the hot leg and cold leg sides of the IHX allow these isolation valves to be part of the primary coolant pressure boundary and part of the containment building boundary. Isolation valves are also necessary to perform maintenance on the heat transport loop. Figure 3-24 presents a diagram of a potential high temperature isolation valve (HTIV) being developed for use on HTTR by the Japan Atomic Energy Agency (JAEA). For HTTR, a ½ scale prototype of the HTIV has been tested. The valve, as shown in Figure 3-24, is an angle valve with internal glass wool insulation. The rod body and seat were made of Hastelloy X and the seat had a coating metal of Stellite No. 6 and 30 wt% Cr_3C_2. The casing of the valve was made of carbon steel which was limited to 350°C due to the internal insulation. Testing was performed at 4.0 MPa and 900°C.

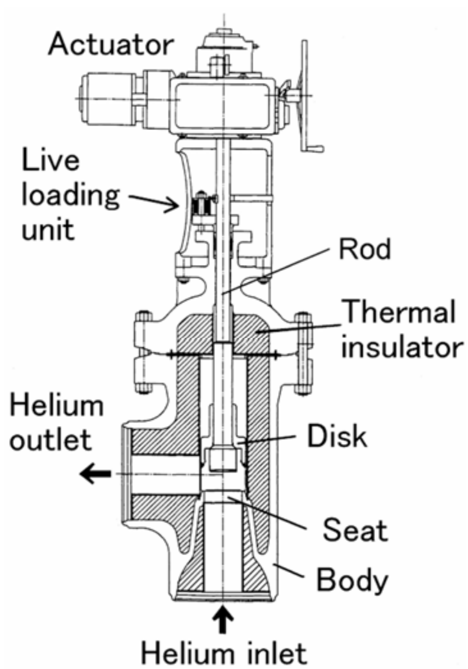


Figure 3-24. High Temperature Isolation Valve

The design of the secondary HTS helium circulator is expected to be either the same or very similar to the primary HTS helium circulator. The major difference between the secondary HTS circulator and the primary HTS circulator is the absence of a loop shutoff valve whose function is performed by the secondary loop isolation valves. The total pressure drop in the secondary HTS is the sum of pressure drops in the IHX, PHXs and piping system. The design of the PHX for either the SI or HTE hydrogen production process is expected to have a pressure drop no greater than the pressure drop in the IHX. Assuming the system uses a single circulator with an efficiency of 80%, the total system pumping power would be 1.6 MWe. An alternate configuration is to have two circulators – one dedicated to SI hydrogen production and the other dedicated to HTE hydrogen production. The SI secondary HTS circulator would be sized at 1.45 MWe while the HTE secondary HTS circulator would be sized at only 120 kW.

3.8 Hydrogen Production Systems

3.8.1 High-Temperature Electrolysis System

As described in [Richard 2006b], GA and INL developed a pre-conceptual commercial H2-MHR design based on coupling the MHR to SOE modules. In that H2-MHR concept, the SOE modules are based on the planar-cell technology being developed by INL and Ceramatec of Salt Lake City, UT under the NHI.

For the current study, GA has worked with Toshiba Corporation to develop a concept based on tubular-cell technology. The tubular-cell concept requires more cell area per unit volume (which may impact capital costs), but appears to have fewer technical issues with regard to sealing individual cells, which can have a significant impact on long-term performance. GA believes both the planar-cell and tubular-cell technologies are promising concepts for future commercialization, and recommends that both concepts be developed through at least the pilot-scale demonstration stage so that tradeoffs between capital costs and long-term performance can be accurately characterized.

High-temperature electrolysis requires SOE cells that can operate at temperatures up to approximately 850°C. Figure 3-25 shows a schematic of the Toshiba SOE cell design. The electrolyte is Yttria-Stabilized Zirconia, the anode (oxygen electrode) is LSM (Strontium-doped Lanthanum Manganite), and the cathode (hydrogen electrode) is Ni-YSZ (a mixture of metallic Nickel and Yttria-Stabilized Zirconia).

The HTE-based hydrogen production plant for the NGNP will utilize ten SOE modules, with each module containing approximately 18,000 SOE cells and producing 600 Nm^3 of hydrogen per hour (0.015 kg/s). Modules of the same size would be used for a commercial-scale plant. The SOE module design parameters are given in Table 3-12. Figure 3-26 shows the pre-conceptual SOE module concept. The module pressure vessel is designed to last the 60-yr plant lifetime. The electrolyzer cells are expected to last between 5 and 10 years before requiring replacement, but additional technology development/demonstration is required to determine actual cell lifetime.

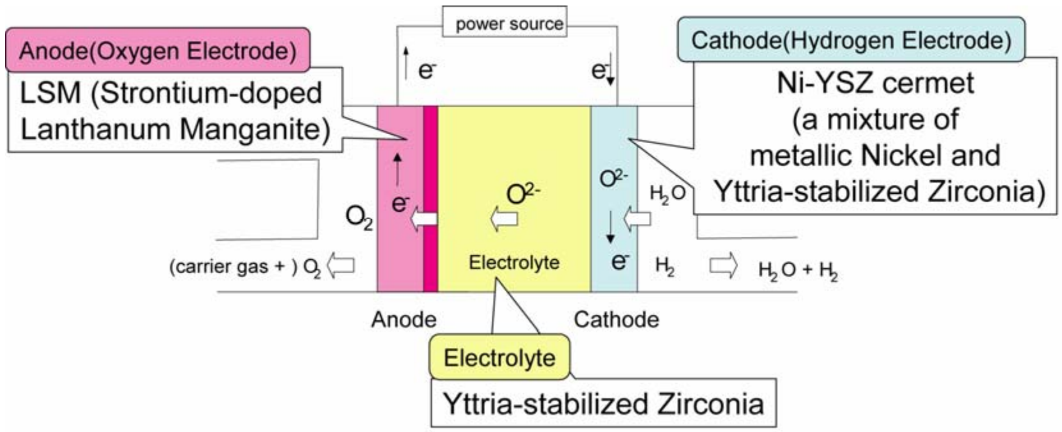


Figure 3-25. Schematic of SOE Cell Concept

Table 3-12. SOE Module Design Parameters

Design Parameter	Value
Internal design temperature, °C	900
Inlet/outlet temperatures, °C	815/849
Vessel temperature, °C	200
Vessel pressure, MPa	5
ΔP between anode and cathode, MPa	0
Hydrogen electrode inlet gas composition, H_2/H_2O mole fraction	0.1/0.9
Hydrogen electrode outlet gas composition, H_2/H_2O mole fraction	0.9/0.1
Oxygen electrode inlet gas composition, O_2/H_2O mole fraction	0.0/1.0
Oxygen electrode outlet gas composition, O_2/H_2O mole fraction	0.6/0.4
Electrolysis cell shape	Cylindrical
Current density, A/cm^2	0.6
Operating voltage, volts	1.304
Electrical energy input, MW	1.86
Hydrogen production rate, Nm^3/h	600

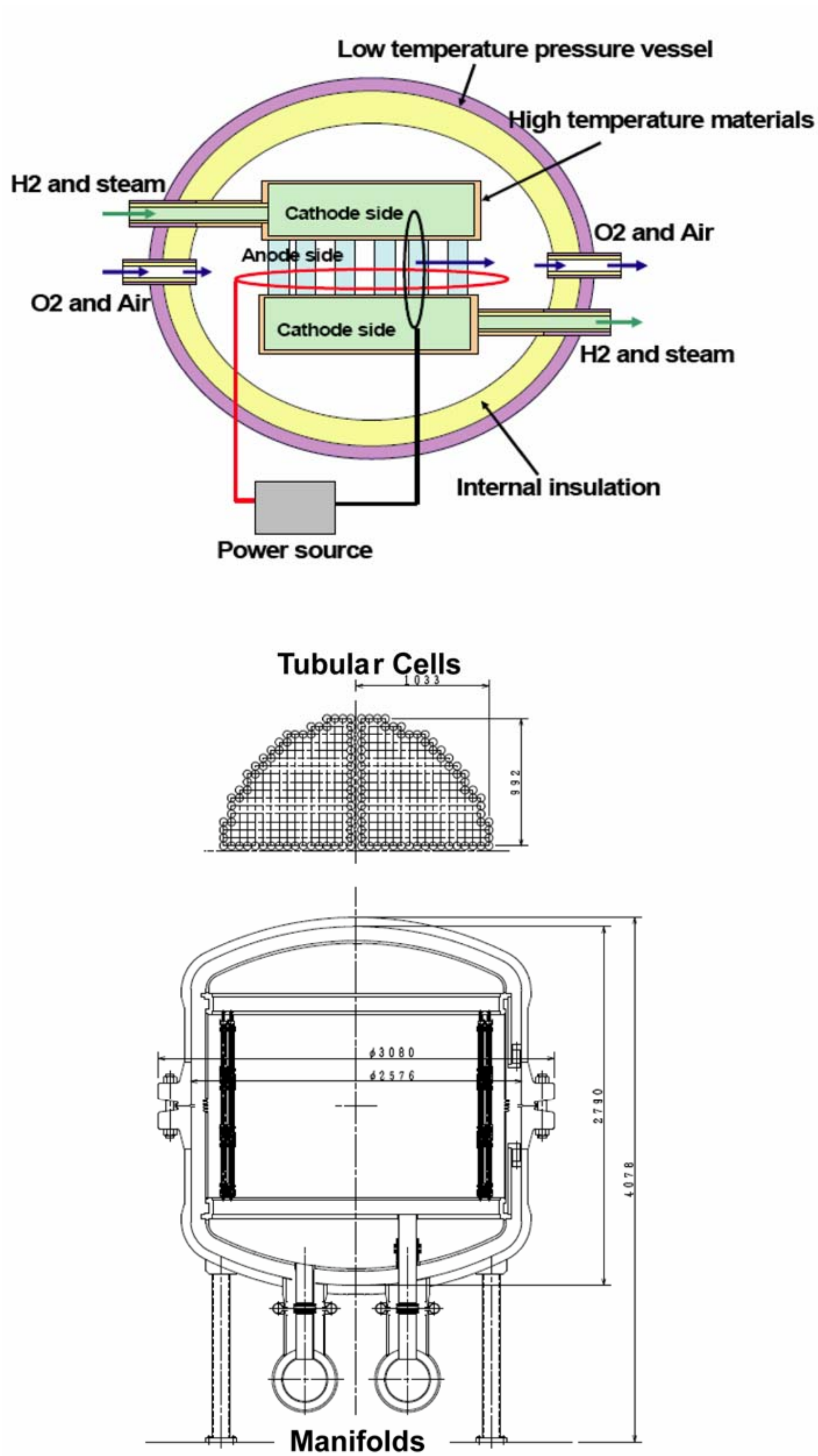


Figure 3-26. Pre-Conceptual SOE Module Concept (dimensions are in mm)

As indicated in Table 3-12, the cells are designed to operate at 1.304 volts and at a current density of 0.6 A/cm^2. Figure 3-27 shows the current-voltage characteristics measured for a single, 15-cm^2 cell at temperatures of 800°C and 900°C with P_{H2O} = 0.5 atm, P_{H2} = 0.5 atm, and P_{O2} = 0.2 atm. The measured open-cell potentials (corresponding to a current density of zero) at 800°C and 900°C were 0.94 volts and 0.91 volts, respectively, which are in good agreement with the theoretical values predicted using the Nernst equation. For this test, current densities of approximately 0.45 A/cm^2 were achieved. In order to operate at higher current densities, it is important that the cells have low area-specific resistance, ASR = (E – E_{OCV})/I, where E is the operating voltage, E is the open-cell potential, and I is the current density. The ASR values were 0.63 ohm-cm^2 at 800°C and 0.37 ohm-cm^2 at 900°C. Testing of a 75-cm^2 cell (3 times the length of the 15 cm^2 cell) at 800°C resulted in a current density of about 0.3 A/cm^2 at the thermal neutral voltage and an ASR of approximately 1.2 ohm-cm^2. Under the same test conditions, testing of a cell assembly consisting of three banks of five 75-cm^2 cells (total of 15 cells, see Figure 3-28) resulted in a current density of about 0.2 A/cm^2 at the thermal neutral voltage and an ASR of about 1.9 ohm-cm^2. Based on these results, additional technology development is needed to achieve high current densities for engineering-scale units.

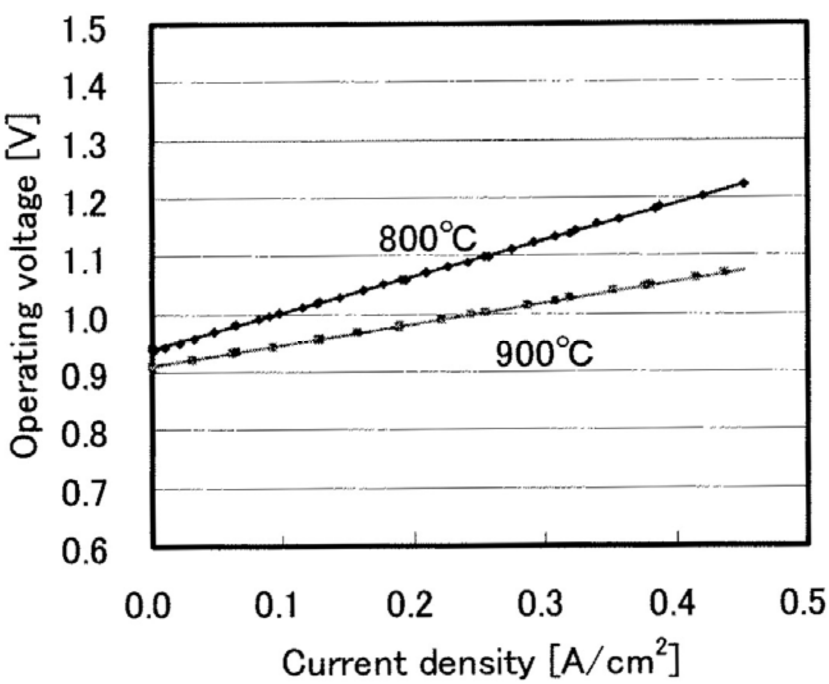
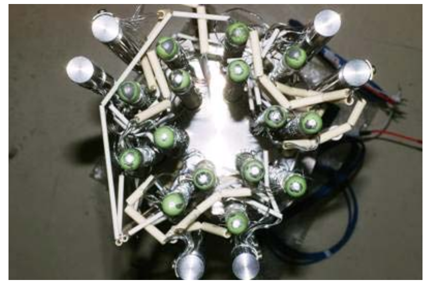
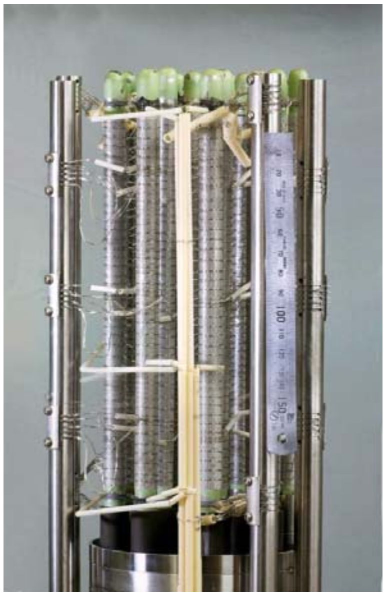


Figure 3-27. Current-Voltage Characteristics of a Single, 15-cm^2 Tubular Cell



Top View



Side View

Figure 3-28. Toshiba 15-Cell Assembly

Figure 3-29 shows the flowsheet developed for the NGNP, which was analyzed using the commercial HYSYS process simulation software. For this pre-conceptual study, heat losses associated with process equipment and piping were neglected, losses associated with AC to DC conversion were neglected, and pressure losses associated with components were assumed to be 1% of the inlet pressures to these components. SOE cell performance was based on data developed by Toshiba as part of their HTE technology-development program [Matsunaga 2006]. Make-up water is mixed with recycled water and then heated and vaporized. The steam is mixed with recycled hydrogen before it is supplied to the SOE modules in order to ensure reducing conditions and prevent oxidation of the hydrogen electrodes. The flow sheet includes heat exchangers to recuperate heat from the hydrogen/steam and oxygen/steam streams exiting the electrolyzer modules and drums to separate moisture from these streams. A small expander turbine (T-201) is used to recover energy from oxygen stream and generate more than sufficient electricity for pumps, compressors, and other electrical loads associated with the process. [PCDSR 2007] includes tables giving the stream compositions, vapor fractions, flow rates, and temperatures for the flowsheet and also the design conditions and equipment size estimates for the heat exchangers, compressors, turbine, drums, and pumps.

