AREVA NP Inc.

Technical Data Record

Document No.: 12 - 9155160 - 000

Pebble Bed Reactor Assessment Executive Summary

BEA Contract No. 000 75310

AREVA

20004-018 (10/18/2010)

Document No.: 12-9155160-000

Pebble Bed Reactor Assessment Executive Summary

Safety Related? ☐ YES ☒ NO

Does this document contain assumptions requiring verification? ☐ YES ☒ NO

Does this document contain Customer Required Format? ☐ YES ☒ NO

Signature Block

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A/A-CRI designates Approver (A), Approver - Confirming Reviewer Independence (A-CRI)

Record of Revision

Revision No.	Pages/Sections/ Paragraphs Changed	Brief Description / Change Authorization
000	Initial Issue	All

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1.0 INTRODUCTION

High temperature gas-cooled reactors (HTGRs) can provide an important addition to the US and the world's energy supply portfolio. Enabling commercial deployment of HTGR technology has gained importance as environmental and energy security issues have become more apparent, and the national resolve to solve these issues has become stronger. The Next Generation Nuclear Plant (NGNP) Project authorized by the Energy Policy Act of 2005 (EPAct) provides for a collaborative effort between government and industry to enable the commercialization of HTGR technology.

The NGNP Project is intended to enable a coordinated program including technology development, design, licensing, and demonstration of HTGR technology. The goal is to meet future energy needs for high temperature process heat and high efficiency electricity production. The current phase of the project addresses a combination of research and development, pre-application licensing interactions with the US Nuclear Regulatory Commission, and conceptual design activities. Phase 2 of the project may address subsequent areas such as detailed design, formal licensing, completion of required R&D, and concept demonstration. As required by EPAct, the Nuclear Energy Advisory Committee (NEAC) will conduct a review when the first phase of the NGNP Project is nearly complete.

Two main technology options are under consideration for the NGNP: the modular prismatic block core HTGR and the pebble bed reactor (PBR) modular HTGR. The evaluation of these two reactor concepts will form one part of the NGNP Project Phase 1 review. Conceptual design information for the prismatic reactor concept has been developed separately. The PBR technology assessment described in this executive summary is intended to inform the evaluation of the PBR concept by NEAC and others.

1.1 Scope of the PBR Assessment

The main purpose of this PBR assessment is to evaluate the current status of PBR technology and its readiness for further development and deployment. This is a broad evaluation covering several aspects of PBR readiness including the status of the current reference design, the maturity of the supporting technology base, the ability of the infrastructure to support key PBR specific needs, and the significance of key PBR technology issues and concerns.

The PBR assessment draws entirely on information from existing PBR development programs and related open literature information. Specifically, the assessment is based on the 200 MWt HTR-Module concept developed by AREVA predecessors in Germany in the 1980s. More recent PBR design and analysis information found in the open literature is also taken into account, but no new design work is performed as part of this assessment.

AREVA's role in performing this assessment is somewhat unique. While AREVA has an important legacy of PBR design and development, AREVA's current main focus is on a prismatic block HTGR concept which builds on the ANTARES concept. Thus, AREVA's only goal for this task is to provide a balanced assessment of PBR technology readiness for future development.

1.2 Assessment Approach

Since the PBR assessment is based primarily on the HTR-Module, the first step of the assessment was to work with AREVA counterparts in Germany to access the latest HTR-Module information. This information included a variety of design information, safety and performance analyses, and other relevant data, but the primary source was the Safety Analysis Report that was submitted to German regulators and formally reviewed by them. This information provided the basis for the description of the technology, and it provided the starting point for the subsequent evaluation of each aspect of PBR technology.

An expert consensus process was used wherever possible to develop the PBR assessment. At the highest level, the approach used can be divided into four basic parts:

- Define basis for assessment
- Identify assessment categories and issues
- Perform initial assessments
- Review results with assessment team experts

Decisions at each level were reviewed with the technical leads on the project, including the identification of issues and concerns and the results of subsequent assessments of those issues.

1.3 PBR Assessment Team

The PBR assessment team included participants from several sources:

- AREVA HTGR experts (US, Germany, France)
- Supporting US AREVA staff
- Other German HTGR experts (e.g., current and former Jülich Research Center staff)

- Shaw Group (A/E support and cost estimating expertise)
- Babcock and Wilcox Co. (particle fuel industrial fabrication capability)

1.4 PBR Assessment Reports

This executive summary provides and overview of the PBR technology assessment results and conclusions. The specific intent is to focus on those topics that are most critical in evaluating the PBR concept. Not every topic addressed as part of the assessment is captured in the executive summary.

The complete results of the PBR assessment are documented in four separate reports:

- Pebble Bed Reactor Plant Design Description, AREVA document 12-9149697-000, January 2011
- Pebble Bed Reactor Technology Readiness Study, AREVA document 12-9151714-000, January 2011
- Pebble Bed Reactor Scoping Safety Study, AREVA document 12-9149863-000, January 2011
- Pebble Bed Reactor Cost and Schedule Report, AREVA document 12-9151202-001, February 2011

The PBR design description report describes the HTR-Module which was the reference concept for the assessment. The description report also identifies potential design advancements based on subsequent PBR and HTGR development work since the HTR-Module was developed in the 1980s. These advancements should be considered, if the HTR-Module design is selected as a basis for future PBR development. The description report also presents a summary of the key NGNP design requirements. Finally, it provides an assessment of the steady state and transient performance of the HTR-Module.

The PBR technology readiness report evaluates the deployment readiness of PBR technology. The main issues and concerns frequently identified with PBR technology are identified and evaluated. The assessment report also evaluates the status of the existing HTR-Module design, considering both design maturity and alignment with key NGNP requirements. The existing Design Data Needs supporting PBR technology are evaluated, and any gaps are identified. The report also examines PBR fuel acquisition alternatives in light of recent global industry developments.

The PBR scoping safety analysis report evaluates the safety characteristics of the HTR-Module. This evaluation is based on the safety analyses submitted to the German regulators in the 1980s. While new safety calculations were not performed, the doses were adjusted to conform to US analysis guidelines. The safety report also discusses the safety impact of the key PBR technology issues.

The PBR cost and schedule report provides a cost estimate for the development and deployment of a first-of-a-kind (FOAK) PBR plant based on the HTR-Module concept. A cost estimate is also provided for a mature Nth-of-a-kind plant containing multiple HTR-Modules. Finally, a project schedule for development and deployment of the FOAK plant is provided.

2.0 REFERENCE PBR DESIGN

2.1 Selection of Reference Design

The reference design selected for the PBR technology assessment is the HTR-Module. The HTR-Module is a 2x200 MWt (dual unit) modular pebble bed reactor design that was developed in Germany in the 1980s for the co-generation of electricity, process steam for the chemical industry, and/or district heat.

The HTR-Module is designed so that in the case of failure of all active cooling systems and a complete loss of coolant, fuel temperature limits are not exceeded, so that there is virtually no release of radioactive fission products. The design concept was reviewed and approved by German regulatory authorities, and progressed to a final design stage.

The HTR-Module design benefited from considerable German operating experience with pebble bed reactors, including the 15 MWe experimental pebble bed reactor, AVR, which operated from 1967 to 1988, and the 300 MWe prototype pebble bed reactor, THTR-300, which generated electricity from 1985 to 1989. The HTR-Module design formed the basis for subsequent modular PBR designs, including the South African PBMR and the Chinese HTR-PM, and therefore was selected as the reference design for this PBR technology assessment.

The primary sources of information on the HTR-Module design for the technology assessment are the HTR-Module Safety Analysis Report dated April 1987 and the subsequent Safety Analysis Report from 1988. The safety analysis reports contain descriptions of plant systems, plant operation, and the supporting accident analyses. The information in the safety analysis reports was augmented with design information from the global AREVA resources in Germany and France and from Jülich Research Centre in Germany.

Deviations from the baseline HTR-Module design were not considered for incorporation into the reference design. The project scope and schedule did not allow sufficient time to thoroughly evaluate the impact of design changes or to re-evaluate the accident analyses. However, some potential advancements to the design are proposed in Section 4.0 for consideration in the next phase of the project. Furthermore, it is expected that changes to the design will result from the process of adopting the German design in the United States and as a result of implementing technologies that have evolved since the 1980s.

2.2 Summary Description of Reference Design

The HTR-Module design consists of two 200 MWt reactor units, each individually coupled to a turbine generator. Each reactor unit consists of a pebble bed core, a steam generator, a helium circulator, a pressure vessel unit, and associated support systems. The helium coolant is forced downward through the core and into the steam generator, where it transfers heat to water and steam flowing in helical coil tubes. With a reactor outlet temperature of 700°C, the steam generator produces 530°C steam, which is used for electricity generation by the turbine generator or for process applications. The two reactor units are housed within a single reactor building, which provides protection against external hazards. Due to fission product retention capability of the fuel particles, a leak-tight reactor building is not required to comply with accident dose limits. To minimize impact on the environment of a postulated primary system break, the reactor building is provided with a sub-atmospheric pressure system, a pressure relief system, and a filtering system.

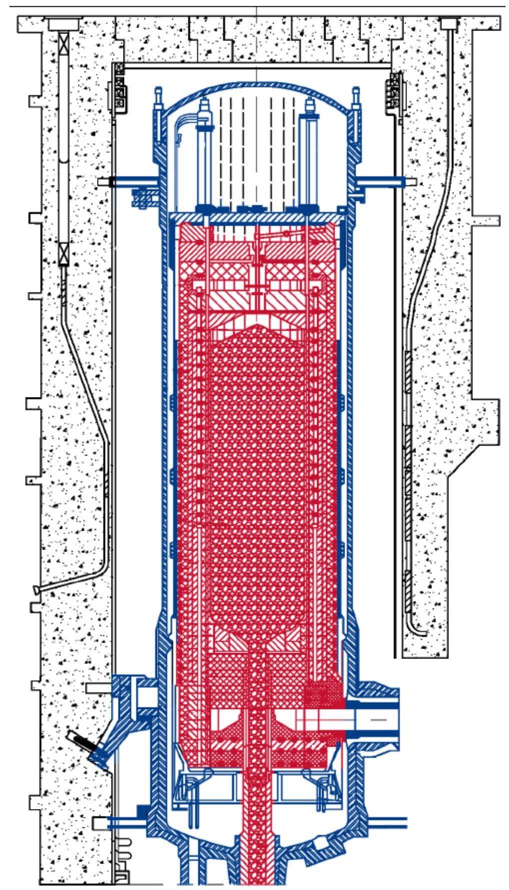
2.2.1 Reactor Core

The active core consists of a loose pebble bed of approximately 360,000 spherical fuel elements. The fuel elements are enclosed by a cylindrical ceramic core structure consisting of side, bottom, and top reflectors that reflect neutrons leaving the core back into the pebble bed. The ceramic core structure is enclosed by a metallic core barrel and the entire core is enclosed in the reactor pressure vessel as shown in Figure 2-1.

The reactor core is designed for operation between 50% and 100% power during normal power operation. Two independent shutdown systems ensure adequate shutdown margin. One shutdown system consists of six B_4C reflector rods that are inserted by gravity into the side reflector on actuation by the reactor protection system. Under normal operating conditions, the reflector rods compensate for changes in reactivity due to load changes.

The other shutdown system consists of 18 small ball shutdown units. The small ball shutdown elements containing B_4C are inserted by gravity into side reflector columns when actuated. The small ball shutdown system is provided for cold and long-term shutdown.

Figure 2-1: Reactor Core

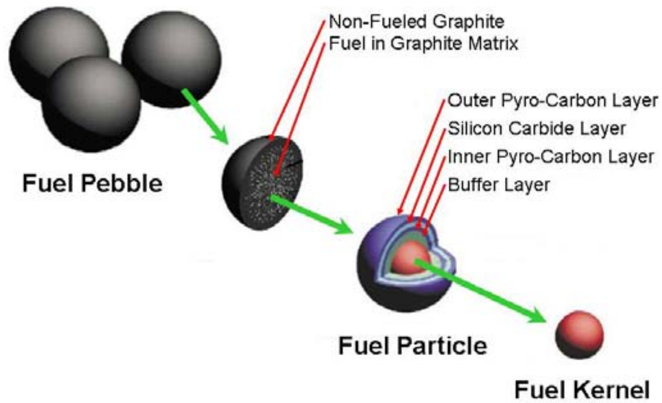


Each reactor has a separate, independent, and dedicated reactor protection system. In the event of an accident, the reactor protection system automatically shuts down the reactor and initiates protective actions.

The power density (3.0 MW/m^3) and geometry (3m diameter) of the reactor core are configured so that a maximum fuel temperature of approximately 1600°C is not exceeded under accident conditions, even without active removal of residual heat from the core.

The fuel elements are continuously recycled through the core to obtain a uniform power density distribution. On average, a fuel element will pass through the core 15 times before it reaches its target burnup of 80 GWd/MTU. The average dwell time in the reactor core for a fuel element is approximately 1000 full power days.

Figure 2-2: PBR Fuel Pebble



The fuel pebbles, shown in Figure 2-2, are 60mm diameter spheres with an inner fueled zone containing 7 grams of uranium at approximately 8% enrichment by weight. The fuel is in the form of spherical, 0.5mm diameter kernels surrounded by a buffer of porous carbon, two pyrolytically deposited layers of carbon, and one layer of silicon carbide (TRISO particles). The coatings provide a pressure boundary that has been demonstrated through extensive qualification testing and operating experience to confine fission products, preventing any significant releases of radioactivity when subjected to accident conditions.

For the first core, a mixture of lower enriched fuel elements and moderator and absorber elements are used to achieve the target core reactivity. The lower enriched fuel elements and moderator and absorber elements are gradually removed by the fuel handling equipment and replaced with equilibrium core fuel elements.