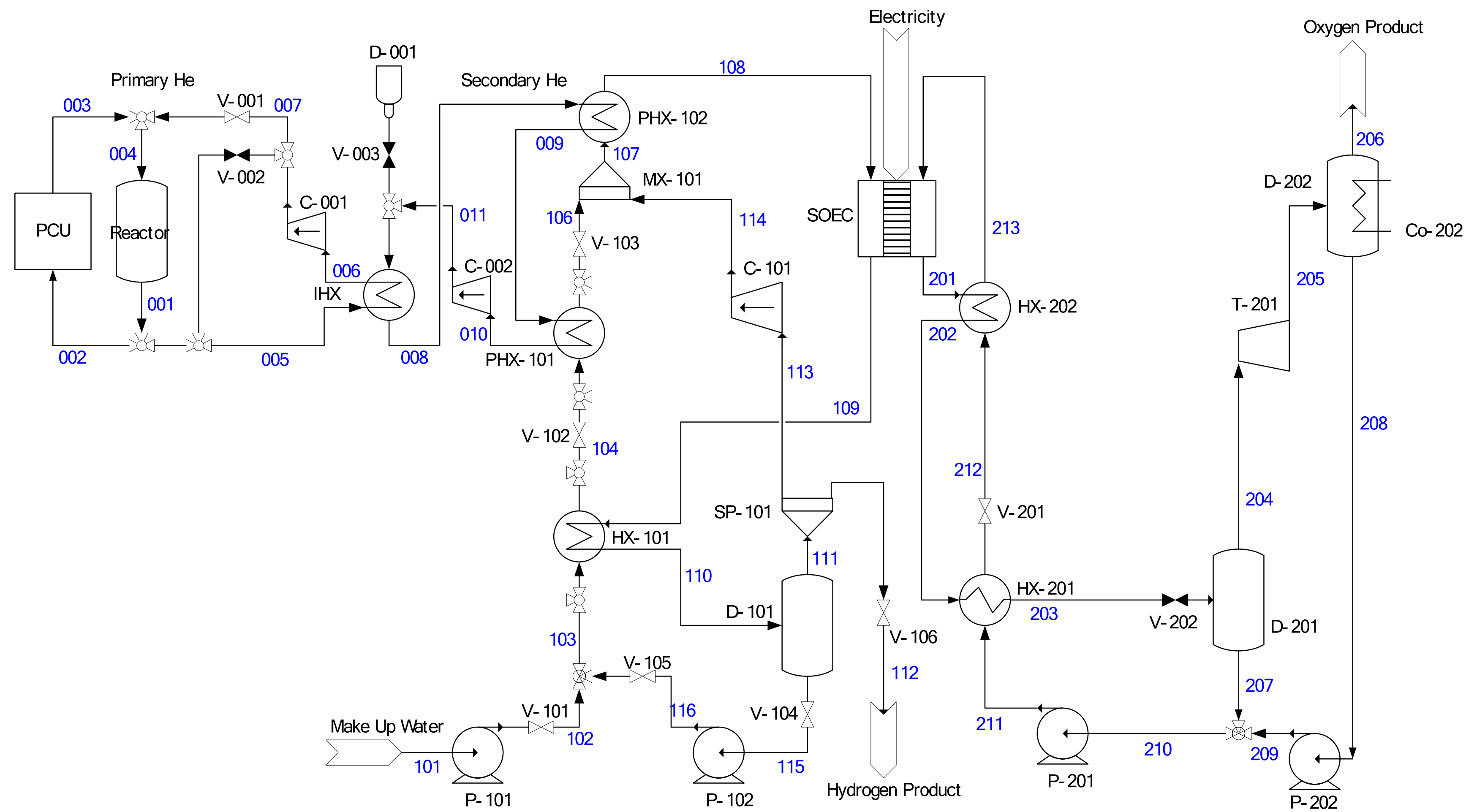


Figure 3-29. HTE Hydrogen Production Plant Process Flowsheet

For the present study, electricity is generated in co-generation mode with a thermal efficiency of 50.5%. The 10 SOE modules require a total of 18.7 MWe and T-201 produces a net 0.41 MWe after process electrical loads are taken into account. A total of 3.59 MWt of heat is supplied to the process through the IHX. Using the higher-heating value of hydrogen, the thermal energy of the hydrogen produced is 21.3 MWt. The overall efficiency of the process is estimated to be:

$$\eta_{HTE} = \frac{21.3}{3.59 + (18.7 - 0.41)/0.505} \times 100 = 53.5\% \quad .$$

Sensitivity studies were performed to determine the effect of operating pressure and temperature on overall system efficiency. For a reduced operating pressure of 0.5 MPa, the predicted efficiency was only about 1% lower. For SOE module inlet temperatures over the range 750°C to 850°C, there is a small increase in IHX heat duty [from 3.42 MWt to 3.67 MWt] and a small decrease in SOE module electric power requirement [from 18.81 MWe to 18.56 MWe], which results in only a slight increase in efficiency (about 0.3%) over this temperature range. This estimate assumes the electricity generation efficiency remains constant at 50.5%.

3.8.2 Sulfur-Iodine Cycle Process

The Sulfur-Iodine (SI) cycle produces hydrogen from water through a series of three chemical steps (or sections) as depicted in Figure 3-30.

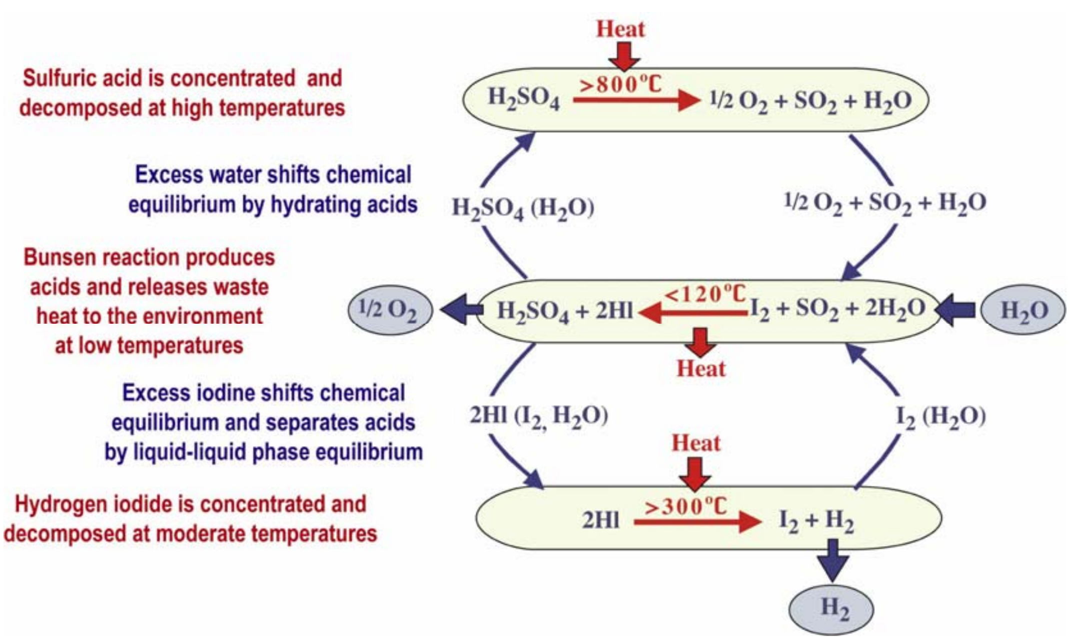


Figure 3-30. Sulfur-Iodine Cycle

Two immiscible acid phases are created by combining excess water with iodine and sulfur dioxide, and their separation is facilitated by an excess of iodine. This step is known as the Bunsen reaction and is designated as Section 1. The resulting sulfuric acid phase is decomposed back into sulfur dioxide for reuse. This step (Section 2) is the highest temperature point (>800^{o}C) in the process. Hydrogen iodide is separated from water and iodine in Section 3 before being decomposed into hydrogen and more iodine. Decomposition typically occurs between 300-500^{o}C. The water and iodine are returned for reuse. It can be driven purely by energy in the form of heat, but electrical energy is often used in flowsheets where appropriate to boost efficiency. Heat pumps and vapor recompression equipment are examples of electrically-powered equipment seen in SI cycle flowsheets.

Figure 3-31 is a block diagram of the SI cycle. It shows the material connections between the chemical steps and the fundamental energy requirements for the key chemical reactions. ΔH is the enthalpy demand and ΔG is the associated Gibbs free energy. The acid-generating step in Section 1 is endothermic, and efficient recovery of this low-temperature heat can boost process efficiency. The decompositions of sulfuric acid and hydrogen iodide are exothermic, and minimizing energy inputs to these sections is the focus of design of high-efficiency flowsheets. Each process section and its flowsheets are discussed in more detail below.

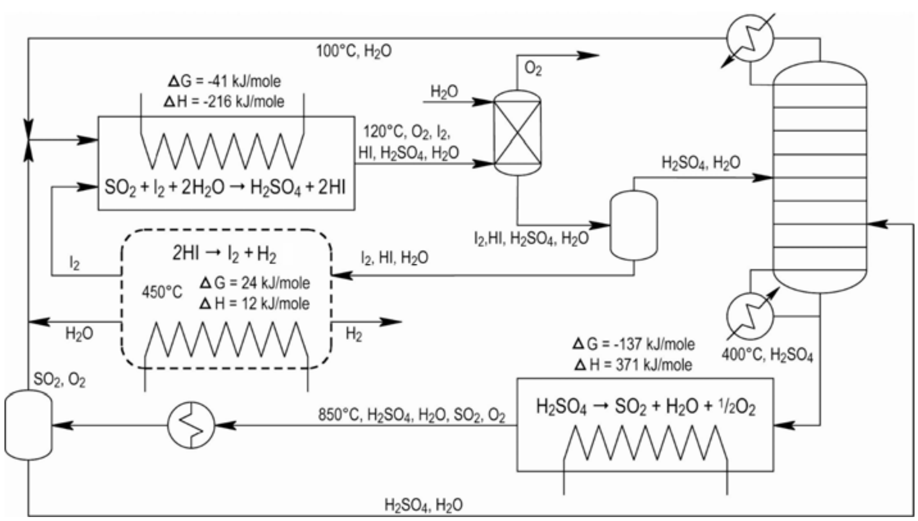


Figure 3-31. Basic Flowsheet of the SI Cycle

3.8.2.1 Bunsen Reaction (Section 1)

The Bunsen reaction is the initial step in the SI process. Gaseous sulfur dioxide is contacted with water and molten iodine to form sulfuric acid and hydrogen iodide (HI). The melting point of iodine is 114°C, so the minimum pressure to run the reactor with liquid water is 1.6 bar (23 psia). Typical operating conditions are 5 to 7 bar pressure and 120°C in temperature. Products are exchanged in each direction between Sections 1 and 2, and between Sections 1 and 3. Acid concentrations are increased in a multi-step process within Section 1, and sulfur compounds are stripped from the HI phase before it is sent to Section 3. The oxygen product from the SI cycle is vented from Section 1. No heat energy is required in Section 1 and only liquid pumping power is necessary. The unused energy of the SI cycle is largely ejected in Section 1 in the form of low-grade heat.

3.8.2.2 Sulfuric Acid Decomposition (Section 2)

The sulfuric acid phase generated by the Bunsen reaction is decomposed back into sulfur dioxide for recycle and reuse in Section 2. Figure 3-32 is a detail of the Section 2 flowsheet that shows the decomposition steps. The acid is concentrated in a series of vaporizers before boiling. Sulfur trioxide is produced in the gas phase and sent to a high-temperature (>800°C) decomposition step to produce sulfur dioxide and oxygen. As shown in Figure 3-32, process heat transported from the reactor via the secondary HTS is introduced into the cycle in Section 2. The only link between the nuclear heat source and the SI process is through heat exchange in the sulfuric acid decomposition step.

3.8.2.3 Hydriodic Acid Decomposition (Section 3)

Several methods have been proposed for decomposition of HI. Electro-electrodialysis has been studied, yet there have been difficulties in experimental verification of the technique. Reactive distillation is attractive, as the flowsheet estimated efficiency is approximately 45%. However, the only recent experimental work done (by GA) did not show promising results. The presence of iodine in the distillation column severely hampered conversion of HI. Thus, extractive distillation (previously demonstrated by GA) of the HI-water-iodine (known as HI_x) feed has been chosen as the technique to be used in the ILS experimental device described in Section 1.3.1.

With this method, the HI_x feed is contacted with concentrated phosphoric acid in a liquid-liquid extraction step. The HI and water are pulled into the acid phase, and the iodine is returned to Section 1. Pure HI is distilled from the water and phosphoric acid and decomposed over a carbon catalyst to produce hydrogen. The acid is reconcentrated and recycled.

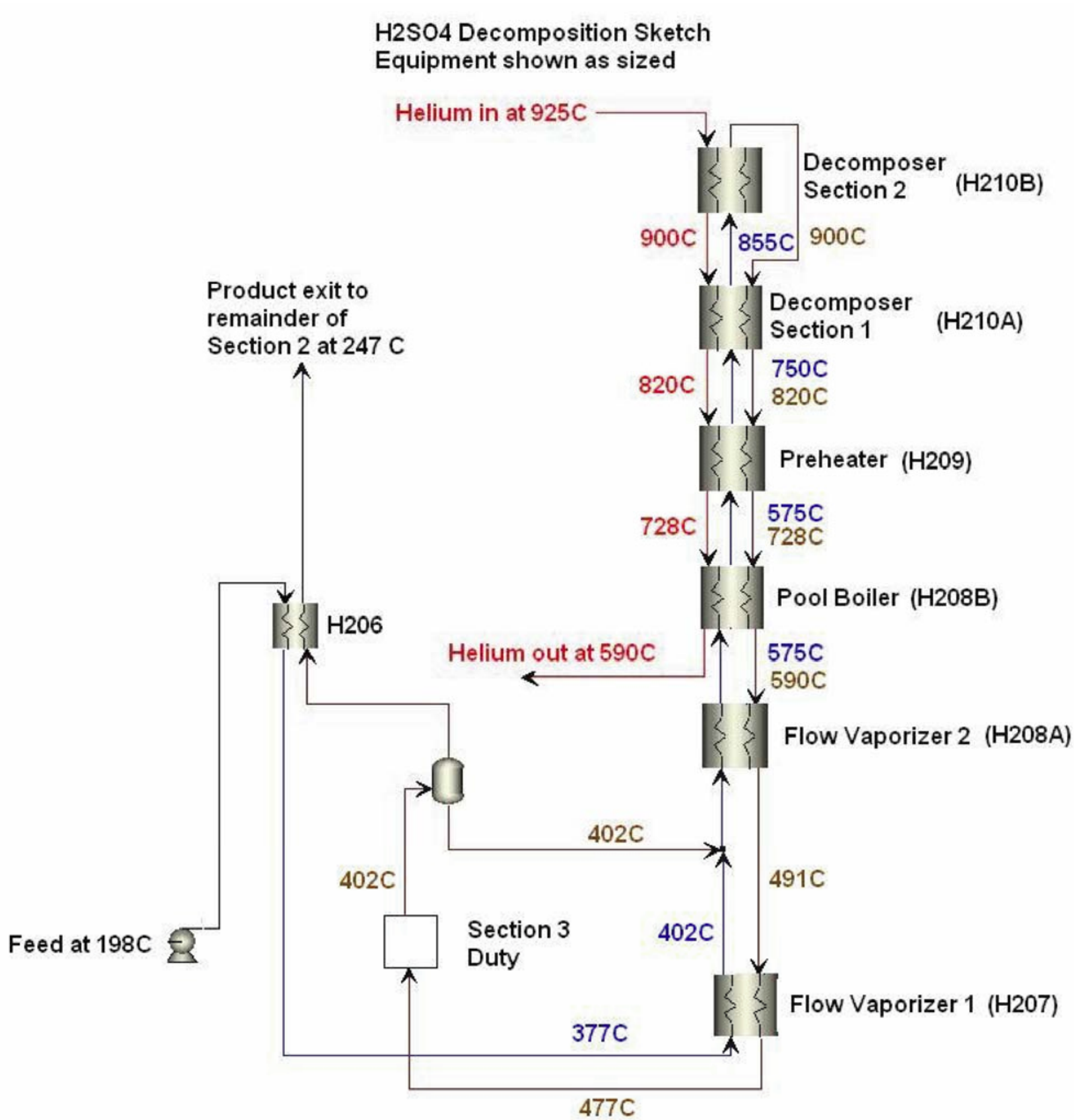


Figure 3-32. Flow Diagram of Sulfuric Acid Decomposition Step

3.8.2.4 Major Equipment Description

The NGNP SI-hydrogen plant will consist of three trains of equipment. This will demonstrate the multi-train process control scheme expected to be used in a full-scale plant. Lists of the vessels, the heat exchangers, and the turbo-machinery to be used in the NGNP SI demonstration plant are provided in [PCDSR 2007].

3.8.2.5 Flowsheet Analysis and Efficiency Assessment

The flowsheets for each portion of the process have been shown in the above. Also, as noted, there is significant heat exchange between Sections 2 and 3. Each flowsheet has undergone analysis and optimization to maximize overall efficiency. However, the overall efficiency will be dependent upon the temperature supplied by the nuclear heat source. Figure 3-33 is a plot of SI cycle process thermal efficiency as a function of temperature.

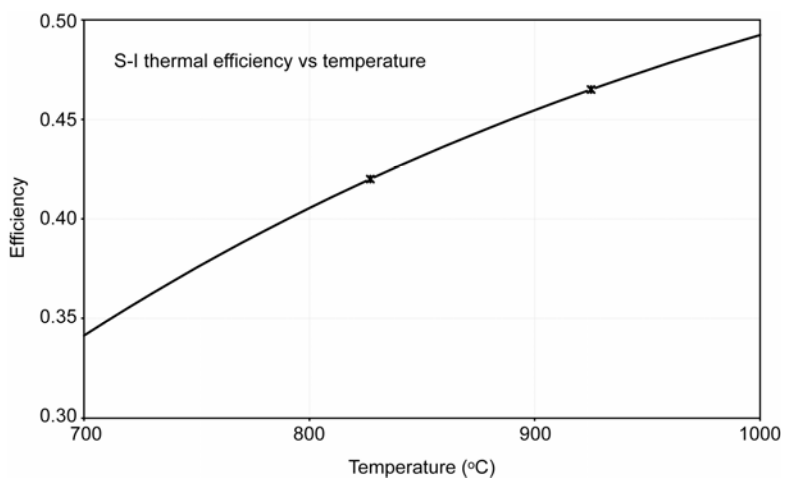


Figure 3-33. SI Cycle Process Efficiency vs. Temperature

The realistic estimates for the overall efficiency of the SI process are under 50% for the temperature ranges within reasonable consideration. However, the thermal efficiency will remain above 40% for temperatures as low as 800°C.

3.9 Helium Services System (HSS)

The Helium Services System includes the Primary Helium Purification System (HPS), the Helium Transfer and Storage System, and the Liquid Nitrogen System. The designs of these systems for the NGNP are the same as for the GT-MHR. However, an additional helium purification system will be needed for the secondary HTS. Although the primary purpose of the secondary HTS HPS will be to remove tritium, it will have essentially the same design as the Primary HPS. These systems are summarized briefly below and are described in more detail in [PCDSR 2007].

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3.9.1 Primary Helium Purification System

This subsystem provides a means to remove circulating impurities from the primary coolant helium, and to transfer those impurities to the radioactive liquid and gas waste systems of the facility. A separate regeneration section within this subsystem is used to remove the impurities that accumulate in the purification subsystem adsorbers. The regeneration section is operated periodically under automatic control whenever regeneration is required.