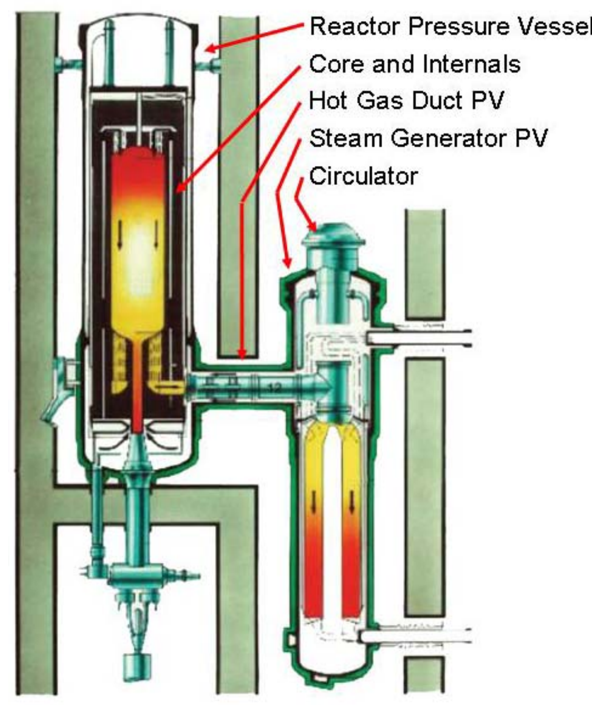
2.2.2 Primary Circuit

The primary circuit for each reactor unit, shown in Figure 2-3, consists of the reactor pressure vessel containing the core and core internals, the gas duct pressure vessel and hot gas duct, and the steam generator pressure vessel containing the tube bundle and the circulator. The primary circuit has an operating pressure of approximately 60 bar and a coolant mass flow of approximately 85 kg/s.

The 700°C helium leaving the reactor core is conveyed along a horizontal hot gas duct to the steam generator, where it flows through the tube bundle (shell side) from top to bottom. The steam generator consists of helical tubes connected to the feedwater and main steam systems. The cooled helium then passes through the annular gap between the steam generator shroud and the steam generator pressure vessel wall to the circulator located in the upper portion of the steam generator vessel. The cold gas leaving the circulator returns through the outer annulus of the gas duct pressure vessel to the lower region of the reactor pressure vessel. It then flows up to the core inlet plenum above the reactor core through channels in the side reflector. This flow circulation pattern and the side-by-side vessel arrangement reduce pressure vessel exposure to hot gas and allow upward evaporation in the steam generator tubes.

The reactor pressure vessel is approximately 25m in height with an inside diameter of 5.9m and a dry weight of approximately 830 tons. The steam generator pressure vessel is approximately 22m in height with an inside diameter of 3.6m and a dry weight of approximately 280 tons. The steam generator contains 220 helical tubes with a heat surface area of 2100m^2.

Figure 2-3: Primary Circuit



2.2.3 Fuel Handling and Storage

The fuel handling system continuously supplies the core with fuel by feeding new fuel elements into the core, recirculating partially depleted fuel elements, and discharging spent fuel elements to storage/shipping casks. Damaged fuel elements and fragments are separated and discharged into failed fuel casks. Every full-load day, approximately 5000 fuel elements are circulated and approximately 360 new fuel elements are introduced into each reactor core. Fuel elements are conveyed in horizontal and vertical tubes either by gravity or pneumatically, mainly by primary helium coolant at primary system pressure and cold side temperature.

The burnup of each discharged fuel element is determined by gamma spectroscopic evaluation (662 keV gamma line of the isotope Cs-137) to decide whether it should be returned to the core or placed into storage for disposal. Spent fuel elements (360 per day per reactor) are buffered and then loaded into storage/shipping casks. Once the filled casks are conditioned, they are transported to an expandable onsite facility for temporary storage. Dedicated casks are provided for a complete core offload in case of extraordinary repairs. Casks in the storage facility are cooled by natural convection.

2.2.4 Cavity Cooling and Residual Heat Removal

The reactor cavity is cooled during operation and after shutdown by a cavity cooler consisting of vertical tubes arranged side-by-side to form a closed panel wall between the reactor pressure vessel and the cavity wall. The cavity cooler is supplied by an operational and a safety grade cooling system. The cavity cooler can operate passively and maintain design basis temperature limits for 15 hours post-accident before active systems or operator actions are necessary.

2.2.5 Energy Conversion Plant

The energy conversion plant receives steam from the steam generators for electricity generation or process applications and returns feedwater to the steam generator. The steam/power conversion system consists of two identical, suitably interconnected units. This arrangement permits uninterrupted continuation of operation during overhauls or failure of plant equipment.

For the purposes of the technology assessment, a representative configuration was assumed in which extraction steam from the high pressure turbine is used to generate high pressure and low pressure process steam in two reboilers. The actual energy conversion plant configuration would be determined based on customer requirements for a given application.

A steam generator isolation system separates the nuclear heat source from the conventional water/steam cycle. Whenever a reactor is tripped, the associated steam generator is isolated by the reactor protection system, which shuts off the main steam and feedwater lines by closing two isolation valves in series in each line.

In the event of a steam generator tube break, the water or steam entering the primary circuit is limited by means of rapid secondary side emptying via the steam generator relief (dump) system. After draining the steam generator, the relief valves close to prevent primary system depressurization.

2.2.6 Plant Systems

Other plant systems, including helium support systems, ventilation systems, electrical and control systems, and cooling water systems, along with building structures, are described in the design description report.

3.0 STATUS OF REFERENCE DESIGN

As part of a comprehensive status assessment of pebble bed reactor technology, the status of the reference design described in Section 2.0 was evaluated to determine its potential to support NGNP program requirements. This evaluation focused on two elements: 1) the alignment of the design with key NGNP design requirements, and 2) the maturity of the design, if it were to be deployed in the United States.

The reference design was evaluated at the system level by technical staff cognizant of the design. Key NGNP requirements applicable to PBR technology were systematically extracted from the NGNP System Requirements Manual (INL/EXT-07-12999 Rev 3, 2009) and Key Design Requirements for the High Temperature Gas-cooled Reactor Nuclear Heat Supply System (INL/EXT-10-19887, 2010). The technical staff confirmed the requirements that were met and identified requirements that were either not met or could not be confirmed as part of this assessment. The technical staff also addressed a set of questions designed to gauge the maturity of the design. The responses to this questionnaire were considered along with historical design status reports from German colleagues to arrive at an overall assessment of the design maturity.

3.1 Alignment with NGNP Requirements

When considering the alignment of the reference design (HTR-Module) with key NGNP requirements, it is important to acknowledge that the HTR-Module design was developed in Germany in the 1980s, with particular customer and regulatory requirements that do not align perfectly with the objectives of the NGNP program. Therefore, failure to meet a design requirement should not necessarily be interpreted as a design deficiency. Many of the design requirements that are not strictly met by the HTR-Module design would be addressed as part of routine design activities if the design were deployed in the

United States (e.g., regulatory requirements, seismic design requirements, operational lifetime). A few substantive areas where the HTR-Module deviates from the NGNP requirements are identified below.

3.1.1 Reactor Outlet Temperature

The HTR-Module reactor outlet temperature of 700°C is below the range specified in the NGNP requirements. However, the HTR-Module reactor outlet temperature is well suited for high temperature steam production. Furthermore, operating experience has demonstrated PBR technology with temperatures up to 950°C. The reactor outlet temperature for a PBR deployment would be optimized during design considering the pertinent requirements.

3.1.2 Passive Residual Heat Removal

The passive residual heat removal mode of the HTR-Module maintains temperatures within design limits for 15 hours under conditions of loss of forced cooling, in excess of German regulatory requirements at the time of design. Beyond 15 hours, active components or operator action is required for continued passive heat removal to maintain all temperatures within design basis limits. An alternative cavity cooling design is proposed that would extend the duration of passive heat removal to meet NGNP expectations.

3.1.3 Net Generation Efficiency

A preliminary heat balance of the reference design with a representative steam cycle configuration indicates net cycle efficiency (for electricity production) of approximately 40%, less than the stated NGNP requirement of 42%. A PBR deployment project would consider efficiency, reliability, and cost in determining an optimal design meeting the pertinent requirements.

3.1.4 Shutdown Margin

For the first core of the HTR-Module, the temperature coefficient of reactivity is insufficiently negative that there is a possibility for the core to return to criticality at temperatures below 100°C, under adverse conditions. This issue is addressed in more detail in the technology readiness study.

3.1.5 Peak Accident Fuel Temperature

Conservative accident analyses result in peak fuel temperatures slightly exceeding the stated 1600°C requirement. However, the accident analyses also demonstrate that dose limits, which are a more appropriate measure of PBR performance under accident conditions, are not exceeded.

In conclusion, the reference PBR design generally aligns well with NGNP requirements. In cases where the design deviates from the requirements, the deviations could be addressed during the design process, considering the relative importance of the various requirements imposed by the project.

3.2 Design Maturity

The HTR-Module was in an early final design stage in Germany in the 1980s. The design received approval from the German regulatory authorities. Industrial customers were engaged in the project, many final design products were complete, and equipment vendor selection and equipment design was in process. However, the adaptation of the HTR-Module design to the United States and, in particular, to the NGNP perspective, would require a design reconciliation phase, followed by preliminary design activities for the US design.

The German design maturity would benefit US design efforts by providing guidance toward a complete design, including design products that could be emulated. Some of the risks inherent in a traditional design effort would be reduced, since the conceptual design has been confirmed. This could lead to a reduction in major design iterations, resulting in reduced project cost and improved schedule performance. In conclusion, the reference design is mature for the current stage of the NGNP program.

4.0 POTENTIAL ADVANCEMENTS

The reference design for the PBR technology assessment is the HTR-Module. If PBR technology were pursued for the NGNP program, AREVA's review of this technology indicates that modifications to the design concept would be expected. A number of potential advancements to the HTR-Module design are identified and recommended for consideration in future designs.

The potential advancements identified do not constitute a comprehensive list of changes to the reference design needed to define a recommended design. Design modifications would be expected as part of the normal design process, including application of modern technology, implementing customer-specific requirements, and conforming to US regulations, codes, and standard practices. A list of potential advancements was developed based on expert panel recommendations and focuses on design modifications that reflect changes in major design requirements or that have interesting implications worth addressing in the technology assessment.

The potential advancements are listed in two categories: near-term and long-term. Near-term advancements should be considered for the first generation of PBR plants developed through the NGNP project, though not necessarily the first-of-a-kind plant. Long-term advancements may be considered in future generations of PBR plants and are included to highlight the potential capabilities of the technology.

4.1 Near-Term Advancements

4.1.1 Increase Reactor Power from 200 MWt to 250 MWt

An increase in reactor power is desirable to improve the economics of the plant by increasing installed capacity without a comparable increase in capital cost. The most logical options to increase the core power without major fuel design are to either increase the core volume or increase the core power density. A balance between these two parameters should be considered. The design solution must not impact the ability to passively remove residual heat from the core. Based on experience with previous designs, passive heat removal does not pose a significant risk in the feasibility of a power increase.

The core operational margins allow for some power increase due to the significant amount of margin that was built into the original design. In particular, new computation capabilities would allow for reducing some design margin to relax certain limits.

A power increase can also be achieved with an increase in thermal conductivity of the fuel element matrix. This would lead to a power increase of approximately 10% in core power density. Independently, the reactor power can be increased by designing a larger core volume. The core radius is limited due to limitations of shutdown margin and passive heat removal. Thus, the volume increase would be achieved by increasing the core height.

The impacts of a power increase must be considered on the fuel handling system design, the steam generator, and the circulator. The increased load on the fuel handling system does not represent a significant feasibility concern. A 250 MWt helical tube steam generator does not pose a significant design problem. The circulator size increase (estimated at 4.4 MWe compared to the baseline 3 MWe) may lead to significant research and design costs. Options to limit the required circulator driving force increase include increasing the primary circuit pressure and increasing the helium temperature rise through the core. Alternatively, two circulators could be installed; parallel circulators have been successfully operated at Fort St. Vrain.

Considering the impacts, a power increase to 250 MWt is considered realistic and achievable.

4.1.2 Shared Turbine

The use of a shared turbine fed from both HTR-Module reactors is recommended for consideration for plants with the primary mission of electricity production, on the basis of improved economics and cycle efficiency. The use of a single shared turbine versus two smaller turbines reduces the overall installed cost for the energy conversion plant, and larger steam turbines exhibit improved efficiency compared to smaller units.

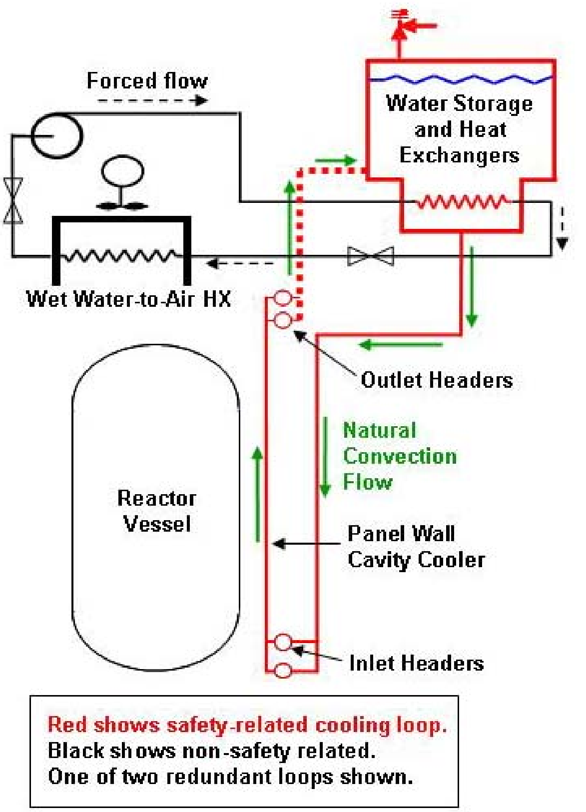
No technical impacts or research and development needs are identified to implement this change. The actual energy conversion plant configuration will be dictated by customer requirements. However, without a project-specific requirement, the use of a shared turbine is preferred.