The primary coolant helium purification subsystem consists of two separate, independent, but identical trains of components as shown in Figure 3-34. One of these trains is always on-line, while the other is either being regenerated or is otherwise maintained in a stand-by status ready for immediate use. All of the components that make up the trains are mechanically passive in nature; however, the adsorber elements become radioactive as the removed impurities are concentrated within the various media. Each purification train must therefore be located in a shielded vault to minimize personnel exposure to radiation.

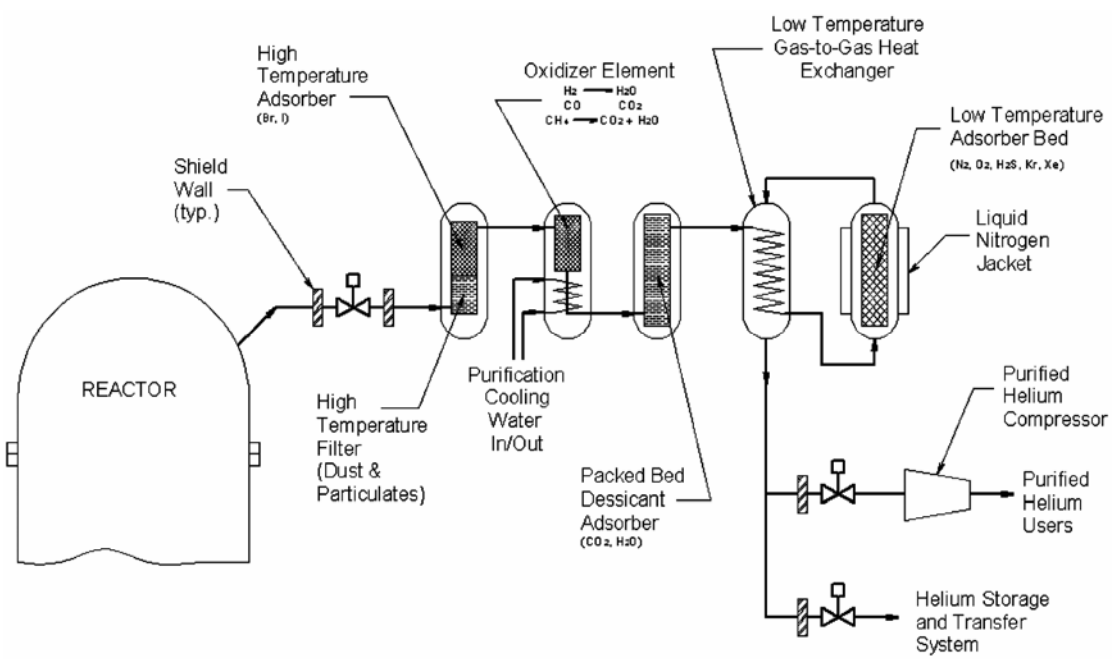

Figure 3-34. Primary Helium Purification Sub-System Schematic Diagram

Helium purification is accomplished by routing a small side stream of helium from the primary coolant system through a series of purification components as shown in Figure 3-34. These components remove the following chemical impurities: Br, I, H_2O, CO, CO_2, H_2 (including Tritium), N_2, O_2, H_2S, Kr, Xe, CH_4, and other hydrocarbons. In addition, certain metallic chemical elements and filterable particulates are also removed. All helium transferred from the primary coolant system must first pass through a purification train before entering the helium storage subsystem.

When the adsorber elements in a purification train become saturated with impurities, the train is taken off-line for regeneration, which is accomplished using the equipment in the regeneration section of the helium purification subsystem. The regeneration section is described in [PCDSR 2007],

3.9.2 Helium Transfer and Storage System

The helium transfer and storage subsystem provides for the movement of primary coolant helium to and from the vessel system and the nearby helium storage tanks. During normal plant operational load changes, helium is either released to the storage system from the vessel system (via the on-line helium purification train), or added [via equalization or the transfer compressor(s)], as required to maintain the correct helium inventory in the vessel system. Helium is also provided by this system as needed for various purging operations around the plant as well as for the maintenance of buffer seal flows and pressures at various locations within the primary system.

3.9.3 Liquid Nitrogen System

Liquid nitrogen is supplied to the low temperature adsorbers in the helium purification systems via vacuum-jacketed (or other equivalently insulated) transfer piping. The liquid nitrogen subsystem provides a flow rate sufficient to service the low temperature adsorbers on a continuous basis. Liquid nitrogen is stored in a large cryogenic tank situated at a physical elevation above that of the low temperature adsorber units in the helium purification system trains. Elevation of the storage tank relative to the adsorbers allows the system to operate on a gravity feed basis, thus allowing the two-phase flow exiting the adsorber to rise by natural convection back to the storage tank. A small cryogenic pump can be used if it is determined that the subsystem pressures are insufficient to provide adequate flow.

3.9.4 Tritium Control

The important function of the primary coolant HPS is to remove chemical impurities from the circulating helium. One of the more important of those impurities is tritium. This isotope of hydrogen has several sources, all of which are inherent in the design of a high temperature graphite moderated reactor. These sources are: ternary fission, neutron activation of the He-3 isotope found in the helium coolant, activation of Li-6 impurities in the core materials, principally graphite, and neutron activation of the B-10 contained in the poison materials used to control the reactor. Some degree of restraint over these sources can be obtained by careful engineering and manufacturing of the elements from which the impurities could be generated. To the extent that the production of tritium from these sources can be minimized, there will always be a certain inventory of tritium circulating in the primary coolant.

The primary means of minimizing the concentration of circulating tritium, beyond the quality of the design and engineering of the sources, is to constantly remove it. There are two primary processes by which this removal can occur in a high temperature graphite moderated reactor. One is through chemisorption on graphite, and the second is by way of a primary coolant helium purification system. Studies and investigations over many years have concluded that a very large portion of the circulating tritium is adsorbed within the core graphite as a routine function of reactor operation. However, it is considered imperative that an active process should be included in the plant design to remove the tritium, thereby maximizing control over this isotope.
As above, helium purification systems for both the primary helium and for the helium working fluid in the secondary HTS are included in the plant design to accomplish the necessary removal process.

3.10 Plant Operation and Control Systems

The Reactor Protection System (RPS) and the non-safety-related Investment Protection System (IPS) in the NGNP specifically provide the “defense in depth” design strategy that is required for modern reactor plants. The Plant Control, Data, and Instrumentation System (PCDIS) provides normal control and instrumentation functions, and also provides overall integration of the control and protection functions into a combined plant control system. This system provides normal (main loop) cooling if possible following a reactor trip, broadening “defense in depth” design features by making the SCS or RCCS less likely to be used for reactor cooling.

3.10.1 Reactor Protection System and Investment Protection System Functions

The function of the combined protection systems is to detect and provide corrective action as follows:

- In the event of changes (including changes in neutron flux and primary coolant flow rate, or temperature) indicating neutron flux elevations in the reactor are beyond the range of normal reactor operation.
- If changes in the RB (including changes in temperature, pressure and radiation levels) indicate a release of primary coolant at a level that could expose the general public to low-level radiation effects.
- If conditions of pressure, temperature or flow indicate an interruption of normal cooling functions.
- If upset of reactor power utilization processes creates a condition which could damage major components, such as the TM.
- If conditions of pressure and temperature, within and around the Vessel System primary coolant boundary, indicate a level of operation that exceeds the normal Vessel System design levels.
- If conditions of environment or service to the reactor system indicate an interruption of normal processes that are not protected by 1E electric services or are not suited for particular environmental events. Conditions such as an earthquake fall into this category.

A specific event that invokes automatic protective action, such as reactor trip, SCS startup, RB Isolation, etc is referred to as a DBE. DBEs are representative of abnormal plant operation scenarios, each occurring frequently enough to exceed the “beyond design” cut-off of $1x10^{-5}$ occurrences per plant year. In the GT-MHR design, these events were separately classified as “safety-related” or “non-safety.” The RPS and IPS provide logic and overrides that interrupt normal control actions during a DBE. Tables 3-13 and 3-14 contain a list of the expected DBEs for the NGNP. In Table 3-13, the events are classified as “safety-related” because they lead to a control rod trip or isolation of the RB, which are both NGNP “safety-related” end actions. The events in Table 3-14 do not lead to “safety-related” end actions.

The Protection Systems incorporate setpoints, processed data, and single or multiple “Trip Request” pathways, plus end-action hardware to perform the necessary “System Trip” operations. A portion of the RPS and IPS hardware contains logic processors which provide outputs that ultimately initiate specific protection actions — this is called the Decision Logic. Previous MHR designs, including the GT-MHR design, used 2-out-of-4 protection logic for the RPS and IPS hardware to provide nuclear safety design redundancy and separation.

Table 3-13. NGNP Design Basis Events for Reactor Protection System

DBE Number	RPS Design Basis Events – Event Description
1	Rapid, sustained control rod withdrawal
2	Slow, sustained control rod withdrawal
3	Loss of PCS Precooler coolant flow
4	Loss of PCS Intercooler coolant flow
5	Turbomachine trip to non-motoring status
6	Loss of BOP heat rejection cooling water
7	Rapid leak of primary helium to Precooler water
8	Rapid leak of primary helium to Intercooler water
9	Rapid depressurization of primary helium to Reactor Building
10	Slow primary coolant leak to Reactor Building (TBD variations)

Table 3-14. NGNP Design Basis Events Requiring Automatic IPS Action

DBE Number	RPS Design Basis Events – Event Description
11	Loss of electric load external to NGNP plant
12	Rapid depressurization IHX secondary helium
13	Detection of Shutdown Cooling Heat Exchanger (SCHE) leak
14	Rapid increase in PCS helium pressure
15	SI Process upset
16	HTE Process upset
17	Loss of IHX Primary Circulator

3.10.2 Plant Control, Data, and Instrumentation System

The RPS and IPS also provide real-time status, warning, and alarm information to the PCDIS consoles and displays in the Control Room. Additionally, the PCDIS receives information regarding protection-events-in-progress to provide follow-up control action and real-time information to the plant operators. Since the PCDIS ultimately provides overall integration of all plant control and operation processes, the scope of the PCDIS design effort must include yet-to-be-developed top-level NGNP operational features as well as development of the Reactor Plant control, operation, and information functions. It is anticipated that, as in past programs, a Control Development Simulator (CDS) model will be developed and used to obtain RPS, IPS and PCDIS algorithm sets.

3.10.2.1 Plant Control Design

Plant control (PCDIS) design relies on a selection process using detailed computer-based simulation of control and plant features. The following control related features are of great importance to operation of the NGNP plant.

Reactor Power Control and Nuclear Instrumentation System. Reactor outlet temperature is either stabilized (held constant) or adjusted up or down by the PCDIS during many of the plant operations (e.g. startup, shutdown, electric power change or load loss, H_2 plant changes, etc) by interaction with the Control Rod Drive system. Also, reactor criticality and low level power control is achieved through control rod movement and use of Source Range nuclear instrumentation.

PCS Electric Power Generation and TM "Motoring" Control. The PCDIS uses the Bypass Control Valve System and the Inventory Control System to establish electric power output. To regulate TM speed, a Static Frequency Converter (SFC) and equipment for interaction with the electric power grid is provided. This equipment also provides the means to drive the electric generator as a motor during off-grid startup and shutdown operations. The PCDIS uses the generator "motoring" feature to establish self-sustaining flow and pressures in the TM during startup and to maintain TM flow below self-sustaining conditions during shutdown.

PCS TM Bypass Control Valve System. The PCDIS not only uses this system for TM control during startup and shutdown, but also during rapid (5% per minute) electric load changes (load reduction only). The Bypass System also provides TM overspeed protection in the event of electric load loss (DBE 11 in Table 3-13).

Helium Supply System Primary Helium Charging and Removal/Purification Inventory Control System. This system allows electric power output adjustment under fixed-speed TM operation without impairing NGNP power generating efficiency because inventory management allows primary helium mass flow rate to change without significant deviation from the ideal TM operating line. If bypass control is used as the means of adjusting electric power output instead, efficiency is greatly diminished as the electric output is reduced[3]. Of further note is the GT-MHR requirement that only purified helium should be stored during "inventory control" operation. While electric power output adjustments at a slow rate, such as 0.5% per minute, can be accomplished with inventory control, helium cannot be purified fast enough to reduce electric output at a fast rate, such as 5% per minute. This resulted in combined inventory/bypass control for rapid electric power reduction. However, short-term assistance from the bypass system does not significantly affect overall efficiency. And for an electric power increase, there is no comparative difficulty since stored helium can be returned rapidly to the PCS if re-injected at the compressor inlet. The NGNP Electric Plant needs both inventory control and bypass control features for electric load adjustment. The PCDIS operates these simultaneously to obtain the required electric power ramps or steps.

Helium Circulation Systems. The primary helium circulator allows control of a portion (approximately 11%) of the total reactor flow. It is anticipated that the Primary Helium Circulation System will include variable frequency speed control electronics, and that these features will control that portion of the reactor flow which is used by the IHX. The Secondary Helium Circulation System will similarly control secondary helium flow to manage temperature at the level needed for the hydrogen production plants. Figure 3-35 shows the top level temperature control scheme for the Reactor Plant as well as Electric Plant inventory and bypass control features. It is assumed that the GT-MHR inventory-bypass control-command scheme can be used for the NGNP Electric Plant. The NGNP Electric Plant includes Reactor Plant features, shown in Figure 3-35, along with necessary facilities such as waste-heat rejection, electric supply, electric power generation, etc contained in the BOP.

Hydrogen Production Plants Pressure and Flow Control Systems. The PCDIS will require specific control related functions in each of the Hydrogen Production Plants to adjust/balance flow rates and to adjust temperatures in tertiary flow systems. Real-time simulator analysis sequences will be completed to determine the exact nature of these functions.

[3] However, less efficient electric power generation might be acceptable for some dual-production operations not requiring formal demonstration of the optimized electric power production capability.

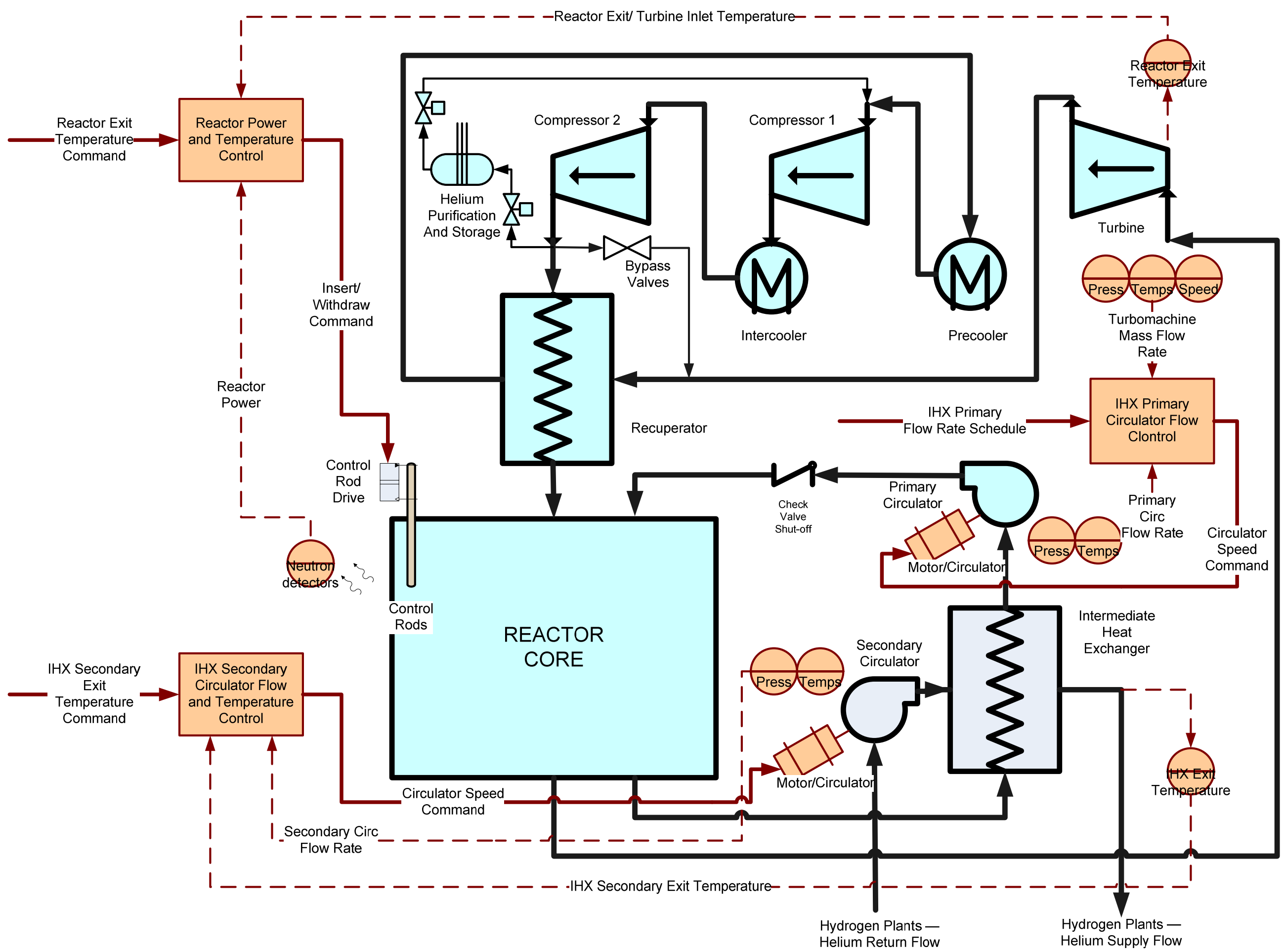

Figure 3-35. NGNP Reactor Plant Temperature Control

3.10.2.2 NGNP Startup

The plant control scheme developed for the GT-MHR provided a preliminary basis for the type of control actions depicted for NGNP in Figure 3-35. Since documentation describing GT-MHR analytical modeling, control algorithms and command parameters related to the electric production might also be useful to the NGNP control, one area explored in the GT-MHR design — plant startup — is discussed below.

A GT-MHR simulation run helped determine the essential operations involved in starting the GT-MHR electric plant from cold reactor conditions. It is anticipated that the NGNP can utilize essentially the same method, with the exception that IHX warm-up will have to be started near the end of the electric plant startup sequence. The GT-MHR startup sequence took approximately 12 hours, so it is likely that NGNP will require a longer period to bring all facilities to full operating condition. The steps identified in the GT-MHR sequence are listed below.

1) The Vessel System is pressurized to a low level (about 7% of operating inventory). The heat rejection system is started with cooling water flow through the Precooler and Intercooler.

2) The TM overspeed bypass valves are fully opened. (It appears that this is part of a strategy to minimize the generator "motoring mode" loads, and thus reduce the required Static Frequency Converter [SFC] power required for TM spin up and acceleration to operating speed.) The TM shaft is levitated and centered prior to the start of rotation and the SFC is then used to motor the TM up to operating speed[4].

3) The reactor starting conditions are zero power and zero decay heat with all rod banks inserted and the reactor subcritical. Control rod withdrawal is initiated after about 1 ½ hours in the GT-MHR simulation run. (This could be started sooner.) Nuclear power control is initiated at a very low power level (< 0.5%), and the inventory and reactor power are raised manually[5] in small increments and the bypass is gradually closed to about 15%. In about 6 ½ hours the inventory had been increased to 50%.

[4] As for the GT-MHR, a SFC will be needed to motor the NGNP TM during startup and shutdown. The required SFC power level varies, but should not exceed the 20 MW capability provided for in the GT-MHR design. Per OKBM, the SFC design power for the U.S./RUSSIAN International GT-MHR for w-Pu disposition is only 6 MW.

[5] Design of the startup automation features for the GT-MHR was never fully completed.

4) At about 7 ½ hours, with reactor power raised to approximately 20% for core heat up, and with reactor exit temperature transiently at roughly 350°C, the TM becomes capable of self-sustained operation at zero net power and is synchronized to the grid.

5) At about 9 ½ hours, after reactor temperatures stabilize, the automatic reactor exit temperature control loop is closed and the reactor exit temperature setpoint is advanced to 850°C at about 5 1/2° C per minute. This is actually accomplished by advancing the electric power setpoint to 50% because this automatically advances temperature at the 5.5°C per minute rate if it is low (as during startup). The bypass valve, which is automated as part of the electric output control (when inventory control is insufficient), then automatically closes as the operation ensues.

6) The approximate 50% operating conditions are reached in about 11 hours. The inventory control loop is then closed and the electric power setpoint is advanced from 50% to 100%. This allows inventory and reactor power to increase automatically in response to the electric power rate-of-advance schedule selected by the operator for the load-ramp up to the 100% condition. The GT-MHR plant is stable at 100% electric power output in about 12 hours.

In the NGNP startup operation, it is anticipated that a series of operations following step 6, above, will be developed to bring the hydrogen production plants on-line after reaching an "electric only" production level at about 89% reactor power. This is the "All Electric 1" mode shown in Table 3-15 below. This intermediate stage was excluded in the GT-MHR because the dual production objectives of the NGNP did not exist. It may also be necessary to include preliminary steps for IHX warm-up and secondary pressurization in earlier stages of the NGNP startup process as well.

3.10.2.3 NGNP Operating Modes

The anticipated NGNP operating modes shown in Table 3-15 should provide flexibility to demonstrate stand-alone electric production as well as the dual-mode NGNP production capabilities.

Table 3-15. NGNP Operating Modes

Production Mode	Reactor MW Electric	Reactor MW Hydrogen	Production Objective
All Electric 1	535	0	Achieve approximately 90% electric output capacity with hydrogen production or full electric options available. This mode is attained during NGNP startup and precedes NGNP shutdown.
All Electric 2	600	0	Achieve maximum electric output at 100% reactor power. This mode is used to demonstrate electric production capability such as load following to 50% output, step load change, etc.
Electric/ Hydrogen	535	65	Produce hydrogen and electricity. Maintain stable hydrogen production, but allow load change if necessary.

The Hydrogen Production capability will involve Reactor Plant control features to manage IHX heat transfer, secondary helium flow, and IHX temperatures. These features were, of course, not included in the GT-MHR electric plant. Figure 3-35 shows some of the new NGNP controls with those previously developed for the GT-MHR plant. Further explanation follows briefly below:

Reactor Power and Temperature Control. This previously developed control scheme is used for NGNP steady or transitory Reactor Exit temperature control. The control uses an outer temperature control loop, feeding an inner reactor flux control loop, and connected to a Control Rod Drive System. Control rod withdrawal/insertion sequencing is based on selective “one-at-a-time” rod withdrawal or insertion from predetermined control rod banks. Also, as in past designs, a non-linear configuration of the temperature control algorithm is applied through inclusion of total reactor mass flow rate to adjust for reactor core thermal “time-constant” variation over a wide range of reactor flow rate. This is based on the sum of the two primary flow measurements which are shown. This scheme allows consistent “tight” adjustment of reactor power through the operating range in spite of the large core thermal effects which are characteristic of HTGR reactors.

Primary Helium Circulator Flow Control. This control feature is added for NGNP. The control scheme will be developed on the basis of maintaining IHX primary flow in proportion to the reactor flow. Circulator Δp and speed measurements to obtain primary helium flow rate will need to be added to the instrumentation scheme to ensure that the total reactor flow rate measurement is available for control and protection.

Secondary Helium Circulator Flow and Temperature Control. Also an added control feature. This control will be used to balance flow rates, but its primary purpose will be to maintain secondary helium exit temperature at the level required for hydrogen production.

Helium Inventory and Bypass Valve Control. These are primary control features for the electric production plant. The Bypass Valves are the only means of arresting the TM speed transient following a load loss event.

None of the auxiliary system controls, including those for the hydrogen plants have been considered at this point. It is likely that the hydrogen production facilities will require automation features to assure compatibility with Reactor Plant operations and to deal with upset events where termination of the hydrogen production operations is required. Several types of Hydrogen Plant shutdowns are identified in Table 3-16.

Table 3-16. Hydrogen Plant Shutdown Type

No.	Shutdown Initiator	H_2 Production Status	Shutdown Type
1	Reactor Trip	Production temperature lowered immediately by Reactor Trip controls	Automatic
2	Normal Reactor Plant Shutdown	Production temperature will be lowered during shutdown process	Automatic with Operator Notice
3	Electric Load Loss	Production temperature can be maintained, but not reactor flow rate. Recovery possible, but Reactor Plant standby time at temperature is limited	Automatic
4	Multiple H_2 Plant Upset	Production temperature can be maintained, but maintaining H_2 Plant at standby condition may not be possible.	Automatic or Operator Decision
5	Single H_2 Plant Upset	Production temperature can be maintained	Operator Decision

3.10.2.2.3 NGNP Plant Interactions

Table 3-16 also identifies areas of plant interaction to be considered in the next stage of the control development effort. It will be necessary to develop the plant simulation capabilities mentioned previously in order to identify interactions which require automatic overrides or shutdown controls.

Important plant interactions that should be addressed in the preliminary control development analysis efforts are explained as follows:

- Reactor trip (initiated by the RPS) will immediately initiate the Electric plant shutdown sequence. A major objective in the control response sequence following a control rod trip is to lower the reactor exit temperature as rapidly as possible. This is to protect PCS components. For this reason the Electric Plant remains connected to the grid for two minutes to allow TM flow rates to remain high and to maintain helium exiting the turbine at normal temperature levels during the early part of the shutdown transient while the reactor is cooling. In the GT-MHR, Reactor Exit temperature was reduced about 200°C within the first 2 ½ minutes following a reactor trip, and was reduced from 850°C to about 540°C within 15 minutes after the trip. It is anticipated that Hydrogen Plant processes will have to be stopped in a similar rapid-response fashion.

- Overspeed protection following loss of the Electric Plant grid connection (initiated by the Investment Protection System) requires diversion of a large part of the reactor flow. Reactor flow rate drops to about 70% within 10 seconds following loss of the grid connection. While the bypass control scheme for Electric Plant can recover and maintain the TM speed, it is undesirable to stay at high reactor power and temperature because the large heat rejection loads through the Precooler and Intercooler pose limiting requirements for this event. It is anticipated that the Hydrogen Plant processes will have to be shutdown immediately in this event. But, it is also possible that protection action to isolate the IHX will be required to assure that reactor flow rates stay as high as possible. Since reactor pressure also drops about 10%, it may be necessary to operate the IHX Pressure Balance System[6] during this transient. The GT-MHR scheme to reduce inventory and temperature for Electric Plant standby in this event will also have to be re-evaluated.

- Hydrogen Plant multiple upset events pose a new category of Design Basis Events (DBEs). A loss of the secondary helium circulator is one new DBE that would cause a multiple Hydrogen Plant upset. Protective action may be required because of a potential

[6] Table 3.10-4, end-action 7 shows the IHX Pressure Balance System. Its purpose is to reduce high-temperature IHX pressure loads.

36° C rise in reactor inlet temperature following this event. Although the Reactor Exit Temperature control system reduces exit temperature following a rise of reactor inlet temperature, a thorough evaluation of this scenario should be completed to assure that Electric Plant stability can be retained. An important consideration is that Reactor Exit temperature (aka Turbine Inlet temperature) is a reactor trip parameter.

- Hydrogen Plant single upset events must also be included as a new category of Design Basis Events (DBEs). For example, a disruption of the heat utilization process in one of the Hydrogen Plants might in turn cause a proportional reduction in secondary helium flow. However, this should not prevent continued operation of the other Hydrogen Plant or the Electric Plant. The design of the “IHX Secondary Circulator Flow and Temperature Control,” shown in Figure 3-35, might include adjustment of the “IHX Primary Flow Rate Schedule” to stabilize the remaining Hydrogen Production plant following this event.

3.11 Balance of Plant and Auxiliary Systems

No work was performed on design of the NGNP Nuclear Plant BOP systems as part of preconceptual design studies. Consequently, in developing the NGNP cost estimate, it was assumed that the NGNP BOP systems would be essentially identical to those for the GT-MHR and that the capital costs would be the same as those developed by GA for a one-module GT-MHR prototype plant. Summary level descriptions for the Nuclear Plant BOP systems are provided in [PCDSR 2007]. A more detailed description of these systems can be found in [Shenoy 1996].

- Waste Heat Rejection System
- Spent Fuel Cooling System
- Nuclear Island Cooling System
- Essential Plant AC Electrical System
- Essential Plant DC Electrical System
- Nuclear Island HVAC System
- BOP HVAC System
- Power Conversion Component Handling System
- Radioactive Waste and Decontamination System
- Balance of Hydrogen Plant

4. BUILDINGS AND STRUCTURES

The plant layout, shown in Figure 2-3 (in Section 2) consists of the RB, the two hydrogen production plants, the Reactor Service Building (RSB), Operations Center (OC), heat transport pipes, and other buildings and structures that provide various supporting functions for the overall complex. Systems containing radionuclides and safety-related systems are located in the Nuclear Island (NI) area, which is separated physically and functionally from the remainder of the plant.

4.1 Reactor Building

The RB for the NGNP 600-MWt reactor is classified as a vented low-pressure containment (VLPC). The RB consists of a below-grade multi-celled, embedded structure and the RCCS inlet/outlet structures, both of which are constructed of cast-in-place reinforced concrete. The degree of embedment was selected to serve a number of objectives, including reduced cost and complexity of construction, ease of operation, minimization of shielding, and good seismic performance. The operating floor of the plant is set at site grade, with a maintenance enclosure covering the operating area, which is traversed by refueling equipment. Figure 4-1 shows the RB and the above-grade maintenance enclosure.

There are two floors below grade with a rectangular footprint which are used to house mechanical, electrical, and instrumentation systems. Below elevation -30 ft, the RB is configured as a cylinder to enable it to resist soil and groundwater pressure. This portion of the RB is called the silo. The Reactor System, Vessel System, and PCS are located within this space as shown in Figures 4-1 and 4-2. The RV, PCS vessel, and IHX vessel are housed within separate concrete compartments of roughly equal dimensions as shown in Figure 4-3. The IHX is expected to be only about 3.8 m in diameter and 16 m high, but the compartment for the IHX is about the same size as the compartment for the PCS. This large cavity for the IHX allows for use of a much larger heat exchanger should this become desirable or necessary. The reactor core and IHX are connected by a cross vessel that is tapered to allow for IHX size adaptation while maintaining vessel integrity at the reactor nozzle end.

The length and diameter of the PCS vessel control the dimensions of the silo. The silo depth must also accommodate the machinery used to service the shutdown cooling circulator and heat exchanger. Access to and from the cylindrical portion of the building for piping, electrical services, personnel, and the concentric RCCS ducting is made from the rectangular portion of the building between elevations -30 ft and grade.