4.1.3 Alternative Reactor Cavity Cooling System Design

The HTR-Module cavity cooler can passively remove residual heat and maintain temperatures within design limits for 15 hours following a loss of forced cooling event. After 15 hours, reactor safety is not compromised but certain non-fuel temperature limits may be exceeded unless the operator takes action to replenish the cavity cooler water supply though fire hose connections. To address NGNP requirements for fully passive heat removal during loss of forced cooling, an alternative design is proposed that could extend passive heat removal to durations expected by the NGNP project.

The proposed design, depicted schematically in Figure 4-1, uses a large water reservoir, sized to provide passive cooling for extended durations, to feed the cavity cooler. The size of the water reservoir depends on the desired duration of passive cooling. Natural circulation within the cavity cooling loop provides the necessary heat removal. The reservoir is cooled by an operational closed cooling system during normal operation and when power is available. During accident conditions when the cooling system is unavailable, the heat removed from the reactor cavity is absorbed by the water volume in the reservoir and eventually by boil-off of excess water in the tank. No valves or pumps are required to change state to continue cavity heat removal. Preliminary calculations and simulations have demonstrated the thermal hydraulic feasibility of the concept.

Figure 4-1: Alternative Reactor Cavity Cooling System



One desirable aspect of the proposed design is that the process for heat removal from the cavity remains the same under all conditions, normal and accident. No valves or pumps are required to change state and the flow field does not have to be reestablished.

No significant technical challenges are presented by the proposed design. The design is considered a reliable solution to the requirement for completely passive residual heat removal.

4.1.4 Improved Plant Availability

The HTR-Module design specified a modest unit capacity factor target of 80% and a planned unavailability target of 10% for the first-of-a-kind plant. For the Nth-of-a-kind PBR based on the HTR-Module, the following availability targets are proposed:

- Equivalent availability factor 95.0%
- Equivalent unplanned outage factor 2.5%
- Equivalent planned outage factor 2.5%

The HTR-Module is designed for continuous refueling; therefore, planned refueling outages are not required. For this reason, maintenance activities should be performed online wherever possible to avoid unnecessary outages. Maintenance activities that cannot be performed online should be scheduled to be performed simultaneously so as to minimize the number of planned outages. The scoping study identifies major maintenance and inspection activities and identifies a representative outage schedule that achieves the planned outage factor target.

For unplanned outages, the study examines forced outage rates for similar systems in operating nuclear plants and shows that 2.5% is a realistic target. Additionally, the study investigates operating experience from other pebble bed reactors (AVR and THTR) and identifies the major issues and impacts. For each major issue, the study identifies how the HTR-Module design addresses the issue to demonstrate that improved performance can be expected.

The scoping study is necessarily preliminary and a full reliability, availability, maintainability, and inspectability (RAMI) analysis would need to be performed to confirm that the proposed targets are reasonable. However, the results of the study indicate that a 95% availability target is reasonable based on historic experience and calculation.

4.1.5 Magnetic Circulator Bearings

The circulator in the reference design uses conventional oil-lubricated bearings, although magnetic bearings are identified as an alternative design. The magnetic bearing alternative should be considered, primarily to eliminate the risk of lubricant leakage into the primary circuit, and also for ease of maintenance.

The representative active magnetic bearing design contains two radial bearings and one axial bearing, each with an associated catcher bearing serving as a backup when the magnetic bearings are not available (e.g., loss of power). The radial magnetic bearing is a cylindrical magnetic guide composed of electromagnetic steel sheets fixed on a bushing which is fitted on the shaft. Each radial bearing has a set of radial displacement sensors that allow active regulation of the shaft position, including assistance for vibration management. The axial bearing also consists of a stator/rotor, displacement sensors, and an electronic control system. The catcher bearings consist of conventional ball bearings using dry lubricant and contact the shaft by means of friction cones.

Active magnetic bearings are used in various industries, notably in the natural gas industry, in turbo-machinery at power levels beyond 15-20 MWe. The unique feature of PBR applications beyond these conventional applications is the helium environment and its specific tribology, which results in development needs, specifically for the catcher bearings. Tests supporting the German HTR-500 project demonstrated severe wear of the catcher bearing friction cones in the helium environment. Tests are needed to demonstrate the performance of the magnetic bearings with the integral rotating equipment assembly and its regulation system throughout the whole range of operating conditions, including the anticipated number of actuations of the catcher bearings.

4.2 Long-Term Advancements

4.2.1 Long-term Potential for PBR Technology

The HTR-Module design was developed to serve a wide variety of potential applications, including electricity production, supply of high temperature process steam, and various cogeneration configurations. The potential of HTGR technology to go beyond traditional configurations and applications has long been recognized. HTGRs, including PBRs, have the potential to go to even higher temperatures to serve very high temperature applications, they have the potential to support diverse power generation and energy transfer configurations, and they have exceptional fuel cycle flexibility.

With current materials, core outlet temperatures in the range of 700-750°C are readily achievable. Core structural materials are capable of temperatures well beyond this range. Fuel operating temperatures are placed under increased strain as reactor outlet temperature is increased, but core designs with outlet temperatures in the range of 900-950°C are achievable and have limited operating experience. The greatest challenge to very high temperature applications is the heat delivery system. Direct heat delivery systems require some form of intermediate heat exchanger (IHX) to transfer the heat from the primary loop to a secondary heat transport loop. Significant materials challenges remain to be resolved for very high temperature IHXs. Nonetheless, workable solutions are thought to be achievable with adequate research and design innovation.

The PBR concept is also adaptable to future power generating systems such as a direct Brayton cycle. A direct cycle using the helium coolant as the working fluid in a gas turbine could take advantage of the higher generating efficiency of a Brayton cycle at high reactor outlet temperatures, eliminate significant secondary equipment to save cost, and eliminate thermodynamic inefficiencies associated with heat transfer to a secondary working fluid. The most challenging aspect of deploying a PBR in a direct Brayton cycle is the development and integration of the power conversion system hardware (helium turbine, compressor, and recuperator). The selection of a more advanced power generating system for the PBR would be a complex decision involving development cost, capital cost, and system performance.

The neutronic characteristics of HTGRs, including PBRs, allow substantial fuel cycle flexibility. Fuel cycle studies have evaluated a variety of scenarios to take advantage of this flexibility, including cycles using mixed oxide (MOX) fuel and pure plutonium, cycles designed to burn actinides and cycles using spent light-water reactor fuel, and thorium cycles based on early HTGR development. More advanced concepts that have been proposed include the use of multiple fuel forms within the core. Pebble bed reactors do not allow effective spatial zoning for different fuel forms, but online refueling presents the possibility of different core residence times for different constituents to customize fluences and burnups.

The PBR concept offers a broad range of capabilities that can support additional new markets and energy needs. With suitable additional development work tailored to the specific application of interest, the experience gained demonstrating the current PBR concept will provide a solid foundation for advanced concepts in the future.

4.2.2 Supercritical steam cycle

HTGRs, including PBRs, have the ability to generate high temperature and pressure superheated steam for efficient power conversion. Raising steam generator temperature and pressure increases power conversion efficiency but is limited by available materials and the economics associated with construction of increasingly high pressure vessels and steam turbines. Higher efficiency power cycles, such as current supercritical and proposed ultra-supercritical designs, increase the amount of useful energy which can be produced, thus reducing the capital cost per installed capacity. Research to develop key components that enable higher conversion efficiencies should be considered to leverage the value of HTGR technology.

5.0 TECHNOLOGY ISSUES/CONCERNS

This section of the report describes a series of issues that have been identified by various stakeholders as potentially problematic for deployment of the PBR technology. These issues represent a mix of both technical and perceptual challenges. For each of these issues, an assessment has been conducted, that encompassed:

- A description of each issue identified by various stakeholders as problematic for PBR deployment
- An assessment potential impact on implementation of PBR technology
- Identification of remaining open questions than need to be addressed during design process
- Identification of design changes which may alleviate identified concerns

No "showstoppers" were identified for deployment of the PBR concept. The following list identifies the eight technology issues investigated and summarizes the conclusions reached.

Stochastic Core: It is well understood and manageable with appropriate design margins.

Core Compaction: Reactivity insertion is manageable.

Graphite Dust: Dust produced in the PBR core and fuel handling system must be evaluated for the specific NGNP design, but it is not expected to be a showstopper. Current data indicates that potential dust generation and release during credible accidents is within acceptable amounts.

Broken Pebbles: Pebble breakage and trapping are both minimized by current design.

Proliferation Resistance: PBRs have both advantages and disadvantages for proliferation resistance. Proliferation risk must be evaluated for each specific PBR concept.

Shutdown Margin: HTR-Module shutdown margin is marginally adequate. Required available shutdown margin must be addressed as a licensing issue and must be considered in detailed PBR core designs.

Online Refueling: Refueling system has potential to impact plant reliability, though minimal negative impact on plant reliability expected with mature fuel handling system design.

Tritium: Tritium generation and transport expected to be manageable. Tritium transport must be considered during detailed design activities in light of specific end product contamination requirements.

5.1 Stochastic Nature of PBR Core

Issue: In the PBR core the fuel position is not known deterministically. Imprecise knowledge of local conditions may result in localized high temperatures, i.e., "Hot Spots". The AVR melt wire results might illustrate this.

In a PBR, partially-burned fuel pebbles are continuously reintroduced on top of the pebble bed along with fresh pebbles. They then slowly move downward through the reactor core; therefore, their physical properties can only be estimated statistically on an average basis. Since there are a large number of fuel pebbles in the core, about 360,000 in the case of the HTR-Module design, only the average behavior can be investigated. The question to be addressed is how to determine the uncertainties associated with the average properties in relation to the maximum power peaking and maximum fuel temperature in the reactor specifically for the HTR-Module design.

5.1.1 Uncertainties in Pebble Movement

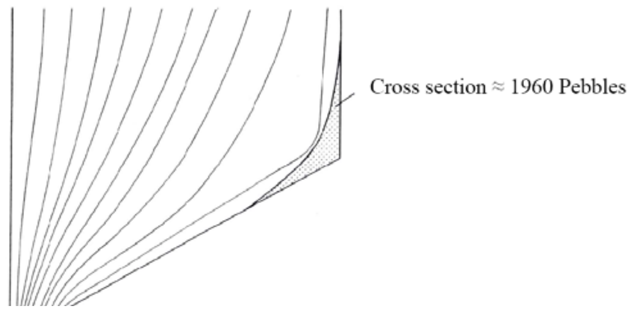
The pebble flow behavior in a PBR is important for temperature distribution within the core (both fuel and coolant) and the loading scheme. Extensive AVR and model experiments have been performed to investigate the pebble flow behaviors in the PBR. Categorized findings from these investigations are summarized below:

Feeding of Pebbles: The radial spreading of pebbles at the top of the fuel pile is a complex function of drop height, dropping rate, location of the dropping point, and angle of repose of pebbles. Since the HTR-Module uses single central fueling tube, unlike those of AVR design, it is expected to have a pure vertical fuel loading forming one central fuel pile on the top surface of the core.

In-Pile Behavior: The major findings of the in-pile pebble flow behavior are:

- The pebble flow is well organized and streamlined as shown in Figure 5-1. The pebble throughput time can be predicted with high accuracy.
- The streamlines do not cross each other.
- Pebble flow velocities are slower near the top of core, but increase sharply towards the defueling cone.
- The HTR-Module design meets the criteria on the defueling cone inclination > 30° and core diameter-to-height ratio > 0.8; therefore, a uniform pebble flow velocity profile can be assumed for the HTR-Module core.

Figure 5-1: AVR Flow Lines and Stagnant Zone



Stagnation Zone: The ANABEK experiment demonstrated a very uniform velocity profile can exist in the core given certain geometric conditions. This result has been interpreted using the HTR-Module core geometric parameters, and it is concluded that in the case of HTR Module geometry no stagnant zone occurs.

Crystallization: In order to avoid the crystallization of pebbles along the reflector walls, the HTR-Module reflector inward surface is slotted with indentations to enhance random motions of the pebbles near the wall. These slots force the pebbles to move slightly radial on the way down, thus crystallization is avoided in the HTR-Module.

Bridging Formation: The probability of formation of a dome arching over the discharge tube depends on the ratio of the fuel discharge tube diameter to the pebble diameter. Experimental findings show that no domes are formed when the fuel discharge-tube-to-pebble diameter ratio is 5 or greater. The tube-to-pebble diameter ratio for the HTR-Module is 10; therefore, no dome formation is expected to occur in the HTR-Module core.

5.1.2 Other Uncertainties

The statistical pebble bed packing fraction is about 0.61. Operational experiences from AVR and THTR have demonstrated that the void factor remained constant during continuous refuel operation. The statistical analysis results are acceptable representation of the actual packed core in the HTR-Module.