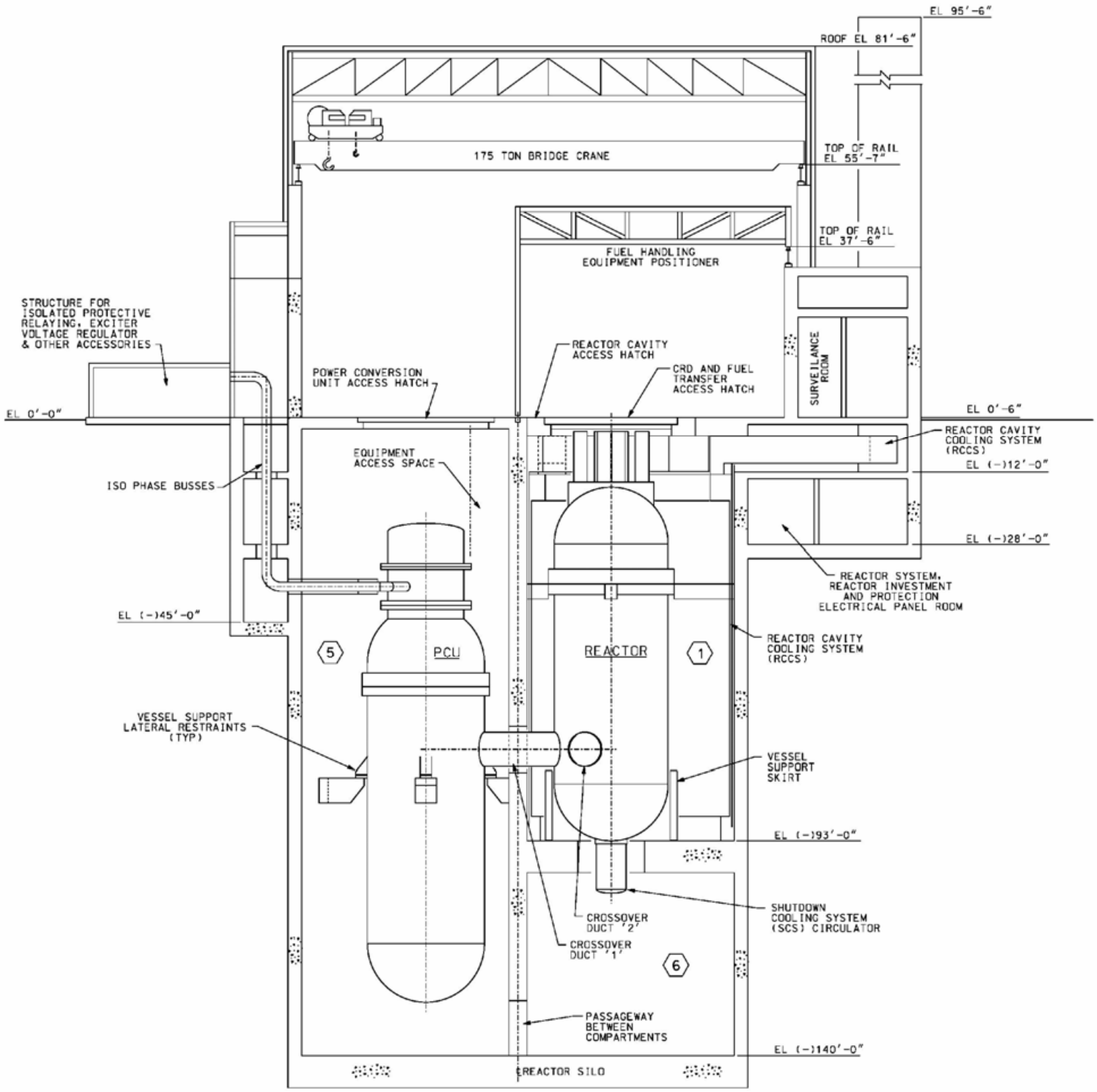

Figure 4-1. Elevation View of Reactor Building – Section A-A

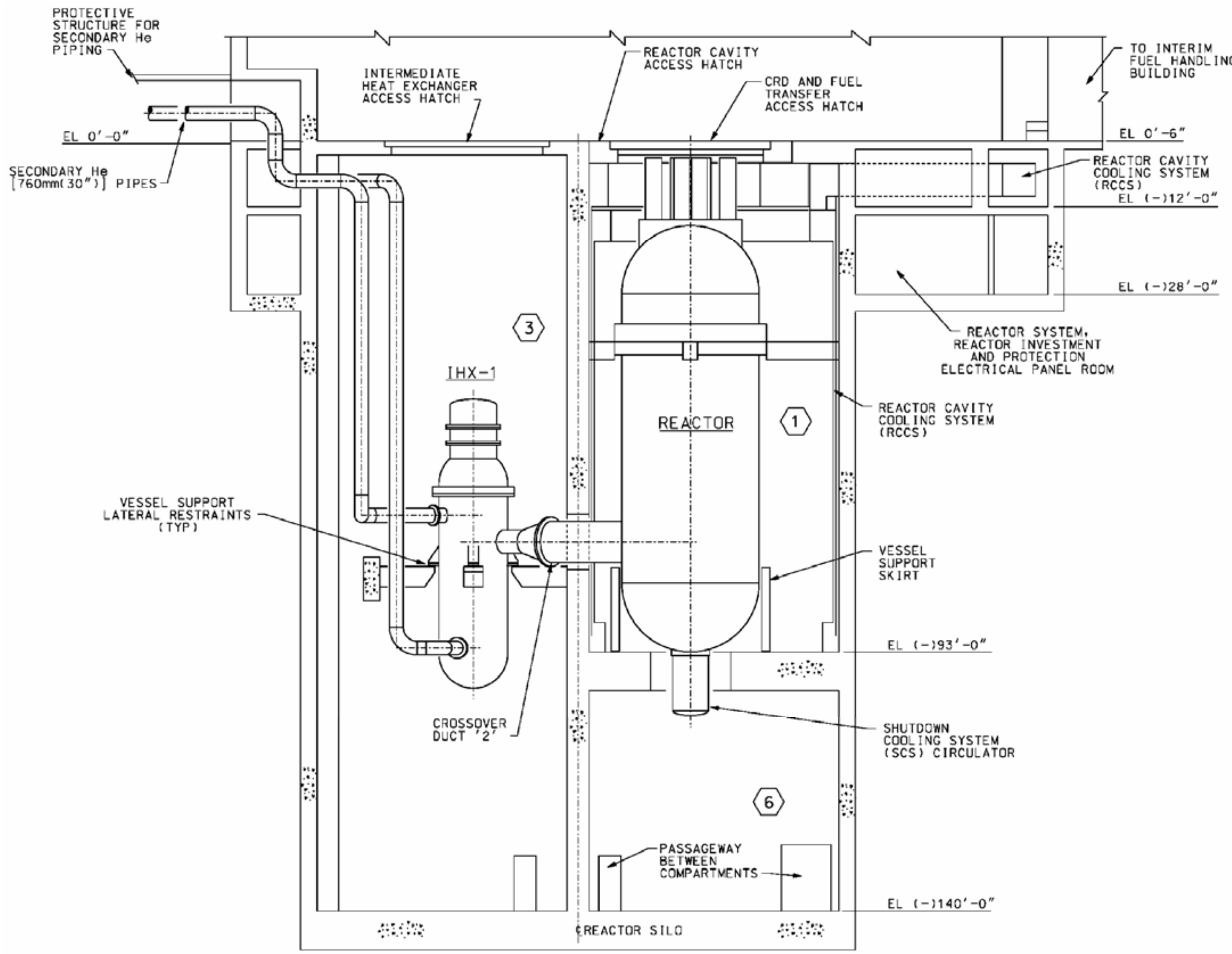

Figure 4-2. Elevation View of Reactor Building – Section B-B

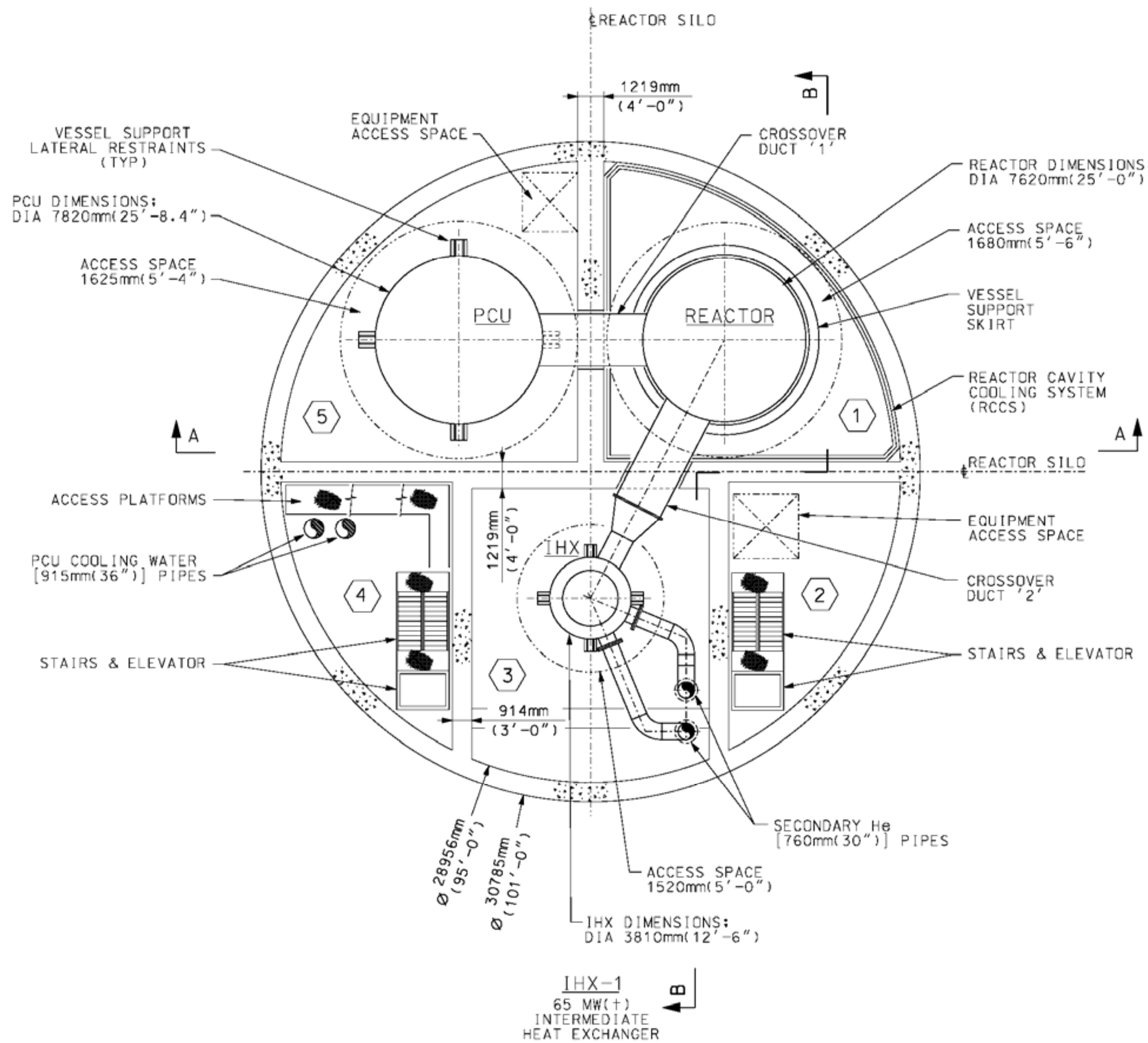

Figure 4-3. Plan View of Reactor Building – Elevation -93' –0"

The maintenance enclosure is a 29-m (95-ft) high rectangular steel framed structure that spans the area above the below-grade RB. Access for refueling and for major maintenance activities is from this operating floor. There are two extensions of the reinforced concrete RB above grade. To one side of the RB, the reinforced concrete portion of the building extends to elevation +95 ft 6 in. (29 m) to serve as the RCCS inlet-outlet structure. The above-grade area, called the Maintenance Enclosure, houses the part of the Fuel Handling System that transports the fuel and reflector elements between the receiving facility and the reactor core, the helium transfer and circulation system, and various piping and electrical equipment.

The RCCS panels, at the location where they enter the closed portion of the RB, are regarded as part of the VLPC boundary. In essence, air flowing inside the RCCS ducts and panels is outside the containment boundary. The walls, doors, plugs, and other barriers which separate the closed, recirculated portion of the building from the once-through cooled portion of the building or from the outside environment (including the RCCS panels and ducts) constitute the fourth containment barrier. Leakage from within this portion of the RB to the other part of the RB or to the environment has the potential to transport fission products from the containment to the environment. This space is also the portion of the RB that is affected by the specified building leak rate. It is expected that essentially none of the leakage which occurs will be from the surfaces of the building which are in contact with the soil, and that the specified leak rate represents an upper bound on the exchange which could occur between the building interior and the environment, since the pressure (and therefore the leakage) will normally decrease over the course of an accident.

4.2 Reactor Service Building

The Reactor Service Building (RSB) is a three-story reinforced concrete structure at grade level next to the RB. The fuel handling area is located within the RSB. This area includes facilities for introducing new fuel, for loading and shipping spent fuel casks, for storing new fuel, and for inspecting new and spent fuel. The Helium Services System, which includes the helium purification system and the helium transfer and storage system, is also located in the RSB.

The Hot Service Facility is located inside a shielded vault in the RSB adjacent to the fuel sealing and inspection facility. The Hot Service Facility is used for inspection, maintenance, and repair of reactor service equipment and tools. The facility includes viewing windows, operating galleries outside the vault, manipulators to perform the inspection, maintenance, and repair services, as well as portable decontamination equipment.

4.3 Operations Center

The OC building is a steel framed structure founded on grade beams and individual footings. The Operations Center (OC) houses:

- The plant security access and egress area,
- Security administration,
- Plant operation offices and engineering space,
- Primary and secondary alarm stations,
- Training rooms, conference rooms, and lunch areas.

The total floor area of the OC is approximately 56,000 ft^2. It is two stories high, with a 2,000 ft^2 basement. The above-ground area houses the security, engineering, and administration functions. The basement houses the central alarm station.

The ground floor of the OC contains plant access and egress, security administration, the secondary alarm station, the electronic equipment room, lunch room, first aid suite, classroom and training area, a mechanical equipment room, and an emergency electrical power source room. Plant access and egress areas contain the inspection, detection, and access control into the vital and nonvital areas of the plant. Physical protection from acts of sabotage against the plant access and egress areas will satisfy Federal Regulations.

The second floor contains the plant administration areas for the operation, maintenance, and technical divisions, the control room, and a mechanical equipment room. The administration areas will provide office space, conference rooms, a reception and waiting area, storage space, and an engineering office area.

4.4 Hydrogen Plant Area

For the preconceptual design stage, the following conditions are specified for the area that contains the SI-based and HTE-based hydrogen production facilities:

- The production plants will be located a minimum of 90 m (~300 ft) away from the nearest boundary of the RB in order to preclude damage to the building as a result of a hydrogen plant accident.
- The plants will be open to the environment, i.e., buildings are not provided to enclose the equipment. A perimeter fence will enclose the area.
- A 2 ft high earthen berm will be provided along the perimeter of the production facilities' area to contain any potential spills. Because of the separation distance between the

hydrogen plants and the RB and because of the underground location of the NGNP MHR, added protection for the MHR against hydrogen plant accidents (e.g., a high earthen berm or other blast containment structure) are not included.

- One 2.1 m (7 ft) diameter underground Hydrogen Storage Tank is included in the preconceptual design. At a pressure of 450 bars, this tank will be capable of storing approximately 100 kg of hydrogen. A truck loading area is also included in the preconceptual design.
- Extended areas as shown in Figure 2-3 (in Section 2) are included in the overall plant layout to accommodate potential expansion of the hydrogen plants and the hydrogen storage area.

4.5 High-Temperature Helium Transfer System Pipes

Heated helium from the IHX will be routed via pipes from the RB to the hydrogen plants and back. The helium supply and return lines will run parallel to each other. They will be supported on regularly spaced concrete piers and will be provided with adequate thermal expansion loops. Because of the high temperature of the external surfaces of these pipes while in service, a protective roof will be placed above the pipes to prevent excessive exposure to rainfall (to reduce evaporation and consequent heat loss). The roof will be supported on regularly spaced metal columns and will be vented for heat relief (see Figure 2-3, Section A-A). As a safety precaution, the protective enclosure will feature a perimeter fence and bird screens.

4.6 Other Facilities

There are several facilities that provide important support functions for the overall NGNP complex. These include:

- Personnel Services Building
- Radioactive Waste Management Building
- Spent Fuel Storage Building
- Helium Storage Structure
- Auxiliary Building
- Nuclear Island Warehouse and Turbo-Machinery Maintenance Facility
- Fire Protection Services Buildings and Structures
- Water Treatment Building
- Standby Power System Building
- Remote Shutdown Building

5. PLANT ASSESSMENTS

As part of this pre-conceptual design study, assessments of the NGNP were performed in the areas of safety, licensing, and economics.

5.1 Safety Assessment

The following sections describe the safety features of the NGNP and assessments of bounding accidents involving loss of flow and loss of coolant.

5.1.1 Key Inherent Safety Features and Design Provisions

Passive safety features of the MHR concept include the (1) ceramic, coated-particle fuel that maintains its integrity at high temperatures during normal operation and loss of cooling events; (2) an annular graphite core with high heat capacity and a low power density that limits the temperature rise during loss of cooling; (3) a relatively low power density that helps to maintain acceptable temperatures during normal operation and accidents; (4) helium coolant that is inert, remains single phase, and is neutronically transparent; and (5) a negative temperature coefficient of reactivity that ensures control of the reactor for all credible reactivity insertion and loss-of-coolant events. These features assure sufficient decay heat removal to an ultimate heat sink by the natural processes of radiation, conduction, and convection, to preclude any significant particle coating failure or radionuclide release under all conditions of loss of forced cooling or loss of coolant pressure. The fuel, the graphite, the primary coolant pressure boundary, and the low-pressure vented containment building provide multiple barriers to the release of fission products.

In the design of the NGNP, the desirable inherent characteristics of the helium coolant, graphite core, and coated fuel particles are supplemented with specific design features to ensure passive safety. The release of large quantities of radionuclides is essentially precluded by the fuel particle ceramic coatings, which are designed to retain nearly all fission products during normal operation and to remain essentially intact during licensing basis events. The integrity of the particle coatings as a barrier is maintained by limiting heat generation, assuring means of heat removal and by limiting the potential effect of air and water ingress on the particles under all potential accident conditions. These characteristics tend to dominate the safety of the plant as a whole and serve to prevent and mitigate accidents. In particular, these characteristics, supported by safety system design, are effective in retaining radionuclides at the source within the coated fuel particles. Containing radionuclides at the source reduces all risks, including health and safety risks, environmental risks, and risks that operation will be interrupted by a release and lengthy recovery time.

Although the fuel particle ceramic coatings are the most important barrier to the release of fission products, there are actually five principal fission product barriers in the NGNP as shown in Figure 5-1. Three of the barriers are pressure-retaining barriers (e.g., the fuel particle coatings, the primary coolant pressure boundary, and the vented low-pressure containment building) that are capable of retaining radionuclides. The other two barriers are the fuel kernels and graphite structural elements; these barriers provide effective retention of some radionuclides.

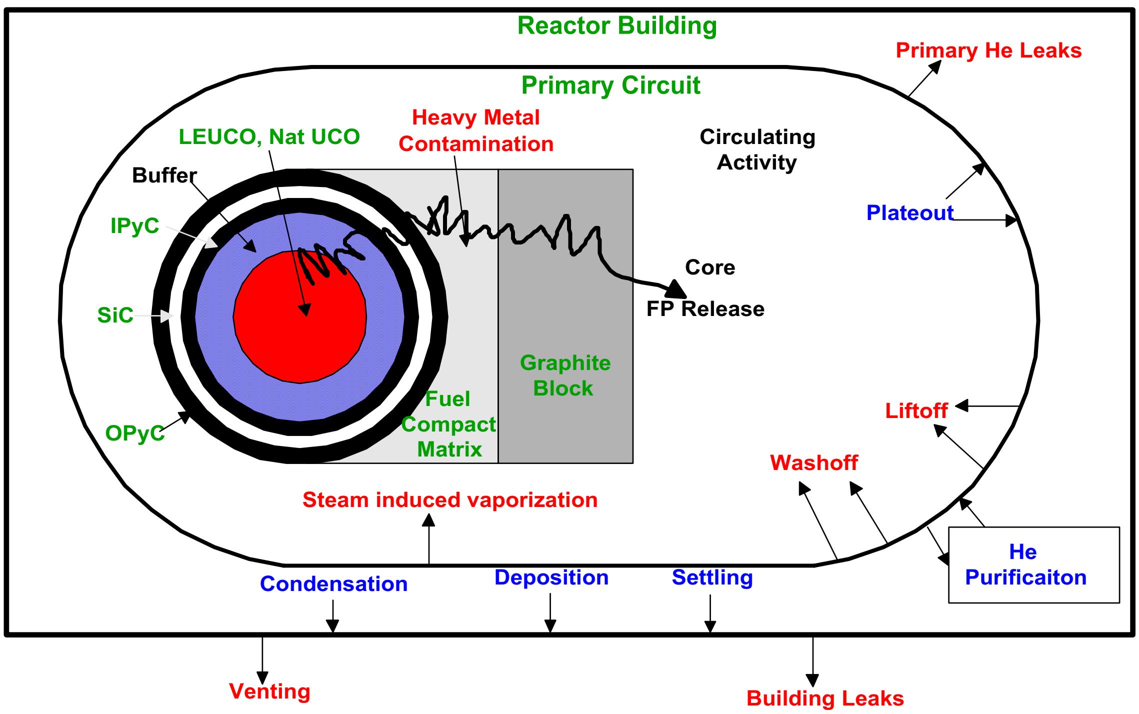

Figure 5-1. MHR Radionuclide Containment System

Kernels. The first barrier to fission product release from the fuel is the kernel itself. The fuel kernels retain a significant fraction of the radiologically important, short-lived fission gases such as Kr-88 and I-131. However, the effectiveness of the kernel for retaining gases can be reduced if exposed kernels are hydrolyzed by reaction with trace amounts of water vapor. The fuel kernels will also retain long-lived, volatile fission metals such as Cs, Ag, and Sr, depending upon temperature and burnup.

Particle Coatings. The second and most important barrier to radionuclide release is the fuel particle coatings. The coatings provide a high-integrity pressure vessel which is extremely retentive of radionuclides. The layers of the TRISO-coated fuel particles have specialized purposes. The purpose of the buffer layer (low density carbon) is to provide a reservoir for fission gases released from the kernel and to attenuate fission recoils (fuel particles only). The most important coating is the silicon carbide (SiC), which provides most of the structural strength and dimensional stability. In the fuel particle, it serves as the primary barrier to the release of fission products, particularly metallic fission products, because of their low solubilities and diffusion coefficients.

Graphite. The carbonaceous fuel compact matrix materials and the core structural graphite collectively are the third release barrier. Core graphite is highly retentive of some fission products (i.e., Sr, Rb, Cs, rare earths), but is virtually nonretentive to others (i.e., noble gases). For example, under typical core conditions, the fuel element graphite attenuates the release of Cs and Ag from the core by more than an order of magnitude, and Sr is essentially completely retained.

Primary Coolant Pressure Boundary. The fourth release barrier is the primary coolant pressure boundary. This barrier is provided by the steel pressure vessels, which will be designed and constructed to ASME Section III Division 1 requirements. The chemically inert helium coolant minimizes corrosion and eliminates the need for the complications of steel internal cladding. The entire reactor module is protected by the underground RB from external events and is conservatively designed to accommodate internal events. The helium purification train is very effective at removing long-lived fission gases and contaminates from the primary coolant. However, for short-lived fission gases, the dominant removal mechanism is radioactive decay, and for the condensable fission products, the dominant removal mechanism is deposition, or plateout, on the various helium-wetted surfaces in the primary circuit.