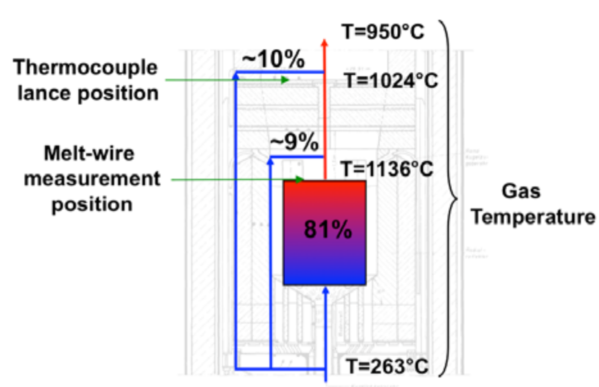
A number of experiments and numerical simulations have been performed to quantify the uncertainties concerning the core power and temperature profiles in terms of statistical distributions or bounding margins. The studies include the evaluation of AVR melt-wire temperature experiment and hypothetical simulations of clustered high-reactive fuel pebbles for the bounding values based on the deterministic hot-spot analyses performed by the PBMR and INL. These studies have indicated acceptable results for all realistic pebble clustering scenarios.

5.1.3 AVR Melt-Wire Experiment

The 1986 AVR Melt-Wire Experiment provides valuable information on the maximum fuel temperature distributions as fuel pebbles pass through the core. Based on the analysis of the experimental data, the following observations can be made:

- The least squares fit of temperature measurement data yields the two Gaussian distributions with temperatures of 1100 ± 66°C for the inner core and 1220 ± 100°C for the outer core. The relative population of each distribution corresponds to the a-priori known fraction of melt-wire pebbles in the inner and outer AVR core.
- This statistical evaluation of the AVR melt-wire experiment results provides a clearer understanding about the average values and uncertainties of the AVR fuel temperatures in the inner and outer core regions, and the necessary information for fuel performance predictions.
- The reasons for the large difference between mean exit temperature and maximum fuel temperature are specific to the AVR design with the four graphite buttresses protruding into the core and the strong effects of core bypass flow. In particular, reevaluation of the AVR coolant flow distribution confirms that bypass flows were significantly higher than expected, resulting in reduced core flow as indicated in Figure 5-2.

Figure 5-2: Estimation of AVR Core Bypass Flows from Measurements



- The probability of the maximum AVR fuel temperature exceeding the TRISO particle fuel accident temperature guideline of 1600°C is less than 0.001.

- The HTR-Module core design consists of one homogeneous fuel zone with an average core outlet coolant temperature of 700°C; therefore, it can be concluded that its maximum fuel temperature profile also follows a Gaussian distribution with a lower mean value than that of AVR.

These observations support the conclusions that the bypass flows in AVR were higher than had been considered in the core performance analyses and that reanalysis of the core taking into account current bypass calculations aligns very well with the meltwire results.

5.1.4 Hot Spot Analysis in PBRs

In pebble bed reactors, the loading and movement of fuel pebbles through the core were thought to follow random processes. These stochastic processes generate concern that the possible development of “hot spots” resulting from clustering of low burnup pebbles which may form in the regions of high thermal neutron flux, thus generating excessive local power and fuel temperature.

Several studies were performed investigating the effect of accumulation of fresh (or highly reactive) fuel pebbles, on their power and temperature loads during normal reactor operation and in accident scenarios such as a depressurized loss-of-flow cooldown (DLOFC) event.

In a hypothetical INL study on the PBMR core peaking power, a batch of 20 fresh fuel pebbles were introduced into the region with the highest power where the maximum volume averaged power peak occurs. The results of this study show that addition of 20 fresh fuel pebbles in the region of peak power has little effect on the power density and only increases the maximum fuel temperature by about 17 °C in normal operation.

A simulation study performed at INL analyzed the consequences of the formation of clusters of pebbles of 2 up to 18 combined with an estimation of the probability for their occurrence in a 300-MWt pebble-bed reactor with an annular core.

The INL study leads to the following observations:

- The peak fuel temperature in all cases is unchanged from the nominal unperturbed value in normal operation.
- Though maximum power does increase due to pebble clustering, the temperature impacts during normal operation are moderate.
- In an extremely unlikely case of agglomeration of 18 fresh pebbles clustered in the highest core power location results in a less than 60°C fuel temperature increase during DLOFC event

5.1.5 Stochastic Core Conclusions

Based on experimental and analytical results, the stochastic nature of fuel pebbles movement in the PBR core is adequately understood. Due to the simple continuous geometry of the HTR-Module and the chamfered edge at the upper end of the defueling cone, problems that have been observed in the past, such as the presence of a stagnant zone, will not occur. Furthermore, pebble flow paths through the core do not cross readily and are predictable with statistical methods.

The AVR melt-wire experiment provides valuable information on the maximum fuel temperature distributions as fuel pebbles pass through the core. Analyses of the AVR data using statistical approach show that fuel temperature in the outer core region is higher than the inner core. Although the AVR temperature measurements were higher than expected, detailed 3D CFD studies show that the temperature differences were mainly due to higher than estimated coolant bypass flows and inner/outer core radial power and temperature distributions, which were not included in the original analysis.

Studies show that although the maximum power delivered in a fuel pebble may increase due to low burnup pebbles clustering, the maximum fuel temperatures increases only moderately in normal operation. In a DLOFC scenario, fuel temperature guidelines are exceeded only slightly in a small region for a short duration under very conservative assumptions.

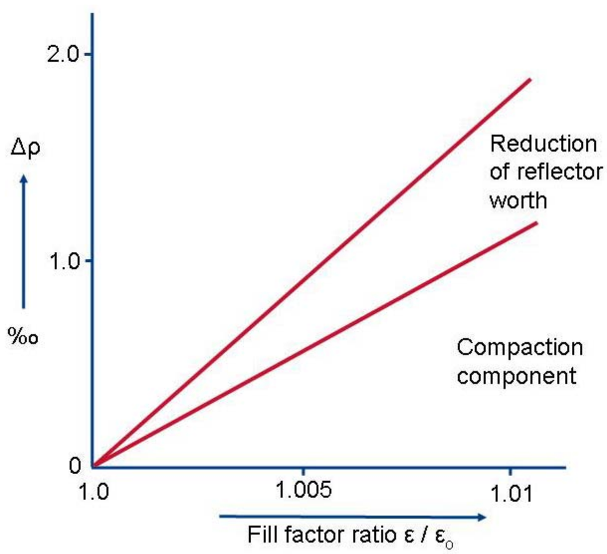
The analysis of AVR melt-wire data and simulations of hot-spot power peaking factor have shown that the design margins of HTR-Module are adequate to accommodate the statistical variations inherent with the PBRs. The uncertainties in the pebble bed core are well understood, and the core design margins adequately compensate for these uncertainties.

5.2 Core Compaction Scenarios

Issue: Compaction of the pebble bed during seismic events can introduce a core reactivity increase.

One of the postulated design basis accidents for the PBR is a seismic event or earthquake. An earthquake can cause the pebble bed fill factor to increase and hence, within a short time, reactivity to be inserted. The reactivity increase is due to reducing the neutron leakage from the pebble bed and movement of the pebble bed surface relative to the reflector control rods. For a postulated horizontal earthquake acceleration of 0.5g, the fill factor increases from 0.61 to 0.614 within approximately 6 seconds at constant excitation. This compaction is estimated from shaker tests of simulated PBR cores. The inserted reactivity amounts to 1.25°/oo due to compaction and 0.5°/oo due to movement of the pebble bed surface relative to the reflector rods. Figure 5-3 shows the reactivity increase due to reduced reflector worth and component compaction as the pebble bed fill factor increases.

Figure 5-3: Reactivity Change Due to an Earthquake as a Function of Fill Factor



The reactivity change due to core compaction to the maximum density was also calculated. In this case, a core model was generated accounting for a 0.64 packing fraction, which corresponds to the theoretical maximum fill state.

Given the number of 360,000 pebbles in the core, the only difference between the reference core model (0.61 packing fraction) and this disturbed model is the height of the pebble bed, which decreases by approximately 44cm with the 0.64 packing fraction.

The reactivity is calculated for a full core with both the nominal and compacted packing fraction. This analysis did not explicitly consider the impact on relative control rod insertion. Reactivity is shown to increase by slightly less than 3°/oo when the core is compacted.

Core compaction of the PBR reactor is an understood and manageable phenomenon. The mechanisms of compaction during seismic events are understood.

Reactivity is expected to increase due to pebble movement and reflector rod worth; however, the reactivity transient resulting from a seismic event is understood and manageable. Thermal-Hydraulic impacts are understood and of lesser consequence that the reactivity impacts. No significant consequences are anticipated during normal operations or following a design basis accident. The results are manageable and bound by other reactivity events.

5.3 Graphite Dust

Issue: Excessive graphite dust generated in the core and FHS may facilitate increased fission product release under accident conditions. Dust in the reactor cavity may interfere with RCCS function and may detonate.

Fine graphite dust particles are generated by pebble abrasion and friction inside the reactor core and fuel handling system, due to continuous circulation of graphite fuel pebbles during operation. Dust particles carrying fission products may be of particular safety concern in a depressurization accident because of their mobility. Furthermore, since very fine graphite dusts in high concentration can be combustible in some air environments; potential dust ignition/explosion following a depressurization accident is another safety concern for the PBR.

Experimental investigations in the AVR and THTR have been performed on dust generation, deposition, and remobilization associated with graphite pebbles leading to a large pool of knowledge in the industry. This body of knowledge forms the bases for the conception, design, construction, and operation of the HTR-Module.

5.3.1 Graphite Dust Generation Mechanisms

Graphite dusts in PBR are mainly originated from the partially-graphitized matrix material in the fuel pebbles due to abrasion of pebbles in the pebble bed and friction in the piping and valves of the fuel handling system. Abrasion from reflectors plays a relatively minor role. Carbonaceous flakes can also be produced due to ingress of air or water causing structural graphite corrosions, or decomposition of lubricant oil leaking into the primary circuit as in AVR.

5.3.2 Graphite Dust Deposition Mechanisms

The relevant dust deposition mechanisms are: inertial impaction, diffusion, thermo-phoresis turbulent deposition and gravitational settling.

Several successive deposition/re-suspension steps may occur before the dust particle reaches its final location. This effect ensures that all particles except for a small amount currently participating in the deposition/re-suspension equilibrium with the fluid have sufficient adhesive forces to the surface to remain fixed during normal operating conditions. This Darwinian hardening leads to crust-like multi-layer deposits. If the dust particles have been on the surface for a long time at high temperatures, sintering may also contribute to this hardening effect.

5.3.3 Dust Experiments in AVR

Most of the data on the production and characteristics of graphite dust was obtained from the experiments performed in the AVR within a period of about 15 years regularly from 1973 to 1988. Table 5-1 summarizes the characterizations of all the AVR dust measurement results.

Table 5-1: Characteristics of AVR Measurement Results

Parameter	Value
Average Concentration of He-borne Dust, Stationary Conditions	5 µg/m³ (STP)
Scatter Band of Ave. Conc. Stat. Cond. in 16 years	1 - 40 µg/m³ (STP)
Particle Size of the Dust	0.5 – 40 µm
Mean Diameter of the Particle Distribution of the Dust	0.76 µm
Content of Metal in Weight Percent	5 -15 %
Generation rate of dust, educated guess	3 kg/yr
Total mass of dust, end of life (EOL), educated guess	60 kg

The scatter bands of AVR graphite dust size and varieties have been influenced by several unintentional events:

- The ingress of some 100m³ (STP) of air led to damaged fuel element surfaces due to the "peeling effect," which also contributed to the dust production.
- The biggest influence is attributed to the ingress of 120 liters of oil, after which large amounts of fluffy dust were observed on surfaces of the primary circuit.

5.3.4 DEACO Dust Experiments

In the DEACO experiments conducted between 2008 and 2009, two sections of piping taken from the fuel handling system of AVR were cut and carefully examined for the dust contents and activities.

The results from the DEACO experiments are summarized as follows:

- The mechanical decontaminations reveal a strong binding of the dust, as a closed layer, to the inner pipe walls.
- No loose dust could be obtained by hammering on the pipe walls.
- The dose rate and activity distribution reveals the inhomogeneous deposition of dust on the pipe walls.
- The radionuclide distribution in the removed layer material is inhomogeneous, e.g. Eu-154 is located nearer to the surface whereas Co-60 and Cs-137 are mainly located in the mechanically removable (by scrapping) dust layers.
- The removable dust layers have high iron content, as result of a water ingress accident.
- The dust layers consist of mainly of very small particles with an average size smaller than 1µm.
- The de-convolution of the particle size distribution indicates different kinds of particles.
- The mean value of the surface scraped-off dust mass was 2.36mg/cm². Given the surface areas of the steam generator of 1762m² and of the pebble bed of 1130 m², this yields the estimated total mass of dust amounts to 70 kg.
- The mean value of the average thickness of the dust layer is 16µm.

5.3.5 THTR Dust Experience

Graphite dust was not a particular issue for THTR-300, since it had dust filters built into the helium purification system, fuel handling system, and moisture monitors. Nevertheless, in the THTR-300 a graphite dust layer with a thickness of about 5 to 10µm was found on all metallic surfaces, after an equivalent full-power operation time of 16 months.

5.3.6 Estimated Dust Generation in Large HTRs

In the large PBRs such as HTR-Module, frictional forces in the active core and fuel handling system are about an order of magnitude larger than in AVR, due to inverse Helium flow, greater pebble recirculation flow rate, and longer fueling pipes; therefore, dust production in the HTR-Module will increase dramatically compared to AVR. However, all future HTRs can be supplied with a dust filter in the fuel handling system to filter out fine dust particles.

In a rough estimation by up-scaling on the base of the respective values of AVR data, the following dust generation results for the HTR-Module have been obtained:

- The total mass of dust generated per full-power year (FPY) is 22.7 kg/FPY.
- The total mass of dust is 727kg after 32 full power years.
- The mass of Helium-borne dust at steady normal operation is 210mg.
- The mass of Helium-borne dust available for release during the depressurization phase is 45g.

5.3.7 Radionuclide Adsorption on Dust

The total amount of activity carried by the dust in a PBR depends on the fuel performance. In general, two mechanisms can be identified that lead to contamination of dust:

1. For dust particles produced due to abrasion of fuel elements, any radionuclides present in the fuel element matrix will lead to the same specific activity on the dust.
2. Volatile fission products in the coolant can be adsorbed by dust particles that are either circulating with the helium or deposited on the metal surfaces of primary circuit and fuel handling system.

The following results were obtained concerning fission product releases in the VAMPYR experiments performed at AVR:

- Dust is deposited primarily on the surfaces or in the wake flow areas of the primary loop under steady-state operating conditions.
- The radionuclides found in the primary loop were largely attributable to initial defective fuel particles and uranium contamination in fuel element matrix graphite.
- Graphite dust of the AVR primary circuit also contained small but radiotoxic relevant quantities of actinides (Pu-241, Am-241), mainly caused by pebble rupture and destruction of coated particles from the shell-type fuel pebbles of initial core.

5.3.8 Graphite Dust Remobilization

A number of dust remobilization experiments were performed at AVR in 1986, initiated by a quick increase of the blower speed from 1500 rpm by a factor of 2 or 2.5. Results of the "Dust Remobilization" blower transient experiments in AVR are:

- The maximum concentration of Helium-borne dust (final blower speed=4000 rpm) was 1050 $\mu g/m^3$ (STP), and the half-life of dust depletion time was 80 minutes.
- The maximum concentration of Helium-borne dust for the slower transient (final blower speed=3000 rpm) was 280$\mu g/m^3$, and the half-life of dust depletion time was 31 minutes.
- There was a strong dependence of remobilized dust concentration on peak flow velocity.
- The maximum remobilized dust concentration was at least two orders magnitude greater than the normal dust concentrations.
- The total mass of remobilized dust was less than one-tenth of one percent of the total dust deposit inventory.

5.3.9 Impact of Dust Remobilization on HTR-Module

In the Depressurization Phase of a primary system break for the HTR-Module, the calculated maximum gas flow rate out of each break end is 15kg/s. Since the flow rate of the helium in the primary circuit is 85 kg/s during normal operation, this break flow rate results in an increase of primary gas flow rate by about 20%, which is within the range of AVR remobilization experiments.

5.3.10 Combustion of Dust Cloud in the Reactor Cavity

The fact that fine graphite dust is combustible under certain conditions has given rise to the question whether or not dust explosion is possible as a result of the depressurization phase of a primary system break. Experiments conducted in support of the ITER project that investigated combustion of fine graphite dust in air, though produced by different mechanisms, provide some useful insight. These experiments indicate that for a fine graphite dust size of 4μm diameter, combustion occurs at a minimum concentration of 70g/m^3 in air. However, the maximum re-mobilized dust concentration for the HTR-Module is on the order of only 1050$\mu g/m^3$ following the depressurization phase of a primary system break. This is more than three orders magnitude smaller than the ignition threshold. Therefore, it can be concluded that dust combustion/explosion will be quite impossible in HTR-Module.

5.3.11 Dust Cloud Interference with Radiative Heat Transfer to RCCS

The absorption and scattering of thermal radiation due to presence of particulate or dust in the reactor cavity atmosphere can play an important role in the overall energy transfer between the RPV and RCCS, especially under a depressurization accident scenario. However, the empirical attenuation coefficient of thermal radiation obtained for dilute cloud of pulverized coal and ash shows that the dust effect is negligible in the ranges of dust particle size and concentration of the HTR-Module.

5.3.12 Graphite Dust Assessment Conclusions

Based on the experimental and analytical results conducted in the AVR, THTR, and others, the following assessments can be concluded on the dust issues for the HTR-Module:

- The operations of AVR and THTR demonstrated that dusts did not cause problems affecting the reliability of PBR systems.
- The AVR provided a valuable experimental database on dust issues, including steady state conditions and dust remobilization in accident conditions.
- The dust at inner surfaces of the primary circuit forms a closed layer with strong binding on the surfaces.
- The estimated amount of remobilized dust in the primary circuit of the AVR in a depressurization accident supports an enveloping value of 1kg assumed to released in the safety analysis of the HTR-Module.
- Dust explosion scenario in the reactor cavity as a result of a depressurization accident in HTR-Module is unrealistic due to the extremely low dust remobilization concentration.
- Dust cloud effect on radiative heat transfer between the RPV and RCCS is negligible.

In conclusion, the graphite dust generated in HTR-Module poses no real safety risks during normal operation and following a DBA.

5.4 Impact of Broken/Lost Pebbles

Issue: Broken pebbles may become lodged in coolant holes at the bottom of the core and may disrupt pebble flow. Fuel particle failure may result from high burnup and fluence.

Early AVR fuel experienced significant breakage that was linked to the particular design of the fuel pebbles. These pebbles were formed of a 1cm graphite shell into which was formed a 4 cm fuel and carbon matrix inner pebble. During operation, differential dimensional change lead to fracturing of the shell from the inner fuel pebbles, which were small enough to become lodged in the lower reflector flow holes. Subsequent changes in fuel design have eliminated this pebble failure mechanism.

Additional lessons learned from the operation of AVR and THTR have shown that broken pebbles can be kept to a very low level through:

- Refinements in FHS to remove sharp edges and pinch points that can fracture pebbles
- Removal of control rods to reflector to eliminate the major cause of broken and damaged pebbles in the THTR reactor.
- Refinements in pebble design and manufacture to enhance the mechanical strength and resistance to failure of the fuel pebble itself.

A modification has been recommended in the graphite core support structure, where radial elastic support of the core bottom is added by means of spring packs. This prevents the formation of inadmissible gaps between the individual columns due to thermal differential expansions, which might cause jamming of fuel pebbles or fragments. A second modification recommended is that the bottom graphite reflector structure be axially supported by columns fixed to the coolant mixing plenum below the core bottom, to prevent shifting of the bottom graphite reflector blocks and narrowing or widening of coolant flow channels. Changes to the lower core supports flow holes can minimize chances of pebble pieces becoming stuck.

In the AVR, as in other PBR designs, low fluence in the core bottom limits stuck pebble exposure and was not shown to result in excessive particle failure. In addition, there were no indications that broken pebbles in the AVR core had in any way impacted the pebble flow behavior in the core.

The broken and lost pebbles issue is not one of safety; rather it is an economic issue that has already been addressed by the HTR-Module design.

5.5 Proliferation Risk

Issue: Continual pebble circulation and multi-pass fuel management facilitate optimization of pebble plutonium isotopic content. Lack of discrete pebble identification complicates SNM accountability.

Though initial PBR concepts can meet IAEA safeguards standards, review of specific reactor design and protocols is required for final conclusions to be made regarding the ability of the PBR concept to address the Generation IV goal of improved proliferation resistance.

The PBR concept has some inherent features that make proliferation of fissile materials from this type of reactor difficult. These include, for example, the very large quantity of pebbles needed to obtain a Significant Quantity of Special Nuclear Material. PBR core design characteristics make diversion of such large quantities of pebbles both difficult to achieve technically and very noticeable from a fuel supply standpoint. Nevertheless, it would be imprudent to rely too much on these features to establish the overall proliferation resistance of the PBR. The recirculating nature of the PBR online refueling scheme allows the opportunity to tailor the isotopic content of pebbles removed from the reactor. In addition, the lack of item accountability complicates tracking and control of SNM.

Though inserting target pebbles heavily loaded in natural or depleted uranium and retrieving after only one pass in the reactor for obtaining plutonium with high fissile content is feasible, it is very easily detectable long before obtaining a Significant Quantity of fissile material: the fresh fuel supply must be drastically increased to maintain the reactor critical. Therefore, control of the fuel supply can be used, in order to be able to force a diverted reactor to stop long before accumulating a significant quantity of plutonium.

Concerning the existence of a robust safeguards approach, no definitive conclusion can be drawn for the time being, as the approach and the criteria are still in development, but a hybrid scheme has been proposed and seems to offer the required robustness, with sufficient defense in depth features. No showstopper was found concerning its feasibility and effectiveness. This approach involves incorporation of some Item Accountancy features similar to those used in existing LWR power reactors with some Bulk Material Accountancy features similar to those used in fuel cycle facilities, such as enrichment plants. More analyses involving in particular the IAEA and more R&D and qualification work should be performed to confirm this assessment.

The proliferation resistance of a PBR should be built from the integration of safeguards concerns into the details of the design in order to minimize the possible diversion paths and to facilitate safeguards inspections and measurements. It is mainly by taking into account proliferation concerns in the details of the design and by developing advanced safeguards measures that a robust safeguards approach can be adapted to this type of reactor.

The PBR design is not significantly more attractive than other types of reactors around the world in terms of quality of the fuel fissile materials that can be diverted from it. Moreover, using a PBR may not be the optimal solution for production of fissile materials for nuclear explosives. As with other risk assessment activities, the proliferation resistance of the PBR concept should be viewed in light of similar conclusions and drawn from analysis of other Generation IV concepts.

5.6 Shutdown Margin Adequacy

Issue: During plant startup, sufficient shutdown margin may not be available to reach cold shutdown conditions

The available shutdown margin for high temperature pebble bed reactors is an issue that must be addressed by the core design. The core geometry and moderator and fuel reactivity temperature coefficient play an important part in determination of sufficiency of available shutdown margin.

HTR-Module core design and power level indicates that sufficient shutdown margin can and has been engineered into the design this PBR core. This has been accomplished without the use of in-core control rods.

The problem of core recriticality is at low temperature (< 50°C) early in the life of the reactor, during the "running-in" period. This prevents the core from being cooled down to ambient temperature for maintenance without re-criticality. However, this may not be an operational issue for the HTR-Module core because the on-line refueling capability drastically reduces the need to reduce temperatures to these low levels. Available absorber elements that can be introduced into the core, and the possibility of core full or partial unloading, will allow low temperatures to be reached if necessary.

From a licensing standpoint, required available shutdown margin must be addressed with the NRC and must be considered in detailed PBR core designs. Overall, the safety benefits of the PBR reactor concepts including HTR-Module design characteristics of: a) large negative temperature coefficient, b) large heat capacity, and c) no coolant phase change, even in a startup accident should far outweigh the low temperature shutdown margin design issue.

5.7 Online Refueling

Issue: Online refueling positive and negative impacts on overall plant availability need to be considered.

Online refueling is a specific feature of the PBR that potentially allows particularly high availability by eliminating outages for refueling. The following necessary conditions to meet that expectation, however, deserve particular attention:

- The reliability of the fuel handling system shall be sufficiently high so as not to cause significant forced outages for maintenance or repair
- The expected performance of the fuel handling system (number of pebbles removed from the core per day) shall be achievable

The question whether these two conditions can be met is the focus of this issue, because experience feedback does not provide an immediate answer.

5.7.1 Fuel Handling System Reliability

Experience feedback of AVR and THTR constitutes the main information source regarding PBR fuel handling systems reliability.

Forced outages caused by fuel handling system unavailability represented 3% of the AVR operating time. This number would need to be reduced to reach high availabilities. This objective appears reachable for two reasons:

- The issues encountered in the past, particularly helium Tribology issues have been solved by redesign of the affected components.
- The actual AVR fuel handling system unavailability was 12.8%, of which only 3% caused forced outages. This shows that a large fraction of fuel handling system malfunctions can be addressed without power supply interruption

In THTR, fuel pebble blockages in the fuel handling system were a recurrent problem at beginning of operation. These were resolved through update of the system design so that, during the last period of operation, the THTR fuel handling system operated normally at the expected circulation rate.

Overall, the outages of the AVR and THTR caused by their fuel handling systems should be considered as inherent to operation of a prototype facility, but not as intrinsic to the PBR fuel handling system concept. Careful consideration of lessons learned, as well as extensive qualification testing, should therefore largely support highly reliable operation of a new PBR fuel handling system.

5.7.2 Burnup Measurement Performance

In order to support required pebble recirculation rates, the time available for the burnup measurement system to decide whether a pebble has reached the burnup target or not is limited to 10 seconds in the HTR-Module, with a 5% relative statistical error.

Modern consensus for this measurement is to use high precision spectroscopy to determine the ^{137}Cs content of the pebble, as a signature of its burnup. The precision of the measure increases with increased decay time, measurement time or pebble burnup. However, performance of the detectors, computation speed and software optimization, and the design of the whole system have an impact on the performance of the measure and should also be taken into account in a detailed comparison.

Overall, it appears that the HTR-Module burnup measurement speed requirements are faster than those of available technology and that the qualification of this system would require detailed evaluation of the past experiment in the DIDO test rig, if not additional R&D work. However, it should also be considered that in case the required speed can not be achieved, the use of multiple burnup measurement systems in parallel does not appear as a significantly costly or complex solution. Increase of the decay time or decrease of the total number of passes of the pebbles through the core to relax the measurement time constraint may also be considered.

5.8 Tritium

Issue: Permeation of tritium through metallic heat exchanger tubes may impact downstream process heat applications.

The mechanisms by which Tritium is produced in the HTGR are well understood, as are the mechanisms of tritium transport and permeation of metallic barriers. These phenomena are not unique to the PBR design concept.

Results from out-of-pile experiments and measurements in the AVR fuel pebbles have shown that most of tritium fission products are retained inside the intact TRISO fuel particles. On the other hand, tritium produced in the graphite matrix or reflector due to impurities can rapidly diffuse through the graphite components into the coolant, or vice versa through chemical adsorption process.

Most impurities including tritium in the coolant can be removed by the helium purification systems provided in the primary cooling system. There is a small amount of tritium that can be transported to the process side by permeation through the heat exchanger tubes. Even though tritium permeability through steam generator tube metal increases with temperature, it is reduced by up to two orders magnitude with the buildup of thin oxide layer on the surface of metal during normal operation. The HTR-Module design uses the helium purification systems and an indirect steam cycle using steam reboilers before subsequent process heat applications to reduce tritium transfer.