Containment. The reinforced concrete, vented low-pressure containment is the fifth barrier to the release of radionuclides. It is a normally closed space, located below grade. It is equipped with a vent that opens if the pressure inside the containment exceeds its design set point, releasing mass and energy associated with a blow down and protecting the integrity of the building and the RCCS. Even if the vent opens, natural removal mechanisms (including radioactive decay, condensation, fallout, and plateout) reduce the concentration of radionuclides in the containment atmosphere, reducing the offsite releases. While the vent allows the release of radionuclides released promptly, the release of associated gases early in the event eliminates the driving pressure that could transport the delayed source term out of the building. After release of the initial blow-down energy pulse, the vent is designed to close for containment of radionuclides that might diffuse out of the fuel during time-at-temperature conditions. Robust design features protect the containment function from degradation by external events. Inclusion of a broad spectrum of DBEs protects the containment function from damage by internal events

5.1.2 Safety Related Systems, Structures, and Components

On the basis of prior safety assessments, the major systems, structures, and components (SSCs) that are relied upon to perform one or more safety functions (e.g., ensuring safe shutdown and protection of the primary coolant pressure boundary), or are otherwise relied upon to meet the dose criteria at the site boundary are as follows:

- Reactor System including neutron control assemblies, ex-vessel neutron detectors, the reactor internals, reactor core, and fuel.
- Vessel System including the ASME Section III vessels and pressure relief.
- RCCS including the entire system as required for removal of residual heat.
- Reactor Protection System (RPS) including all sensors, control logic, and housings supporting safety trips.
- Fuel storage pools and wells which are part of the Reactor Service Building.
- Essential AC and DC power systems.

These safety systems have been provided to mitigate the consequences of all design basis accidents and to protect the five barriers to the release of radionuclides. Some of these systems act to protect the fuel particles; some protect the primary coolant pressure boundary; some protect the containment; and some protect several barriers at the same time. Maintaining barrier integrity constitutes the NGNP safety function; accident prevention and mitigation is the process by which these functions are accomplished. Consistent with the simple, yet robust, safety design approach of the GT-MHR, only this relatively modest number of SSCs is seen as being important for ensuring public health and safety. Equally important, this equipment can be seen to reflect the utilization of passive features.

5.1.3 Accident/Transient Analysis

The bounding design basis events (DBEs) for the NGNP will be a loss of flow leading to a high pressure conduction cooldown (HPCC) and loss of coolant leading to a low pressure conduction cooldown (LPCC). The HPCC event is typically initiated by trip of the PCS. The RPS automatically initiates a reactor trip on low flow or turbomachine trip. Because the system remains at high pressure, the decay heat is more uniformly distributed within the core and vessel than during a LPCC event. The LPCC event is typically initiated by a small primary coolant leak, causing the system to depressurize to atmospheric pressure. The RPS automatically initiates a reactor trip on low coolant pressure. For both events, the SCS fails to start and decay heat is removed by thermal radiation and natural convection from the RV to the RCCS.

These events have been analyzed in detail for a MHR operating with a reactor outlet coolant temperature of 950°C, and the results show that peak fuel temperatures remain below the design goal of 1600°C, and the temperatures for the vessel and other safety-related SSCs also remain below acceptable limits. For the LPCC event, the peak fuel temperature is 1525°C and occurs about 60 hours following initiation of the event. For the HPCC event, the peak fuel temperature is 1349°C and occurs about 50 hours following initiation of the event. The calculated peak vessel temperatures for the HPCC and LPCC events are approximately 478°C and 517°C, respectively. For both events, the peak vessel temperatures occurred about 72 hours following initiation of the event.

5.2 Licensing Strategy

All nuclear power plant applications in the United States require a safety review, an environmental review, and an antitrust review by the NRC. The NGNP is subject to Title 10, "Energy," of the Code of Federal Regulations and those regulations applicable to a Class 103 Commercial Power Reactor, as defined in 10CFR50.22. The regulations and licensing options that are potentially available for licensing the NGNP include 10CFR50, 10CFR52, 10CFR53 and "License by Test." The licensing approach for the NGNP is expected to reflect existing regulatory regulations and guidance, deterministic safety criteria, and risk-informed evaluations.

10CFR50, “Domestic Licensing of Production and Utilization Facilities” is the two-step licensing process used by currently operating commercial nuclear power plants. The 10CFR50 licensing process has been in use for more than forty years and is well understood. Thus, the risks associated with licensing under 10CFR50 are known. This process requires both a construction permit and an operating license. NGNP licensing using this process will require a Preliminary Safety Analysis Report, a Final Safety Analysis Report and the supporting Environmental Report and Environmental Impact Statement. The 10CFR50 licensing process supports plant design and construction as parallel activities. Taking advantage of the ability to start construction in parallel with design evolution has potential schedule advantages, but also involves some risks.

Implementation of a 10CFR52 licensing process for the NGNP would be problematic given that the plant design will lack the design maturity needed to support the required Combined Operating License Application (COLA). The project schedule does not support development of a detailed COLA. In addition, there is considerable uncertainty associated with the 10CFR52 process.

10CFR53 is not a viable option for the NGNP because two major activities yet to be developed by the NRC in support of Part 53 include development of the technical basis for 10CFR53 and rulemaking development of the regulations and associated guidance.

A “license by test” philosophy for the NGNP has been discussed in a variety of forums including meetings of the Advisory Committee on Reactor Safeguards. The discussions have centered on building a full-size demonstration facility and performing a series of tests to identify the dominant risk contributors for the facility and to preclude extra features in the design that do not provide additional margin of safety. The data from the testing would by used to certify the design. However, no current regulatory framework exists and no regulatory framework has been formally proposed by the NRC for “license by test”.

Besides the absence of a licensing framework, a "license by test" approach would be a high-risk option. Testing could severely stress structures, systems and components (SSCs) necessitating repair, supplemental analysis, reductions in qualified life, and possible component replacements. This could adversely affect the ability of the facility to achieve its long-term mission of 30-years operation, and the potential loss of availability and additional operational costs could significantly impact investment. The ability to secure financial backing could also be adversely affected given the implications of negative testing results on an essentially completed plant.

Based upon the above factors, a "license by test" approach does not appear to be viable for obtaining a NRC license for the NGNP demonstration facility. However, while testing alone will not be sufficient for facility licensing, testing will undoubtedly be a very important constituent of the NGNP licensing process. Testing can be used to validate many of the analytical results presented in the SAR regardless of the licensing process used. One key area of concern is whether full fuel qualification can be achieved in time to support the planned NGNP operation date and so fuel qualification may be a candidate for licensing by test.

In conclusion, following the 10CFR50 licensing process for the NGNP is the most prudent approach at this time. The 10CFR50 licensing process supports plant design and construction as parallel activities. Taking advantage of the ability to start construction in parallel with design evolution has schedule advantages and a chance to manage the better understood financial risk. Also, the 10CFR52 (or perhaps 10CFR53) licensing documents needed to support NGNP-based follow-on commercial plants should be developed based upon the NGNP 10CFR50 Operating License (OL) phase documents including the Probabilistic Risk Assessment as approved by the NRC. This will facilitate the submittal and approval of future commercial plant license applications. Under either 10CFR50 or 10CFR52, extensive pre-construction permit application interaction with the NRC will be necessary to apprise the NRC staff and to better define the acceptance criteria for licensing the NGNP.

In addition to NRC licensing requirements, environmental permitting will also be required. Permits will be required from the US Environmental Protection Agency, the Idaho Department of Environmental Quality, the US Army Corps of Engineers, and other cooperating local agencies.

5.2.2 Preliminary Hydrogen Plant Hazards Assessment

A preliminary hazards assessment (PHA) for the NGNP prototype SI-Hydrogen plant was performed. Both the SI-based plant and HTE-based plant have hazards associated with hydrogen, electricity, and high-temperature heat, but the hazards for the SI-based plant are expected to be more bounding because of the chemicals involved.

The results of the NGNP SI-hydrogen plant PHA are typical for a modern chemical plant built in the United States. The unit operations in the hydrogen plant (distillation columns, chemical reactors, heat exchangers, etc.) are standard chemical processes with mature technology that will be extensively tested prior to deployment in the NGNP. There is no currently anticipated inherent excessive risk in the thermochemical production of hydrogen that would preclude licensing of the NGNP or commercial-scale hydrogen productions plants based on the processes demonstrated in the NGNP.

An attractive feature of the GT-MHR plant for electricity production is siting flexibility, because no plan for public evacuation is required as the result of the MHR's passive-safety features. For a commercial-scale H2-MHR, a potential issue that requires further evaluation is whether or not a public evacuation plan is required because of potential accidents that could cause chemical releases from the SI-hydrogen plant. However, chemical releases should not impact the passive safety of the reactor system.

5.3 NGNP Cost Estimates

5.3.1 Capital Costs

An NGNP capital cost estimate was prepared using the Generation IV International Forum Code of Accounts [GIF 2006] in organizing the capital costs. The estimate was based on prior capital cost estimates for the GT-MHR as supplemented by hydrogen plant cost estimates developed by the GA Team. Table 5-1 summarizes the NGNP capital costs. The groundrules used in developing the cost estimate are given in [PCSDR 2007].

Table 5-1. Summary of NGNP Capital Costs

	GIF COA		2007$ (In 1,000s)
CAPITALIZED PRE-CONSTRUCTION COSTS	**1**	**CPC**	**117,850**
Base Cost	11 – 18		117,850
CAPITALIZED DIRECT COSTS	**2**	**CDC**	**837,447**
Base Cost w/o Initial Fuel Core	21 – 28		703,447
Initial Fuel Core Load			134,000
DIRECT CONSTRUCTION COSTS	**1 + 2**	**DCC**	**955,297**
CAPITALIZED INDIRECT SERVICES COST	**3**	**CIC**	**1,759,873**
FIELD INDIRECT COSTS	**31-34**	**FIC**	**193,003**
Temporary Construction Facilities			74,651
Construction Tools and Equipment			42,475
Payroll Insurance and Taxes			50,929
Permits, Insurance & Local Taxes			1,866
Plant Startup and Test			23,082
TOTAL FIELD COST	**10 - 34**	**TFC**	**1,148,300**
FIELD MANAGEMENT COST	**35 - 38**	**FMC**	**1,566,870**
R&D for Design			492,000
Conceptual Design			139,000
Preliminary Design			279,000
Final Design			593,000
Field Office Expenses			12,558
Field Job Supervision			46,927
Field Quality Assurance			4,385
BASE CONSTRUCTION COST	**1 + 2 + 3**	**BCC**	**2,715,170**
CAPITALIZED OWNER COST	**4**	**COC**	**82,170**
Project Management Expenses			19,226
Staff Training and Administration			43,993
General and Administrative			18,951
CAPITALIZED SUPPLEMENTARY COSTS	**5**	**CSC**	**78,829**
Fees, Taxes and Insurance			16,373
Spare Parts & Capital Equipment			62,456
OVERNIGHT CONSTRUCTION COSTS	**1 + 2 + 3 + 4 + 5**	**OCC**	**2,876,169**
CAPITALIZED FINANCIAL COST (Esc., Fees & IDC)	**6**	**CFC**	**---**
CONTINGENCY (20%)			**575,234**
TOTAL CAPITAL INVESTMENT COST	**1+2+3+4+5+6+ Contingency**	**TCIC**	**3,451,403**

5.3.2 Operating Costs for 30-year Period

The 30-year NGNP operating costs are estimated to consist of (1) operations and maintenance (O&M) costs, (2) nuclear fuel costs, and (3) decommissioning costs. The total estimated NGNP 30-year operating costs in constant 2007$ is $2,975. Each of the 30-year cost components is the product of cost per year in 2007$ times 30 years. The as-spent (or nominal $) 30 year costs would be the summation of the nominal cost for each of the years where the nominal cost for each year is the 2007$ cost per year times the cumulative inflation rate for each year. The methodology used in developing the NGNP operating cost estimate are presented in [PCDSR 2007].

5.4 Economic Assessment for Commercialization

Two commercial nuclear hydrogen plant variations were evaluated with respect to their hydrogen production cost versus a projection of the future market value of hydrogen. The two plant variations include:

1) An nth-of-a-kind nuclear hydrogen production plant consisting of two 600-MWt MHR modules providing process heat to a SI-based hydrogen production plant and two 600-MWt MHR modules dedicated to electricity production to provide the electric power needed by the SI-based hydrogen production plant

2) An nth-of-a-kind HTE-based nuclear hydrogen production plant consisting of four 600-MWt MHR modules providing both process heat and electricity to the HTE-based hydrogen production plant having 292 H2 production units each consisting of eight modules of planar SOE cells

The commercial assessment for each of the plant variations involved development of a capital cost estimate and an operating cost estimate (including O&M, fuel, and decommissioning costs) and an estimate of the amount of hydrogen produced by the plant in order to calculate the unit cost of hydrogen production. This unit cost was then compared against the projected market value of hydrogen that was estimated as part of the NGNP end-products study [Hanson 2007a]. The groundrules and methodology used to develop the capital cost and operating cost estimates and the amount of hydrogen produced are by the plants is presented in detail in [PCDSR 2007].

Table 5-2 summarizes the result of the commercial assessment. The overall hydrogen production cost in the SI-based plant and the HTE-based plant were estimated to be about 2.26

$/kg and 2.22 $/kg, respectively. In both plants the hydrogen production cost is about 10% below the projected market value of hydrogen.

Table 5-2. Commercial Plant Hydrogen Production Costs vs. Hydrogen Market Value

	NOAK SI-H2-MHR Commercial Plant		SI-MHR $/kg Delta @ 2.5 $/kg	NOAK HTE-H2-MHR Commercial Plant		HTE-MHR $/kg Delta @ 2.5 $/kg
	Credit $/kg	Prod. Cost $/kg		Credit $/kg	Prod. Cost $/kg	
H2 production cost w/o credits	NA	3.14	-0.64	NA	2.40	0.10
Electricity Credit @ 106 mil/kWh	0.70	---	---	None	None	None
H2 cost with electricity credits	NA	2.44	0.06	NA	2.40	0.10
O2 Credit @ 23 $/Tonne	0.18	---	---	0.18	---	---
H2 cost with electricity and O2 credits	NA	2.26	0.24	NA	2.22	0.28
NOTE: Delta = 2.5 $/Kg minus the H2 Production Cost in $/Kg						

Sensitivity Analysis for SI-H2-MHR Hydrogen Production Cost

Sensitivity analyses were performed for the SI-H2-MHR model to determine the impact of three factors on the production cost of hydrogen:

- Process risk in terms of the technology maturity of the hydrogen production portion of the plant
- Process efficiency
- Construction time in months

Table 5-3 provides the results of the sensitivity analyses.

Technical maturity was addressed by increasing the contingency applied to the capital cost component of the production cost from 5% to 20% to reflect impact of process risk. This resulted in an increase of 0.30 $/kg in the H2 production cost, or about 13.3%.

Process efficiency was addressed by assessing the effect of targeting a process efficiency of 49% as opposed to the current 45%. This can be achieved by reducing the electric power consumed by the SI process equipment. 49% efficiency can be achieved if the power consumption is reduced by 89 MWe. This also allows the plant to make an additional 89 MWe available for sale on the grid. The additional power sales reduces the H2 production cost by 0.41 $/kg, or about 18.1%.

Table 5-3. SI-H2-MHR H2 Production Cost Sensitivity Analysis

Parameter	Value	Note	Overall Hydrogen Production Efficiency (%)	Annualized Capital Cost ($M/yr)	Hydrogen Production Cost in $/kg
Technical Maturity	5% Contingency	A	No Change	333.62	2.26
	20% Contingency	B	No Change	390.02	2.56
Process Efficiency	45%	A	45%	No Change	2.26
	49%	C	49%	No change	1.85
Construction Time	48 Mo.	A	No Change	333.62	2.26
	36 Mo.	D	No Change	310.61	2.13

Notes:
[A] NOAK Baseline Case
[B] Represents a "process contingency"
[C] Captured as a 89 MWe reduction in power requirements and associated increase in power for sale
[D] Represents a reduction in Interest During construction

The effect of construction learning was addressed by assessing the impact of reduced construction time from 48 months to 36 months. This results in reduced IDC costs. The H2 production cost was reduced by 0.13 $/kg, or about 5.2%.

Sensitivity Analysis for HTE-H2-MHR Hydrogen Production Cost

Sensitivity analyses were performed for the HTE-H2-MHR model to determine the impact of two factors on the production cost of hydrogen:

- Technical maturity in terms of the operating life of the SOE cells
- Construction time in months

Table 5-4 provides the results of the sensitivity analyses

Technical maturity was addressed by doubling the annual maintenance materials component of the annual O&M cost to account for more frequent replacement of the SOE cells. This resulted in an increase 0.18 $/kg of the H2 production cost, or about 8.1%.

The effect of construction learning was addressed by assessing the impact of reduced construction time from 48 months to 36 months. This resulted in reduced IDC costs. The H2 production cost was reduced by 0.11 $/kg, or about 4.9%.

Table 5-4. HTE-H2-MHR H2 Production Cost Sensitivity Analysis

Parameter	Value	Note	Overall Hydrogen Production Efficiency (%)	Annualized Capital Cost $M/yr	Hydrogen Production Cost in $/kg
Technical Maturity of SOE Cells	Maintenance Materials = $48.8 $M/yr	A	No Change	No Change	2.22
	Maintenance Materials = $97.6 M/yr	B	No Change	No Change	2.40
Construction Time	48 Months	A	No Change	392.81	2.22
	36 Months	C	No Change	365.72	2.11

Notes:
[A] NOAK Baseline Case
[B] Represents am annual increase to account for SOE cell replacement
[C] Represents a reduction in interest during construction

6. NEXT GENERATION NUCLEAR PLANT PROJECT SCHEDULE ANALYSIS

The integrated NGNP Project schedule developed by the GA Team is provided in Appendix B of [PCSDR 2007]. As requested by INL, a D-size summary level integrated project schedule and a D-size Conceptual Design phase schedule are also provided in Appendix B of [PCSDR 2007].