Although tritium transfer mechanisms are understood and are expected to be relatively minor, the associated limits on transfer of tritium to the supplied process have not yet been clearly established by US regulators. This remains a technical and licensing challenge at this time.

Tritium transport must be considered during detailed design activities in light of specific product contamination requirements for those processes that are to be supplied with PBR-generated heat.

6.0 SAFETY ASSESSMENT

6.1 Approach

The safety assessment is a limited evaluation of the PBR design concept using results of existing safety analysis information and potential approaches to plant safety to evaluate the viability of the PBR design and safety concept to meet the current and expected future regulatory requirements.

The bases for the PBR safety assessment are the AREVA HTR-Module design and safety analysis results which were developed in Germany in the late 1980s.

Adjustments to the referenced plant design could be considered based on HTGR design experience since the HTR-Module was first developed to meet the NGNP requirements. The pertinent NGNP requirements are reactor outlet temperature of 750°C or greater, electricity production, and heat for other process applications.

The PBR Scoping Safety Study provides an assessment of the PBR safety case. This assessment is based on the original HTR-Module accident analyses. New analyses are not within the scope of this work. The safety assessment includes identification and evaluation of the PBR plant safety concerns and discussion the expected outcomes for each major accident sequence including an evaluation and discussion of expected dose (using original HTR-Module source terms) at the site boundary (about 400m) for accidents with dose releases based on accepted US dose calculation methodology.

Steps utilized to develop this Scoping Safety Study report are:

1) Review previous PBR safety assessments and analyses – the German HTR-Module safety analysis report was used as the bases of the assessment.
2) Evaluate PBR safety characteristics in light of current technology – the safety characteristics of the HTR-Module are reviewed in light of the current regulations, considering that limited HTGR safety regulations exist in the US regulations, comments are nevertheless made concerning potential acceptability of the HTGR safety case in the current and anticipated US regulatory environment for the new and advanced reactor concepts.
3) Identify and evaluate major safety issues and expected outcomes – the scoping safety study includes identification and assessment of the PBR technology issues (identified and studied in the PBR Technology Readiness report) that impact plant safety and provide comments on the anticipated outcomes for major accident sequences.
4) Expected dose calculation - the study includes an evaluation and discussion of expected dose at a plant site boundary for those accident sequences that have dose consequences. This includes calculation of site dose consequences using US NRC accepted methodology for an exclusion area boundary of 400m using previously calculated HTR-Module source terms.

6.2 Relationship to NGNP Licensing

The Next Generation Nuclear Plant (NGNP) licensing approach is defined in the Augusts 2008 Report to Congress. The NGNP Project has adopted the 10 CFR 52 Combined License (COL) application process, as recommended in the Report to Congress as the foundation for the NGNP licensing strategy. This approach is considered as the most expedient means of obtaining regulatory approval based on HTGR technology as applied to the NGNP Project.

The NGNP demonstration plant will be a licensed commercial High Temperature Gas-cooled Reactor (HTGR) plant capable of producing electricity and high temperature process steam for an industrial application. As part of the pre-application communication with the NRC staff for establishing HTGR regulatory requirements a series of white papers are submitted. These white papers identify and address key HTGR generic issues of the COL priority licensing topics. Through these white paper reviews the NGNP licensing and safety basis will be established.

The pre-application interactions with the NRC through a series of white papers addressing and resolving COL priority licensing topics and 10 CFR 52 COL application process are expected to provide schedule advantage for licensing an NGNP reactor concept while providing consistency with Commission policy guidance on the use of probabilistic risk information and insights.

The NGNP licensing and regulatory requirements will be based on a risk-informed and performance-based technical approach that adapts existing NRC LWR technical licensing requirements in establishing comparable NGNP reactor design specific technical licensing requirements. This approach uses deterministic engineering judgment and analysis, complemented by probabilistic risk assessment (PRA) information and insights, to establish the NGNP licensing basis and requirements.

The HTR-Module safety and licensing reviews was performed in late 1980s in Germany. The German licensing review process started with the LWR regulation including internal and external events. The regulations were also adapted to accommodate the HTR-Module specific design requirements, i.e. radionuclides containment strategy. Although the HTR-Module was never licensed or built, it reached advance stages of safety and licensing reviews before the German nuclear ambitions were curtailed due to external factors. The HTR-Module regulatory interaction resulted in a series of licensing basis events that were analyzed and the results of which were reported in the HTR-Module Safety Analysis Report (SAR). The event scenarios were subsequently validated by the design specific PSA (probabilistic safety analysis similar to PRA in the US terminology) and no additional sequences were identified.

6.3 PBR Safety Characteristics

In this section key PBR design safety characteristics of the PBR technology and the HTR-Module reactor concept are discussed. These safety characteristics are the basis for the safety assessment for the safety design of the HTR-Module. The PBR safety characteristics are categorized as (a) barriers against the release of fission products, (b) inherent safety characteristics, and (c) PBR design features important to safety. The safety assessment continues with a review and evaluation of a select set of HTR-Module internal, external, and safety events sequences.

6.3.1 Barriers against Release of Fission Products

The HTR-Module has three primary barriers against the release of fission products: (a) the fuel particles, (b) the primary system boundary and (c) the reactor building. Figure 6-1 provides a pictorial representation of particle fuel fission product generation, transport paths, retention schemes and physical barriers for a pebble bed reactor such as the HTR-Module.

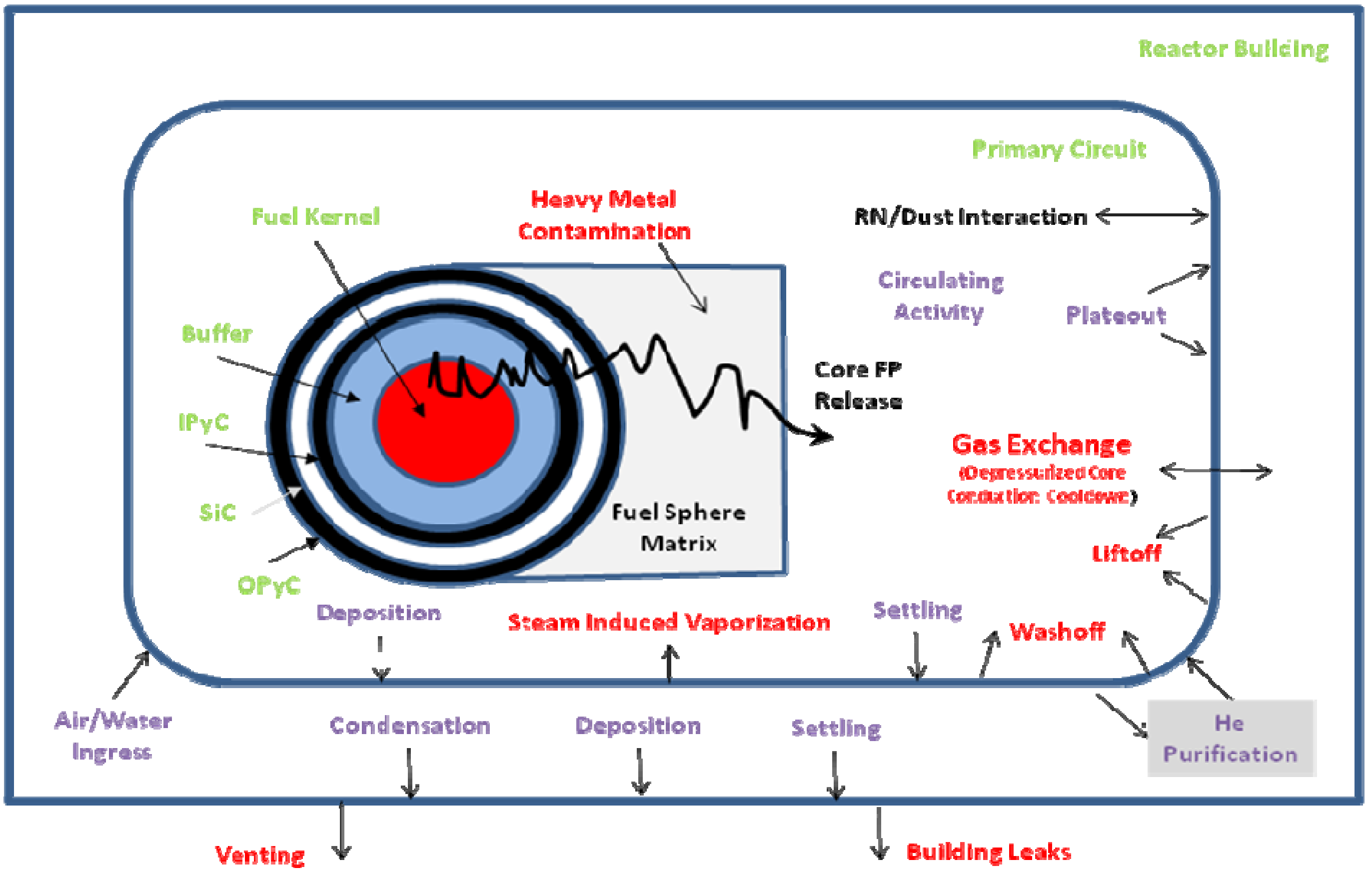
<u>Fuel Particles</u>

Each fuel particle is coated with two high-density layers of pyrocarbon and one layer of silicon carbide. The particles are embedded in a carbon matrix with an unfueled outer zone.

The majority of the radioactive substances produced during nuclear fission are confined within the fuel particle during all operating and accident conditions in such a way that there can be no significant release of radioactivity from these fuel particles with inherent upper limit of ~1600°C, the maximum expected fuel temperature under accident conditions. In particular, for temperatures up to ~1600°C, the silicon carbide layer is so dense that no radiologically significant quantities of gaseous or metallic fission products are released from intact particles.

For design purposes a fraction of coated particle fuels are assumed to have manufacturing or in-service defects. The HTR-Module design basis assumes a fraction of the defective particles could be at the maximum accident temperature of approximately 1600°C. An average of about two defective particles is assumed to exist for each fuel element. This accounts for the manufacturing defects, the burnup induced failures and the core temperature induced failure distribution in normal and accident conditions.

Figure 6-1: HTR-Module Radioactivity Release Barriers and Paths



Primary System Pressure Boundary

Some of the radioactive substances released from the defective particles are retained within the fuel element matrix. The portion that is not retained goes into the primary coolant and is distributed in the primary system. The gas-borne activity in the primary system decreases as a result of radioactive decay, separation in the helium purification system and deposition on the surfaces of the primary system. The primary system pressure boundary thus forms the next barrier against the release of radioactive substances.

Reactor Building

In the event of a break, the radioactive inventory in the helium primary circuit is very small. This includes a portion of the activity deposited on the surfaces of the primary system that could be lifted off and released into the reactor building. Therefore, the HTR-Module design did not place any leak tightness requirements on the reactor building to comply with accident dose limits imposed by the German regulations. This was possible mainly because of the high radioactive retention capacity of the fuel particles. However, to further minimize the impact on the environment, the reactor building is provided with a sub-atmospheric pressure control system and a pressure relief system with filtered release capability.

6.3.2 Inherent Safety Characteristics

The engineered configuration and nuclear design of the HTR-Module is such that even in the event of assumed failures of all active shutdown and residual heat removal systems, the peak fuel temperature stabilizes at approximately 1600°C. Sufficient margin exists between the peak fuel temperature and the temperature at which additional fuel failures are conservatively assumed to occur, and that the core temperature coefficient has enough negative reactivity to shut down the reactor and stop further temperature increase. This safety margin is possible because a nominal temperature differential of approximately 750K exists between the maximum allowable fuel temperature and the maximum operating temperature of the fuel elements in the HTR-Module reactor cores. This is also true even in the presence of accident induced excess reactivity, e.g. water ingress.

Furthermore, residual core heat can be dissipated from the reactor to the surrounding components, structures and surfaces solely through natural physical processes. These include thermal conduction, radiation, and convection. The design choices of low mean power density in the reactor core, the geometric design of the core and the surrounding core internals, and the use of suitable materials, make this inherent (i.e. natural) core decay heat removal characteristic possible.

The HTR-Module also includes active operational systems for residual heat removal. These active systems are normally used to limit loadings on the passive heat removal components and structures. In addition, system, structure, and component (SSC) design margins are selected such that active systems may fail to operate for several hours, the allowable design limits of for the SSCs are not exceeded and the passive engineered safety features will respond in ample time.

The HTR-Module primary systems, core design and material selections reduce the safety requirements on the water/steam cycle and the startup and shutdown systems. Therefore, these systems are designed and operated as purely conventional plant systems.

6.3.3 PBR Design Features Important to Safety

Technical design features of a nuclear reactor that are "important to safety" generally refers to those SSCs that are to be tested to quality standards commensurate with the importance of the safety functions to be performed and that a quality assurance program be established and implemented to provide reasonable assurance that these SSCs will satisfactorily perform their safety functions.

The technical design features of the HTR-Module that are important to safety are listed in Table 6-1.

Reactor fuel elements, core, and internals

The fuel elements feature the following characteristics:

- The retention of fission products within the coated fuel particles in order to fulfill the radiological release requirements for a predetermined plant design,
- Dimensional stability under irradiation to assure transportability of the fuel elements in the fuel handling equipment,
- Mechanical strength to maintain fuel element integrity to assure that the fuel element can be transported and handled and to avoid fission product release due to mechanical damage to the coated fuel particles,
- Transportability even after corrosion due to a depressurization accident (air ingress) or rupture of a steam generator tube (water ingress).

The core design features are that, under all operational and accident conditions, core residual heat can be removed solely by thermal conduction, thermal radiation and natural convection to the surface coolers outside the reactor pressure vessel. The primary heat transport system can be used as the active residual heat transport system. The core and its geometry are so designed such that the reactor can be shut down by the insertion of neutron absorbers in the reflector channels surrounding the core. Because of the core design, the total temperature reactivity coefficient is sufficiently negative such that the inadvertent withdrawal of all reflector rods is mitigated solely by primary gas circulator trip. Furthermore, the core positive reactivity effect caused by a water ingress induced by design bases accident is bounded by the inadvertent withdrawal of all the reflector rods accident. The core height is selected such that un-damped axial xenon oscillations are not possible and thus are ruled out by design.

Table 6-1: HTR-Module Technical Design Features Important to Safety

Technical Design Features
• Reactor fuel elements, core, and internals.
• Primary circuit components
• Confinement envelope
• Decay heat removal
• Helium purification system
• Fuel handling and storage
• Emergency power supply
• Reactor protection system
• Remote shutdown station
• Controlled areas
• Nuclear classification and quality requirements

The HTR-Module ceramic core structure, consisting mainly of side, bottom and top reflectors, surrounds the pebble bed core and also has the task of reflecting neutrons leaving the core back into the pebble bed.

The cylindrical part of the metallic core barrel is constructed of individual courses. At various levels in the core region and above the top reflector, azimuthally guides prevent deflection of the individual stacks of the side reflector.

Two independent and diverse reactor shut-down systems, the reflector rods and the small absorber ball shut-down systems are provided. The absorber elements are inserted into the side reflector for control and shut down of the reactor. The shut-down systems are designed and arranged so that, on demand, the absorber rods drop into their most effective position solely under the force of gravity. Only the first shut-down system is controlled by the reactor protection system, while the second shut-down system is manually actuated, when needed.

The neutron flux measuring instrument is an ex-core instrumentation system in which probes in probe guide tubes are distributed radially and axially in the concrete structure of the reactor cavity.

Primary circuit components

The HTR-Module power plant consists of two reactor facilities having a nominal thermal power of 200 MWt each and sharing some of the same systems.

Referring to Figure 2-3, each reactor consists of:

- The reactor pressure vessel with core, core internals, shut-down systems, and systems for feeding and discharging fuel elements,
- The gas duct pressure vessel with hot gas duct,
- The steam generator with tube bundle and a primary gas circulator.

The reactor, the heat source of the plant, is located at a higher elevation than the steam generator in this system. Natural circulation in the primary system is restricted in such a way that unacceptable temperatures cannot occur in the metallic portions of the system on loss of primary circulation.

Confinement envelope

The confinement envelope of the HTR-Module acts in conjunction with other barriers to the release of radioactive substances to minimize the radiological impact on the environment caused by the accident conditions.

The following features fulfill confinement functions in the HTR-Module:

- Reactor building,
- Secured sub-atmospheric pressure system,
- Building pressure relief system, HVAC systems isolation.

Decay heat removal

The decay heat of the HTR-Module is determined by:

- Directly formed fission products
- Neutron capture in the fission products
- Decay of the actinides

In normal operation and during anticipated operational occurrences, the reactor is cooled down for lengthy outages through the non-safety related main heat transfer system. Additionally, under normal operating conditions, the cavity cooler installed in the reactor cavity serves as a heat sink for the heat dissipated by the reactor. The cavity cooler also protects the concrete structures from reaching unacceptably high temperatures.

Helium purification system

During normal plant operation the helium purification system removes gaseous contaminants, dust and other particles from the helium coolant. In the event of a steam generator tube break, the helium purification system removes the in-leakage water and any corrosion products from the primary system.

Fuel handling and storage

This system handles new, spent and depleted fuel elements. Fuel elements are constantly added to and removed from the core during operation. Both reactors share one charge station for new or used fuel elements and the same systems for storing spent or partially depleted elements.

Emergency power supply

The emergency power generation and distribution system supplies energy to those electric loads that are necessary to maintain plant parameters within predefined operating envelope (i.e., safe conditions) in the event of loss of auxiliary power. The HTR-Module power plant is designed so that the design range load requirements are also met during long-term loss of both auxiliary power and emergency power supplies.

Reactor protection system

The reactor protection system monitors and processes essential process variables in order to detect malfunctions and accident conditions, and automatically initiates protective actions. Each reactor is equipped with a dedicated protection system that is separate and independent from that of any other reactor. Under accident conditions, protective actions are thus initiated only in the reactor concerned. Accidents that affect both reactors are detected separately by each reactor protection system.

Remote shutdown station

The remote shut-down station monitors the plant on loss of control room function. It is housed in the reactor building and is protected against external events and has unlimited accessibility. It is possible to trip the small absorber ball shut-down system from the remote shut-down station; this system ensures long-term sub-criticality. No further actions are initiated from the remote shut-down station.

Controlled areas

The controlled areas of the HTR-Module power plant include:

- Reactor building, except remote shut-down station and its entrance
- Reactor building annex (area of closed cooling water systems)
- Reactor auxiliary building (most portions)
- Spent fuel store (set-down positions; central truck entrance only temporarily when handling shipping casks or for inspection)

Nuclear classification and quality requirements

The systems and components differ in functional and safety-related importance for the overall plant and are classified accordingly. This basically applies to

- Pressure and activity-carrying systems
- HVAC Systems
- Hoists and cranes
- Steelwork items (system and component-specific)

The quality requirements relating to integrity and operability are graded according to nuclear classification.

6.4 Events and Consequences

6.4.1 Internal Events

The HTR-Module is design to protect the plant SSCs from the internal events listed in Table 6-2.

Table 6-2: Internal Events Evaluated for HTR-Module

Internal Events
• Postulated failure of pressure retaining components or components with rotating parts
• Fire
• In-plant explosions
• Dropping of heavy loads

Pressure-retaining components and components with rotating parts are designed to withstand all expected loadings with adequate safety margins, taking into account foreseeable changes in the material properties during the plant's service life. They are fabricated from materials that are suitable for the intended purpose and are manufactured, assembled, installed, tested and operated within the scope of an extensive quality assurance program.

Regardless, the design of the SSCs for the HTR-Module power plant is based on postulated failure of pressure-retaining piping and vessels and components with rotating parts, to the extent that the consequences of failure have a bearing on the safety objectives. Therefore, protective measures against postulated failure of pressure retaining components or components with rotating parts are physical separation or suitable physical arrangement, design for resulting loads, and features for deflecting or retaining missiles.

The HTR-Module fire protection concept is based on a combination of structural and equipment-based features and administrative measures. Adequate consideration is given to the aspects of reactor safety and the protection of persons and property.

Because of the limited use of hazardous materials, the gas mixtures selected, the concentrations that might form and the design boundary conditions selected, no special precautions are taken to protect the plant against in-plant explosions.

The reactor building crane is designed to comply with very stringent requirements such that the dropping of a load onto safety-related systems with unacceptable consequences is considered not credible.

6.4.2 External Events

The HTR-Module is designed to protect against the external hazards and loadings listed in Table 6-3.

Table 6-3: Design of Power Plant Building against External Hazards(1)

	Design Basis Earthquake (3)	Aircraft Crash	Explosion Blast Wave
Reactor building	X	X	X
Reactor building annex	X	-	-
Reactor auxiliary building	(2)	-	-
Switchgear and emergency supply building	X	-	-
Cable ducts	X	-	-
Secured induced-draft cooling towers	X	-	-

X means considered

1) Coincidental loads are taken into account in the analyses. They include lightening, wind, storm, snow, rain, hail, high and low water, and hazardous gases.

2) Seal structure in ground and the main load-bearing structures supporting the seal structure; the other main load-bearing structures are designed in accordance with DIN 4149 to be stable at the intensity of the safe shut-down earthquake.

3) Consequential loads are taken into account in the analyses for safe shut-down earthquake.

The plant features designed to protect against these external hazards include a combination of equipment-based, structural and organizational protective measures that utilize the characteristic safety features of the HTR-Module power plant.

The plant is designed to prevent external hazards from disrupting plant operations or affecting safety-related plant equipment. Because any impairment of the plant results in a disturbance to system behavior or system conditions then no additional signals, other than the existing protection criteria of the reactor protection system are needed to actuate equipment required for external hazards.

To protect the environment from a release of radioactive materials, the following safety functions are assured to be operational during and after the above external hazards:

- Reactor shut-down and long-term subcriticality
- Residual heat removal
- Limitation of the radioactive release

6.4.3 Safety Events

Nuclear power plants are designed for safe operation based on the concept of multiple layers of protection from accidents and their consequences. For the HTR-Module the multiple layers of protection are:

1) Accident prevention via a high quality plant and conduct of operations and by control of operational malfunctions. Safety principles and the precautions taken to ensure safety require that the design and construction of the plant meet stringent technical requirements relating to quality.

2) Accident control by designing the plant to withstand a representative spectrum of postulated accidents. The systems used for accident control are primarily used to keep radiological impacts below the allowable limits and to reduce component and system loadings. Additionally, sufficient time is generally available for repairs to be carried out or alternative actions to be taken.

The accident analyses evaluated, originally performed for HTR-Module licensing in Germany, were chosen based on events and combinations of events that envelope the possible release of radioactive materials and the loadings sustained by components and systems for that category of events. The event categories and events that were evaluated are given in Table 6-4.

The HTR-Module power plant is designed to permit independent operation of the two reactors. Each reactor has a separate, dedicated reactor protection system that monitors the operating parameters and, in the event of accidents, only shuts down the affected reactor.

The design ensures that, for all reactor accidents, only the affected reactor is shut down automatically, while the other can be kept in operation. All accidents lead to uncoupling of the affected reactor from the overall plant by shut down and secondary-side isolation of the steam generator. Likewise, the steam generator relief (dump) that is additionally performed in the event of tube breaks, does not prevent operation of the other reactor.

Depending on their size, primary-side breaks can cause a short-term pressure build-up in the reactor building. In principle, it is possible to continue operation of the unaffected reactor. The effects of events affecting the overall plant (e.g., emergency power operation or external events) are detected by the reactor protection system assigned to each reactor and cause shut-down of the affected reactor. No safety requirements are placed on the water/steam cycle and the start-up and shut-down circuits. They are designed and operated as non-nuclear plant components.

Table 6-4: PBR Events Evaluated for the HTR-Module

Event Categories	Events
Reactivity	• Withdrawal of Reflector Rods or Small Ball Shutdown Elements • Inadvertent Operation of a Small Ball Shutdown Unit or Reflector Rod • Inadvertent Over-speeding of the Primary Gas Circulator • Maximum Decrease in Cold Gas Temperature • Pebble Bed Compaction Due to Earthquake
Main Heat Transfer Malfunction	• Loss of Auxiliary Power Supply • Loss of Primary Coolant Mass Flow • Loss of Feedwater Flow • Inadvertent Closure of a Main Steam Valve • Main Steam Extraction Malfunction • Process Steam Extraction Malfunctions • Inadvertent Opening of Valves in the Water/Steam Cycle • Turbine Trip
Primary System Depressurization	• Break of a Large Connecting Line between Pressure Vessel Unit & Primary System Isolation Valve (Depressurization Phase & Depressurization Followed by Core Heat-Up) • Break of a Large Connecting Line (DN65) Downstream of Primary System Isolation Valve in the Reactor Building • Break of an Instrument Line and Small Breaks • Inadvertent Opening of a Safety Valve
Air Ingress	• Small Breaks in the Primary Coolant Boundary • Large Breaks in the Primary Coolant Boundary • Large-Scale Rupture of Gas Duct Pressure Vessel and/or Reactor Vessel (Beyond Design Bases Event)
Water Ingress	• Steam Generator Breaks/Tube Rupture
Secondary Side Leaks and Breaks	• Feedwater Line Break • Main Steam Line Break • Steam Generator Breaks/Tube Rupture
Disturbances in Auxiliary and Supporting Systems	• Breaks or Leaks in Primary Coolant Conveying Components - Components outside the Reactor Building • Breaks of a Vessel Containing Radioactively Contaminated Water
Failure of the Auxiliary Power Supply & Unavailability of the Emergency Diesel	• Short-term Failure of Auxiliary Power and Unavailability of Emergency Diesel • Long-term Failure of Auxiliary Power and Unavailability of Emergency Diesel
Other	• Seismic Effects on the Reactor Auxiliary Building • Postulated Transient without Reactor Scram (ATWS) • Failure of Cavity Cooler (Beyond Design Bases Event)

For reasons of availability the HTR-Module design has two completely independent water/steam cycles. These are uncoupled from each other to such an extent that the majority of the anticipated malfunctions can be handled by the controls. Therefore in most cases there is no need to shut down both reactors. In addition to being equipped with a separate water/steam cycle, each reactor is provided with a start-up and shut-down circuit. This allows each reactor to be started up and shut down independently of the water/steam cycle.

6.5 Radiological Dose for Bounding Events

The following HTR-Module events are “bounding events” or enveloping accidents for the overall plant in terms of radiological dose:

- Break of a large connecting gas line between pressure vessel unit and primary system isolation valve,
- Break of an instrument line and small breaks,

- Steam generator breaks/tube rupture,
- Breaks or leaks in primary-coolant-conveying components outside the reactor building,
- Break of a vessel in the waste disposal system containing radioactive contaminated water,
- Seismic affect on the reactor auxiliary building.

Using the source terms obtained from the accident analysis, the radiological doses were calculated for these bounding events and are presented in Table 5.

Dose calculation was performed for the HTR-Module using current US methodologies and practices. Title 10 of the US Code of Federal Regulations (CFR) and accompanying regulatory guidance documents primarily address light water reactor licensing. There is currently no regulatory guidance or regulatory acceptance criteria for HTGR accident evaluation in the US The German HTR-Module Safety Analysis Report (SAR), 1988 Version, was used as the basis for the present radiological assessment.