The integrated NGNP Project schedule based on the Work Breakdown Structure (WBS) provided by INL was developed consistent with NGNP Project execution option 2 presented in the NGNP Preliminary Project Management Plan. For this schedule option, Critical Decision-1 is scheduled for 2008, with the expected date for initial operations (following completion of the pre-operational test program) in 2018. A two-to-three-year demonstration period to demonstrate proof of performance, including inspections to assess component performance, follows the start of initial operations. The approach taken in developing the NGNP Project schedule was to “lock-in” the initial operation milestone (2018) and completion of the demonstration period (2021), while working backward to determine the front-end milestones and durations necessary to support project completion in these timeframes. By definition, this “backward pass” through the schedule logic identifies key interface points between the responsible team members and organizations. This approach serves the dual purpose of establishing priorities (critical path) and key decision points for the management team to focus on in order to minimize schedule risk for the overall project.

Consistent with INL’s requirements, the schedule was developed to Level III detail, with additional detail (Level IV) for the Conceptual Design phase. Key schedule milestones include:

- Initiate Conceptual Design/Trade Studies - 2007
- Approve Preliminary Baseline (CD-1) – Oct 2008
- Approve Performance Baseline (CD-2) – Oct 2010
- Approve Long Lead Procurement (CD 2/3) – Nov 2011
- Approve PSAR – Dec 2011
- Approve Start of Construction (CD-3) – Dec 2012
- Issue NRC Operating License – Dec 2017
- Approve Start of Operations (CD-4) – Dec 2018
- Commercial Demo plus Inspections – 2018 to 2021

The schedule was resource loaded using the capital cost estimate. The direct hour loading on the levelized schedule shows a peak craft loading of approximately 625 (FTE). This number does not necessarily represent onsite personnel, as offsite fabrication and multi-shifting will be utilized. Figure 6-1 shows resource loading (man hours) by year based on the levelized schedule. Figure 6-2 shows a cash-flow profile developed from the levelized schedule (with contingency).

Given the current level of available detail, the schedule represents what can generally be viewed as an achievable plan to meet the goal of the PPMP. However, the current execution plan indicated to the GA Team by INL is to start “Conceptual Design” (or more specifically, additional trade studies to support design selection) on or about October 1, 2007 under contract extensions with the current reactor vendor teams. Completion of the Conceptual Design report would be scheduled for July 2009, with CD-1 shortly thereafter. This puts the project schedule approximately one year behind the timeline established in the PPMP. Due to current uncertainties with respect to project execution, this apparent delay is not addressed in the NGNP Project schedule presented herein. However, the following observations are relevant:

- Based on a critical path analysis, Conceptual Design is on the zero float critical path. Therefore, any delay in the completion of Conceptual Design will have a day for day impact on the Project Completion date, unless mitigating steps are taken. Principally, these steps would start with attempting to shorten the overall duration of Conceptual, Preliminary, and/or Final Design, as these phases are in series and can only be minimally overlapped. One specific measure that has been mentioned that may shorten the Conceptual Design phase is the use of the US/Russian International GT-MHR design information. Other options also need to be explored.

- Less likely options for shortening the overall critical path include fabrication and delivery of the RV (36 months is already considered minimal lead time for a non-forged vessel), and the Licensing and Regulatory phase (lead times and review durations fixed by others).

- Although there may be some room for improvement in the construction schedule (which at this time exists at a moderately high level of detail), the likelihood of gaining up to a year in duration is remote.

In summary, it is likely that any recovery time to be gained in the schedule will come from a combination of compressing the design schedule (Conceptual, Preliminary, Final), as well as detailing out the Construction schedule to develop strategies for possible schedule compression during that phase. As discussed, approximately one year will need to be recovered if the 2018/2021 timetable is to be maintained.

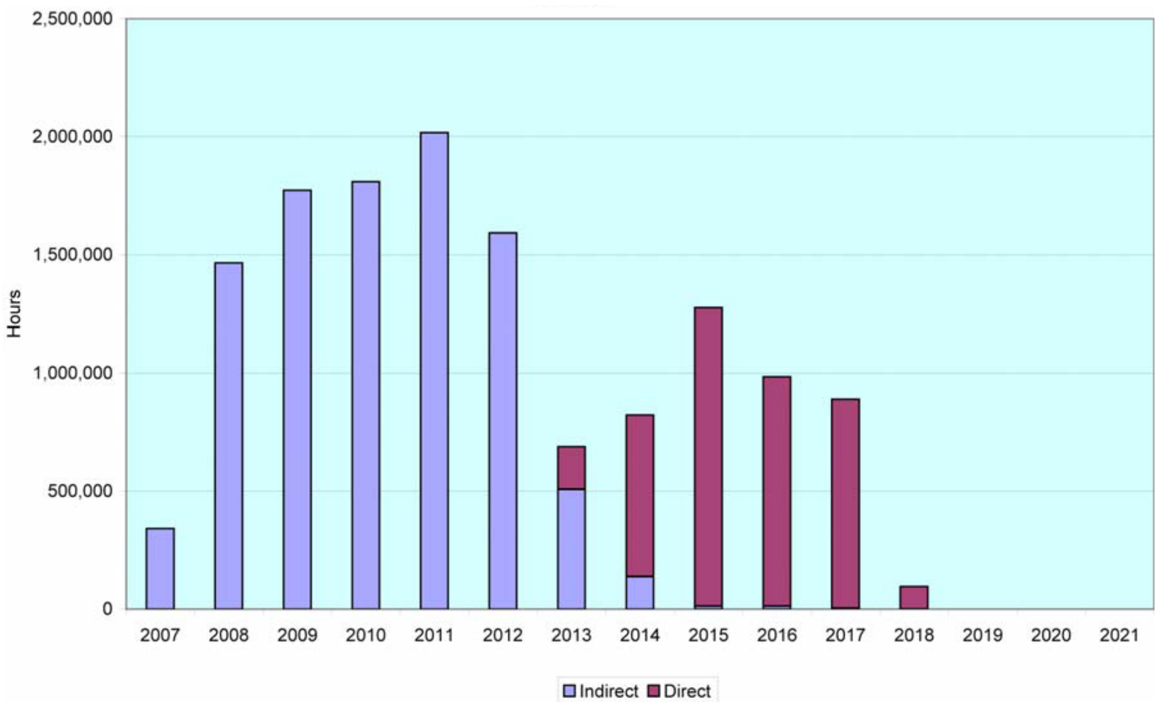

Figure 6-1. Resource Loading (Hours) Based on Levelized Schedule

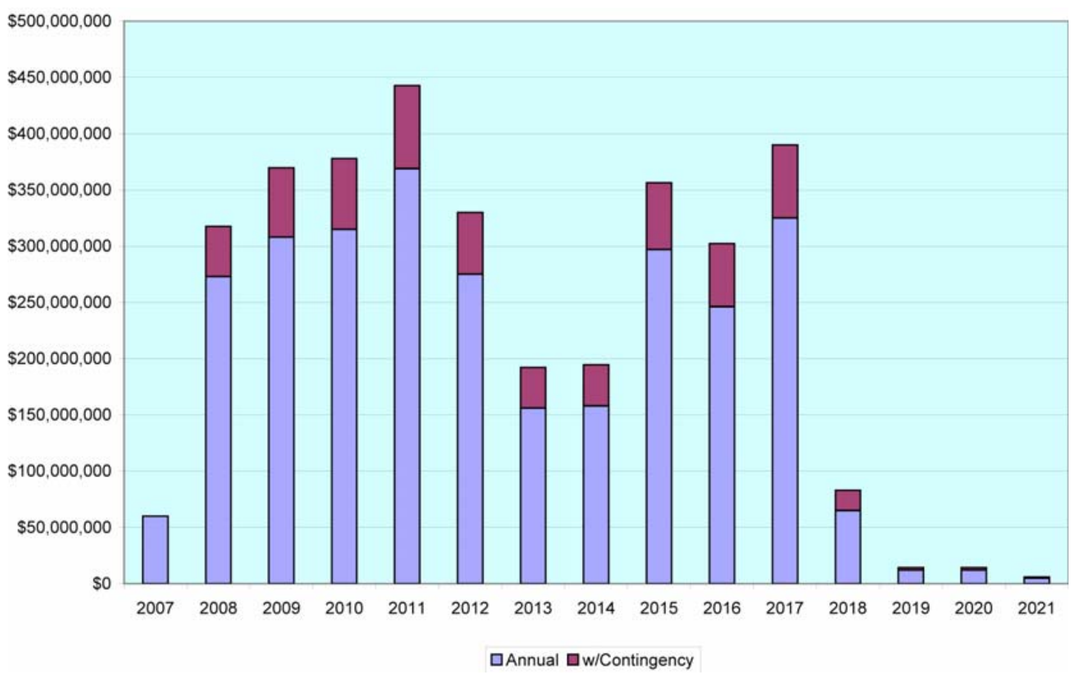

Figure 6-2. Cash-Flow by Year Based on Levelized Schedule (with Contingency)

7. TECHNOLOGY DEVELOPMENT

This section discusses the technology development required for the NGNP. Section 7.1 discusses GA's methodology for integrating MHR design and technology development. Section 7.2 summarizes the Technology Development Plan (TDP) that was prepared as a separate stand-alone document [TDP 2007] to focus and prioritize the R&D programs needed to support the NGNP based on the preconceptual design information presented in this PCDSR. Section 7.3 discusses the NGNP fuel acquisition strategy that GA has developed in recognition of the critical importance of a viable fuel supply to the success of the NGNP Project and for deployment of MHRs in the U.S. This fuel acquisition strategy allows for startup of the NGNP by 2018 and for timely demonstration by the NGNP Project of successful mass-production and irradiation of the UCO fuel that GA believes is essential for commercial deployment of MHRs. Section 7.4 presents recommendations for a testing and inspection program to be carried out at the start of NGNP operations.

7.1 Methodology for Integration of Design with Technology Development

GA uses the protocol illustrated in Figure 7-1 for integration of design with technology development in order to maximize the benefit of the technology-development programs in terms of supporting a plant design and minimizing the technical risk of the design. This model is based on successful Engineering Development and Demonstration programs conducted and managed by GA for DOE projects, including Accelerator Production of Tritium, the Salt Waste Processing Facility, the commercial GT-MHR, and the New Production Reactor.

As shown in Figure 7-1, the process begins by evaluating design requirements and reviewing existing design data from a variety of sources. Design assessments and trade studies are performed, eventually leading to key design selections and a technical baseline that meets all design requirements. It may be reasonable to revise one or more design requirements during the process if the overall impact is small. At this point, a design has been developed that meets all requirements, but requires some technology development to confirm assumptions upon which the design is based. Also, if necessary, the process allows for an early testing path to provide early confirmation of basic assumptions.

The technology development process begins with the design organization preparing design data needs (DDNs), which are formal project documents that include fallback positions in the event the testing programs do not produce acceptable results or the test could not be performed for budgetary or other reasons. The DDNs provide a concise statement of the required data and the associated schedule, quality, and accuracy requirements. In addition to preparing DDNs, the design organization also prepares a Test Specification that defines the data requirements in more detail. The technology organization is responsible for developing Technology

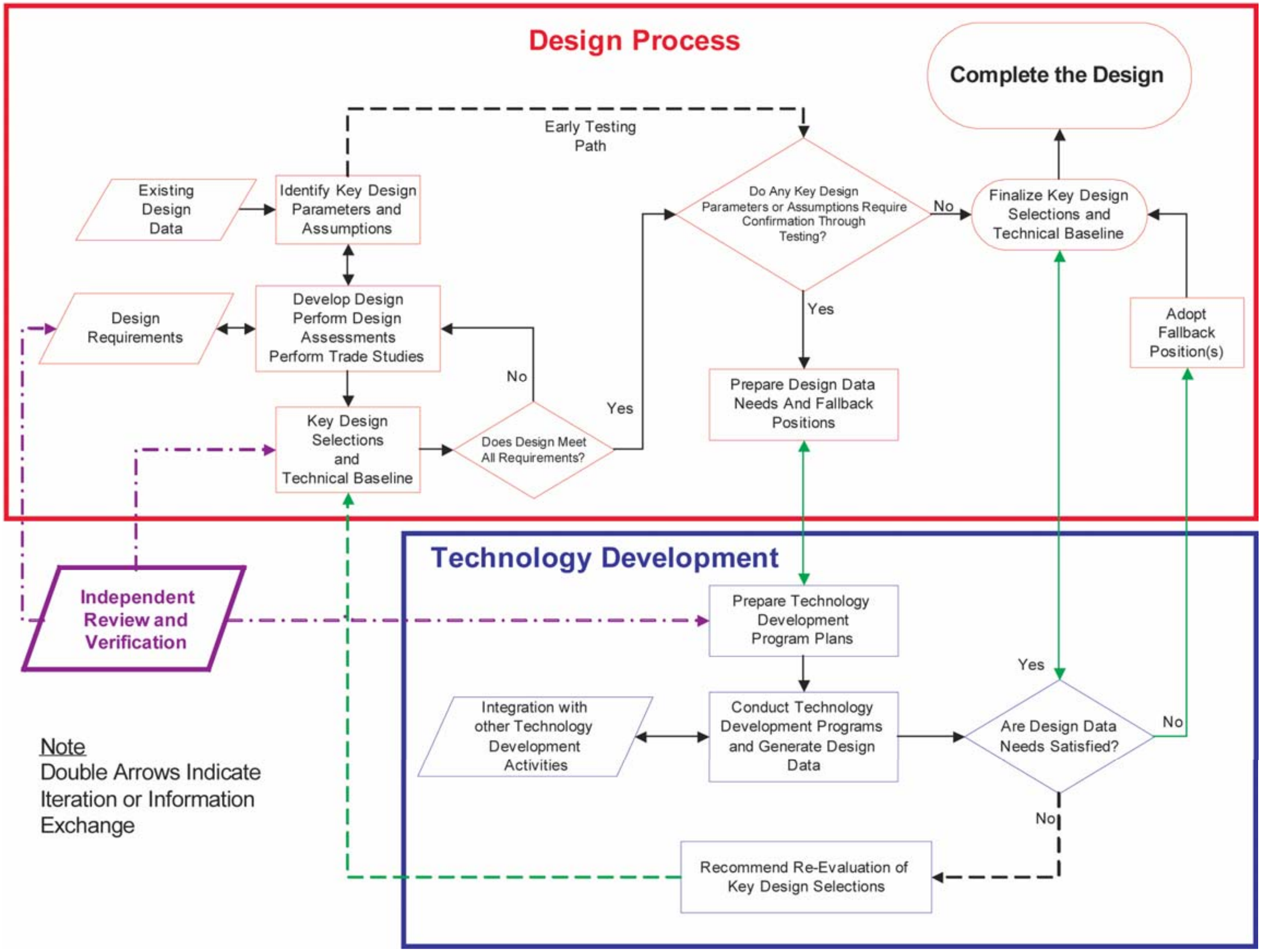

Figure 7-1. Integration of Design with Technology Development

Development Plans and Test Plans for specific tests. The design and technology organizations work together during preparation of the DDNs, Test Specifications, Technology Development Plans, and specific Test Plans.

The technology organization conducts the technology development programs and generates the design data. If feasible, the technology organization may integrate its activities with other (e.g., international) programs in order to minimize costs. After the design data are obtained, the design and technology organizations work together to determine if the DDNs are satisfied. If the DDNs are satisfied, the key design selections and technical baseline are finalized and the design is completed. If a DDN is not satisfied, the most likely path forward is to adopt the fallback position, which could mean additional margin is added to a certain area of plant design

in order to reduce technical risk. However, depending on the results of a specific test program, a more reasonable path forward may be to re-evaluate a key design selection and return to the design process. An Independent Review and Verification organization is established at the start of the process to provide oversight of both the design and technology development processes.

7.2 Technology Development

[TDP 2007] was prepared by the GA Team to focus and prioritize the R&D programs needed to support the NGNP based on the preconceptual design presented in this PCDSR. The status of the various technologies needed to support NGNP design and licensing was reviewed and summarized (to define the state of the technology), and DDNs were defined where the current data base was judged to be inadequate (to define what needs to be done to advance the technology to support NGNP design and construction). The DOE-sponsored technology programs intended to support the NGNP, including the various NGNP R&D programs and the DOE Nuclear Hydrogen Initiative (NHI) programs, were then evaluated and their responsiveness to the DDNs was assessed.

The Statement of Work for the preconceptual engineering services contract under which this PCDSR has been prepared defines critical structures, systems, and components (SSCs) as "those components that are not commercially available or do not have proven industry experience," and requires that the critical SSCs for the NGNP be identified and defined down to the component level. By definition, the critical SSCs are those components for which technology development and/or design verification testing is required. These critical SSCs include essentially all of the components of the reactor system (e.g., the fuel, the control rods, the hot ducts, and other reactor internals); the reactor, PCS, and IHX vessels; certain components of the PCS and the PCS as a whole; the helium circulators, IHX, and isolation valves in the heat transport systems; the process heat exchangers in the hydrogen production processes, the SOEC's in the HTE-hydrogen plant; the SI-process as a whole, and the various plant instrumentation and control systems. The NGNP critical SSCs and the associated design data needs (DDNs) for these SSCs have been systematically identified in the TDP

Consistent, with the above identification of critical SSCs, the effort in preparing the TDP was concentrated in the five specific areas of research, called "Major Project Elements," outlined in The Energy Policy Act of 2005, plus an additional research area that was added in the NGNP PPMP. These areas include:

- High-temperature hydrogen production technology development and validation
- Power conversion technology development and validation
- Nuclear fuel development, characterization, and qualification
- Materials selection, development, testing, and qualification
- Reactor and balance-of-plant design, engineering, safety analysis, and qualification
- Energy transfer, which includes the IHX and the secondary HTS.

In principle, the GA Team agrees that these are the priority R&D areas for the NGNP and this is reflected by the structure and content of the TDP.

Many of the resulting NGNP DDNs, particularly those related to the Reactor System and PCS are the same or similar to the commercial GT-MHR DDNs, but new DDNs have been identified, particularly for the IHX and the hydrogen production processes. When the NGNP reference design is officially declared and subsequently matures, additional DDNs will undoubtedly be defined, but it is anticipated that the major ones have been identified in the TDP. The DOE-sponsored technology programs intended to support the NGNP, including the various NGNP R&D programs and the DOE Nuclear Hydrogen Initiative (NHI) programs, were then evaluated, and their responsiveness to the DDNs was assessed. Table 7-1 summarizes the results of the evaluation.

Overall, the current NGNP and NHI R&D plans appear largely adequate to meet the DDNs with a number of important exceptions that are described below by technology area. However, with the notable exception of the technical program plan for the AGR Fuel Development and Qualification Program (AGR Fuel Program), these R&D plans are, in general, too high level and largely qualitative in nature (e.g., few test matrices, etc.). Consequently, a general recommendation is that the NGNP and NHI program plans be revised to tie them directly to the NGNP DDNs and that they be better quantified. Without more specificity, it is not clear what data will be available at what time, and it is not possible to judge the reasonableness of the R&D cost presented in those plans.

Table 7-1. Key NGNP DDNs and Technology Development Requirements

Technology Area	Key DDNs	Applicable TDP	TDP Responsive To DDNs?	Major Facility Deficiencies	Recommended Resolution
UCO Fuel Qualification	— Fabrication process integration and scale-up — Qualification of NFI 10%-enriched UO2 for initial core — UCO irradiation performance — UCO performance during core heatup	AGR Fuel Plan	— Schedule does not support 2018 NGNP start-up — No integrated fuel fabrication pilot plant — No qualification of NFI UO_2 fuel for initial core	— Limited irradiation capacity in ATR — Limited Hot Cell capacity at ATR — No in-pile facility for reactivation of irradiated fuel — No in-pile facility for R/B measurements	— Increase priority of AGR irradiations in ATR — Consider contracting with HFR Petten — Upgrade Hot Cells — Install King Furnace in NRR TRIGA — Add fuel pilot plant to AGR program — Add NFI fuel to AGR program
Radionuclide Transport	— Ag & Cs release from core — Ag & Cs plateout on turbine and IHX — I-131 release during core heatup accidents — Integral test data for methods validation	AGR Fuel Plan	— Schedule does not support NGNP 2018 startup — Plan for in-pile RN transport loop unrealistic — No RN transport in Reactor Building (VLPC)	— No in-pile RN transport loop — No facility for RN transport in VLPC tests	— Increase priority of RN transport tasks — Evaluate feasibility of loop in ATR — Consider contracting with NIIAR, RF, for use of PG-1 loop — Add VLPC mockup to in-pile loop
Spent Fuel Disposition	— “Non-combustibility” of graphite — Long-term leaching of irradiated TRISO fuel — C-14 production and transport	LEU Spent Fuel TDP	— Spent Fuel TDP not in PPMP	— None identified	— Lab review of Spent Fuel TDP — Add Spent Fuel TDP to PPMP

Table 7-1. Key NGNP DDNs and Technology Development Requirements (Cont.)

Technology Area	Key DDNs	Applicable TDP	TDP Responsive To DDNs?	Major Facility Deficiencies	Recommended Resolution
Core Graphite	— Initial screening of candidate graphites — Qualification of replacement for H-451	NGNP Materials R&D TDP	— TDP appears responsive to DDNs — Excessive number of candidate graphites	— Limited capacity for graphite irradiation	— Prepare stand-alone graphite TDP — Use capsule ACR-1 as screening capsule to reduce number of graphites — Focus on qualifying H-451 replacement graphite — Irradiate graphites at NIIAR as necessary
High-Temperature Metals	— IN 617 for IHX @ 950°C — 2¼Cr-1Mo for RV — 9Cr-1Mo-V as backup to 2¼Cr-1Mo for RV — Turbine blade alloys for 950 C (e.g., IN 100)	NGNP Materials R&D Program	— All metals DDNs addressed, except… — Turbine blade alloys need to be added — Testing needs to be prioritized (fewer candidates)	— None identified	— Add turbine blade alloys — Prioritize tasks and focus on prime and backup alloy for each application
Power Conversion System	— 950°C operation without blade cooling — Recuperator lifetime — Large EM bearings	RF GT-MHR PCU TDP	— All DDNs addressed, except… — Additions needed for 950 °C operation — Prototype testing to verify PCS design	— None identified	— Address Rolls-Royce critique of OKBM design — Add scope for 950°C operation — Monitor OKBM PCU technology program

Table 7-1. Key NGNP DDNs and Technology Development Requirements (Cont.)

Technology Area	Key DDNs	Applicable TDP	TDP Responsive To DDNs?	Major Facility Deficiencies	Recommended Resolution
Design Verification & Support (DV&S)	— Nuclear criticals (annular core) — Integrated RCCS performance — High temperature circulators	No DV&S TDPs	None	— No suitable critical facility in USA — No large, high temperature He loop for testing of components	— Prepare DV&S TDPs — Use RF ASTRA critical facility — Use OKBM He loops
H_2 Production – SI Process	— Reaction kinetics — Process integration & scale-up — Structural materials corrosion	NHI 10-yr R&D Plan	— SI DDNs apparently addressed; plan too qualitative for certainty — Pilot-plant testing critically important	— None identified	— Prepare quantitative SI TDP — Collaborate with JAEA and KAERI on SI development — Expedite pilot-plant testing
H_2 Production – HTE Process	— SOEC sealing — Production costs — SOEC service lifetime — Spent SOEC disposition	NHI 10-yr R&D Plan	— HTE DDNs apparently addressed; plan too qualitative for certainty — Pilot-plant testing critically important	— None identified	— Prepare quantitative HTE TDP — Collaborate with Toshiba on HTE process development — Expedite pilot-plant testing
Design Methods Validation	— Single-effects tests for improved models/properties — Independent integral tests for code validation	NGNP Methods Development Plan	— [Normally, design workscope; not technology task] — Overemphasis on nuclear and thermal/flow codes — Fuel performance & RN transport not included	— No suitable critical facility in USA — No high temperature He loop — No in-pile RN transport loop	— Use industry standard codes — Prepare V&V Plan for RN control codes — Use RF test facilities — Use reactor surveillance data from HTTR

7.3 Fuel Acquisition Strategy

NGNP Fuel Acquisition Strategy

GA believes that for successful commercial deployment of prismatic block VHTRs, UCO fuel with higher burnup capability must be demonstrated during operation of the NGNP. However, it is unlikely that the AGR Fuel Program will be able to qualify UCO fuel in time to support the current NGNP option 2 strategy schedule, which calls for startup of the NGNP by 2018. Another problem for early startup of the NGNP is that there is currently no capability anywhere in the world to mass produce TRISO-coated UCO fuel and it is unlikely that such capability will arise in time to manufacture the first core fuel load for the NGNP by 2018. Furthermore, there is currently no fuel vendor in the U.S. that has the capability to make an initial core of coated-particle fuel of any type for a 600-MWt prismatic-block MHR within a time frame compatible with the option 2 timeline in the NGNP PPMP. GA believes that NFI, which has produced the TRISO-coated UO_2 fuel for the 30-MWt HTTR in Japan has the largest and most advanced capability to mass produce coated-particle fuel at this time.

Given these realities, GA has formulated a fuel acquisition strategy for the NGNP based on obtaining TRISO-coated UO_2 for the first core fuel load from NFI. However, GA views use of NFI fuel for the NGNP first core fuel load (and possibly one or more reloads) only as an expedient to allow startup of the NGNP by 2018. GA strongly recommends that the NGNP Project develop a domestic supply of UCO coated-particle fuel (assuming that the NGNP is a prismatic block MHR) In order to meet the NGNP project objectives.

Because the irradiation testing and accident conditions testing data base for the NFI extended burnup fuel is somewhat limited and the available data are insufficient to show that NFI fuel could meet the anticipated NGNP fuel performance requirements, a proof test of fuel from NFI's NGNP fuel manufacturing line should be irradiated and safety tested in the U.S. to acquire additional fuel performance data to support NGNP licensing. Consequently, GA endorses the approach described in the NGNP PPMP to irradiate UCO fuel and NFI UO_2 fuel in AGR-2 and AGR-2a, respectively. However, consistent with GA's view that demonstration of UCO fuel in the NGNP is essential for deployment of commercial VHTRs in the U.S., GA does not agree that a down selection between these two fuel types be made for qualification testing in AGR-5 and AGR-6. Rather, UCO fuel should be qualified in AGR-5 and AGR-6 as currently planned, and NFI UO_2 fuel should be qualified for use in NGNP based on Japanese irradiation and safety test data, proof testing in AGR-2a, and fuel performance monitoring, as necessary, in the NGNP.

Also consistent with GA's view that it is essential that the NGNP Project demonstrate the viability of economical mass production of coated-particle fuel and develop a domestic source (or sources) of UCO fuel supply, GA recommends that an NGNP Fuel Fabrication Facility (FFF) be built in Idaho to supply the fuel for the NGNP. The NGNP FFF should be designed for a production capacity of 510 fuel elements per year. The facility would be operated at full capacity for two years to produce the initial core and the production rate would then be reduced to 340 fuel elements per year, at which rate the facility would produce a reload segment every eighteen months.

The NGNP FFF would serve as the pilot line for the first commercial fuel fabrication facility. The 510 fuel element/year process line that would be built and demonstrated in the NGNP FFF would be the basic production module that could be replicated in the commercial fuel fabrication facility. Thus, the NGNP would demonstrate the fuel fabrication technology needed for the commercial fuel supply business, thereby greatly reducing the costs and risk that would be associated with a first-of-a-kind facility. The estimated capital cost for design, construction, and licensing of the NGNP FFF based on the assumption that the NGNP FFF would be built using an existing facility on the INL site is about $200M in 2007$

Figure 7-2 shows a potential schedule for NGNP fuel acquisition. This schedule assumes that NFI will make only the first core fuel load and that the fuel would be entirely replaced with UCO fuel at the beginning of 2022 following the NGNP commercial operation demonstration period. Based on NFI's input that they would require five years to fabricate the fuel for the first core fuel load, funding of NFI to begin compact fabrication process development should begin no later than the beginning of 2008.

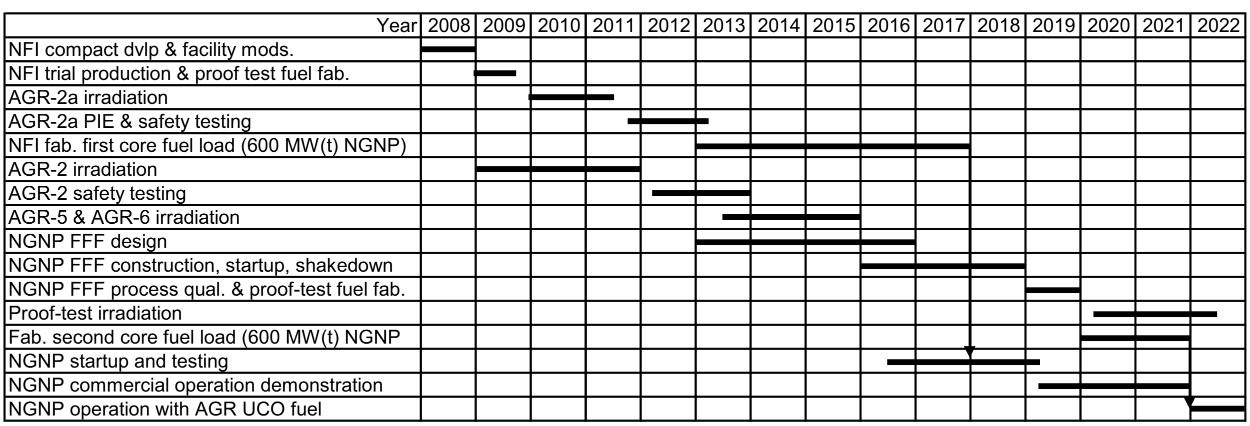

Figure 7-2. Schedule for NGNP Fuel Qualification and Acquisition

An alternate, much more aggressive schedule for the NGNP FFF would be to start design in 2008 and to design and construct the facility in parallel with fuel demonstration and qualification in AGR-2, AGR-5, and AGR-6 under the AGR Fuel Program. Under this schedule, the plant would be designed, constructed, and licensed from 2008 through 2012; started up, demonstrated, and used to make proof test fuel in 2013 and 2014. Based on satisfactory, early on-line fission gas release results from the proof test irradiation, fabrication of the first core fuel load would begin in 2016 and be completed in 2017. Although very aggressive and more risky than the alternate approach of obtaining the initial fuel for the NGNP from NFI, this approach would eliminate the substantial additional costs associated with the NFI approach.

7.4 NGNP Initial Testing and Inspection Program

A testing and inspection program is proposed to be carried out at the start of NGNP operations. The testing and inspection program, as currently envisioned, is expected to be performed over a period of approximately one year prior to startup and two years following startup. The general objective of the testing, beyond qualification of the facility for power operation, is to effectively compress the operating time by inducing events that would not normally be expected to occur during a two year operating period, to support the following NGNP Project objectives:

- Demonstrating the basis for commercialization of the nuclear system, the hydrogen production facility, and the power conversion concept. Essential elements of this objective include:
  - Demonstrating that the requisite reliability and capacity factor can be achieved over an extended period of operation.
  - Demonstrating normal O&M activities including activities required during major outages for equipment replacement or maintenance as well as O&M that might be required in the event of major equipment failures.
- Establishing the basis for licensing the commercial version of NGNP by the NRC. This will be achieved in major part through licensing the prototype by NRC and initiating the process for certification of the nuclear system design.

The proposed testing and inspections to be performed are divided into the following categories:

Preoperational Tests – These tests address the capability of selected SSCs to meet performance requirements, to the extent they can be tested outside of full plant service conditions. Successful completion of preoperational tests demonstrates that individual system performance is acceptable and the plant is ready for hot functional tests. The preoperational tests and inspections to be performed will be specified in the SSC System Design Description (SDD) documents

Baseline In-service Inspection – These are pre-operational tests of all the in-service-inspections (ISI) to be performed through out the plant's lifetime. These tests provide baseline data for comparison with future in-service inspection results.

Hot Functional Tests – In these tests, the nuclear heat supply facility (the reactor primary system) will be operated at full power reactor gas inlet temperature, flow, and helium pressure with heat supplied by motoring the helium compressor and IHX circulator. The tests will provide data on flow performance through out the primary system (pressures, temperatures, vibrations, etc) as well as functional testing of all monitoring instrumentation. In addition, a first check on vessel heat and temperature management and operation of the RCCS will be provided.

Fuel Loading – As fuel loading progresses, neutron flux monitoring results can be compared with predictions.

Startup Tests – Startup testing includes pre-critical, low power, and power ascension testing. Following verification of the core physics design, power is increased in steps to full power operation. Plant operating parameters will be verified to be within design limits, and response to load changes, transition of loads between the PCS and the hydrogen production plants and reactor trips will be demonstrated throughout the power ascension program.

Performance Tests – These tests will subject the plant to less frequent events expected to occur during normal operation including power PCS trip, loss of secondary system flow or pressure, etc.

Response to Accident Tests – These tests are intended to demonstrate the inherent response characteristics of the reactor module. Four basic categories of events are proposed: (1) reactivity transients, (2) pressurized cool down, (3) water ingress, and (4) depressurized cool down. These categories cover the performance of the key systems which provide safety and investment protection

Post Test Inspections and Maintenance Demonstrations – Following the completion of the above testing at power operating conditions, a shutdown would be scheduled for performance of inspections and to demonstrate major maintenance operations. Inspections would be performed of all the systems to ascertain any abnormal effects of the above tests. Major maintenance operations would be demonstrated such as refueling, reflector replacement, performance of remote ISI operations, and removal and replacement of major equipment items such as a TM rotor, IHX heat transfer element, major hydrogen production equipment, and other plant items not designed for the life of the plant.

The anticipated schedule for performing the testing program is shown in Figure 7-3. The data and experience gained during the test program are expected to provide a verification of commercial feasibility and a basis for design certification.

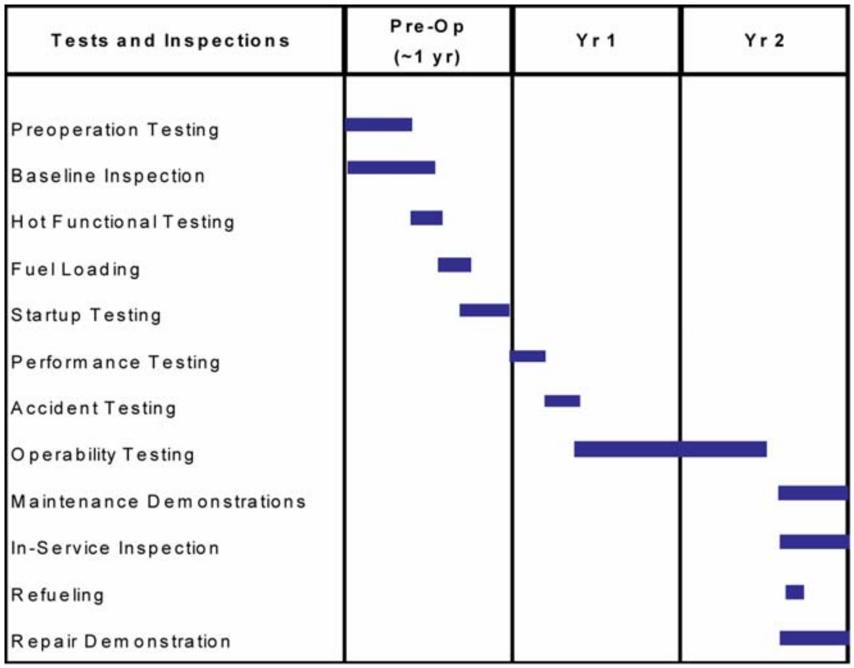

Figure 7-3. NGNP Initial Test and Inspection Program Schedule

Although preliminary planning indicates that the response to accident testing will comprise only a small fraction of the total testing interval, the tests are a major element of the total program. The tests to be performed have been developed based on a preliminary evaluation, and will be adjusted based on further evaluation of design and licensing issues as the project proceeds. The ability to demonstrate the response to low probability events in a full scale plant without damage which would preclude subsequent long term operation is a key feature of the modular helium-cooled reactor. Demonstrating this capability is a vital element in the successful development of a commercial plant which is economically competitive, and generally accepted by utility/users, the financial community, and general public.

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