The radioactivity releases (source terms) for the HTR-Module plant due to design basis accidents are documented in the SAR. In accordance with the German regulations the SAR dose assessments are based on ingestion pathway whereas for the US accident dose calculations must be based on immersion and inhalation.

The recalculation of the doses addressed in this section used the radioactivity released and calculated two-hour immersion and inhalation doses for an exclusion area boundary (EAB). The release of radioactivity was conservatively assumed to be instantaneous.

The atmospheric dispersion assumed a concentration factor corresponding to an EAB of 400 meters. Dose conversion factors are derived from ICRP-30 consistent with the current NRC regulations.

The basic radiological acceptance criteria for the offsite receptors associated with the alternative source term (AST) methodology for light water reactors are found in 10 CFR 50.34(a)(1) with a limit of 25 rem (0.25 Sv) total effective dose equivalent (TEDE). This criterion, however, is used for evaluating potential light water reactor accidents of exceedingly low occurrence probability and low risk of public exposure to radiation. For events with higher probability of occurrence, the acceptance criteria for the offsite receptors are more stringent.

Table 6-5: Summary of HTR-Module Bounding Accident Doses

<table>
<tr><th rowspan="2">Design Basis Accidents</th><th colspan="2">Dose (TEDE) Sv</th></tr>
<tr><th>EAB (2 hour) (1)</th><th>NRC Limit (2)</th></tr>
<tr><td>Break of a Large Connecting Pipe (DN 65) - LBLOCA short-term unfiltered release</td><td>unfiltered: 1.361E-04
(13.61 mrem)</td><td rowspan="5">0.25
0.063
0.025</td></tr>
<tr><td>Break of a Large Connecting Line (DN 65) - LBLOCA long-term unfiltered release with core heat up</td><td>filtered: 2.858E-06
unfiltered: 2.214E-04
(22.14 mrem)</td></tr>
<tr><td>Instrument Line Break Pressure release phase (DN <10)</td><td>filtered: 5.183E-05
unfiltered: 5.782E-05
(5.782 mrem)</td></tr>
<tr><td>Steam Generator Tube Rupture with response of the Pressure Relief System</td><td>filtered: 2.245E-06
unfiltered: 3.464E-05
(3.464 mrem)</td></tr>
<tr><td>Helium Purification System Pipe Break release via stack</td><td>unfiltered: 1.975E-04
(19.75 mrem)</td></tr>
<tr><th colspan="3">Non-Design Basis Accidents</th></tr>
<tr><td>Leakage of Vessel Containing Radioactive Contaminated Water</td><td>Unfiltered: 2.443E-06
(0.2443 mrem)</td><td rowspan="2">10CFR
20</td></tr>
<tr><td>Seismic Effects on the Reactor Auxiliary Building</td><td>Unfiltered: 3.961E-04
(39.61 mrem)</td></tr>
</table>

Notes:

(1) The worst two hour window is used with an atmospheric dispersion factor (X/Q) value of 3.35E-03 s/m^3 corresponding to an EAB distance of 0.249 miles (400 m).

(2) The 0.25 Sv criterion is used for evaluating design basis accidents of exceedingly low occurrence probability and low risk of public exposure to radiation. The criterion for events of moderate frequency is 25% of the 0.25 Sv, or 0.063 Sv. The criterion for events of higher probability of occurrence, the acceptance criterion is10% of the full limit, or 0.025 Sv

7.0 COST ESTIMATE

7.1 Cost Estimate Approach

The cost estimates produced are considered as indicative, pre-commercial estimates with an uncertainly of ±40%. The cost estimates are derived from a previously estimated AREVA GmbH demonstration plant design using an HTR-Module with two 200 MWt reactors.

Nuclear steam supply facility (NSS) costs were derived from the engineering and equipment budgets from the 1991 German work, escalated to 2011 dollars. Construction materials and construction labor for the nuclear steam supply area were derived by factoring such costs from equipment costs, by estimating building volumes and concrete volumes from drawings, and by scaling similar costs from earlier work done for the NGNP Program and other projects. Equipment, construction materials and construction labor costs were derived by factoring using ratios from other nuclear projects.

Energy conversion plant (ECP) and balance of plant (BOP) costs were scaled from other projects based on major equipment performance requirements established by the estimated heat balance. Equipment, construction materials, and labor costs were derived by scaling information from other projects.

The current stage of work for the PBR has no cost input from suppliers and only indicative budgeting derived from other projects.

7.2 Cost Estimate Basis

The base configuration for the PBR Demonstration Plant consists of an HTR-Module with 2 x 200 MWt reactors, a shared control room and other shared nuclear island structures such as fuel storage areas. Each reactor delivers helium to a separate steam generator. Primary steam is combined from the two steam generators and delivered to a single steam turbine generator.

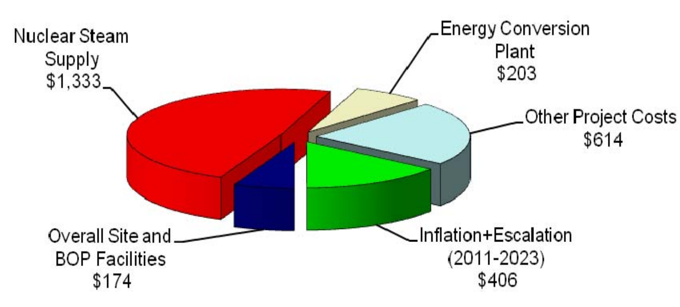
The base configuration for the PBR Commercial Plant consists of the same NSS plant design as the PBR Demonstration Plant improved by experience from a series of projects. The PBR Commercial Plant is defined to be four 2-reactor HTR-Modules on a site, taking advantage of sharing and learning from the other units. The range of applications for the PBR Commercial Plant covers a wide range of possibilities from power generation with some process steam production to all steam production with no steam production. The design described and estimated for this review considers a steam turbine generator with steam extraction and reboilers similar to the PBR Demonstration Plant. It should be noted that as process steam delivery capacity increases and power generation capacity decreases over the range of applications, reduced cost for smaller steam turbine generator and associated equipment is offset by increasing reboiler and high energy piping costs. Therefore, this design and cost estimate for the PBR Commercial Plant is considered representative of a range of applications.

For the purposes of this assessment, it is assumed that the Demonstration Plant will be located on the US Gulf Coast. It will serve to demonstrate a full scale PBR NSS module, an ECP sized for the full thermal output of the NSS and the associated site and BOP facilities. A full scope of costs are addressed, including engineering, nuclear licensing, environmental permitting, fuel, and operations and maintenance through an initial three-year period needed to demonstrate operability and performance.

7.3 FOAK Cost Estimate Results

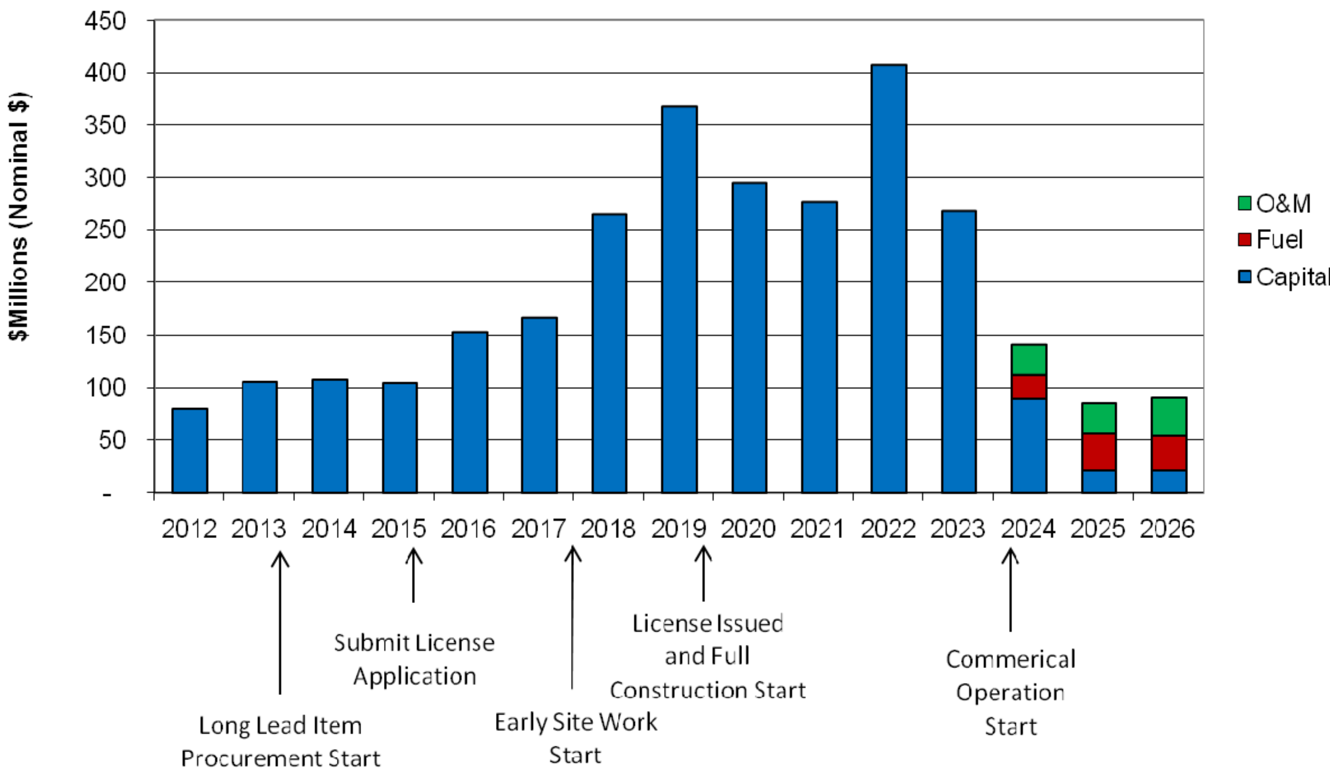
EPC costs for the NSS, ECP, and BOP are presented in Figure 7-1, along with other project costs and inflation. An EPC fee of 5% is included for each of the three EPC contracts. Final design and construction support engineering costs are included in each of the EPC contracts, while preliminary and conceptual design engineering costs are included in Other Plant Costs. These costs are presented in 2011 US dollars, except for the inflation component which converts the 2011 dollars to as-spent nominal dollars.

Figure 7-1: PBR Demonstration Plant Capital Cost Summary (in $millions)



The expected cash flow required for the demonstration plant project is presented in Figure 7-2. During the first three years of operation, generation will be ramped up and the final testing and licensing costs designated as capital costs will be expended.

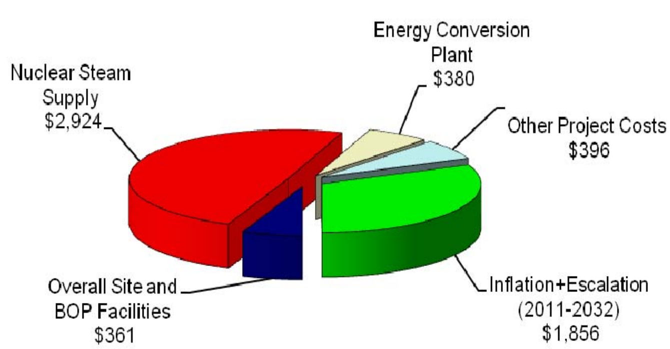
Figure 7-2: PBR Demonstration Plant Cash Flow



7.4 NOAK Cost Estimate Results

The cost results for the PBR Commercial Plant are include the total cost of a 4 HTR-Module facility with eight reactor/steam generators serving two steam turbine generators. These results are shown in Figure 7-3.

Figure 7-3: Summary of Commercial Plant Project Cost (in $millions)



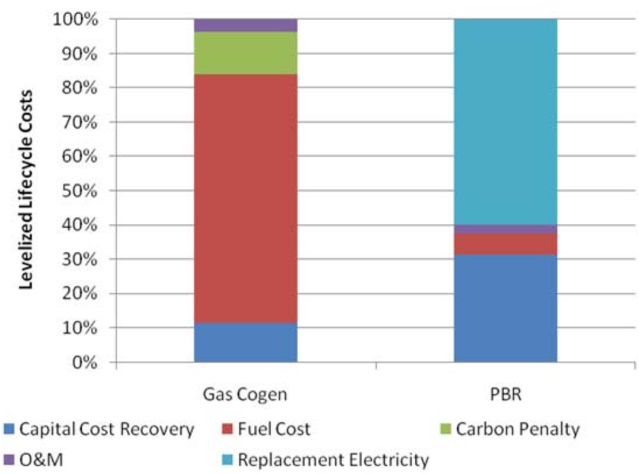
Comparing the overnight capital costs in 2011$ of the PBR Commercial Plant to the PBR Demonstration Plant, the PBR Commercial Plant is 56% less on a per unit thermal energy basis. The PBR Commercial Plant is estimated at $2,539/kWt in 2011$ (equivalent to an average of $1,016M per module), compared to $5,810/kWt in 2011$ ($2,324M per module) for the PBR Demonstration Plant. Economies of scale, sharing between modules, elimination of FOAK costs (such as FOAK licensing engineering, and supply chain development), and learning account for the reduction in cost.

7.5 Economic Competitiveness

The lifecycle economics of the Commercial PBR Plant are compared to that of a natural gas-fired cogeneration plant. Figure 7-4 presents a comparison of the components that make up the lifecycle costs for the PBR Commercial Plant and a conventional natural gas fired cogeneration plant, assuming a 40 year life, with gas price set at the breakeven point of $10.56/MMBtu and a carbon penalty of $25/tonne CO_2. The figure shows that capital recovery is the dominant component of the PBR plant lifecycle cost while fuel is the dominant component of the gas cogeneration plant lifecycle cost.

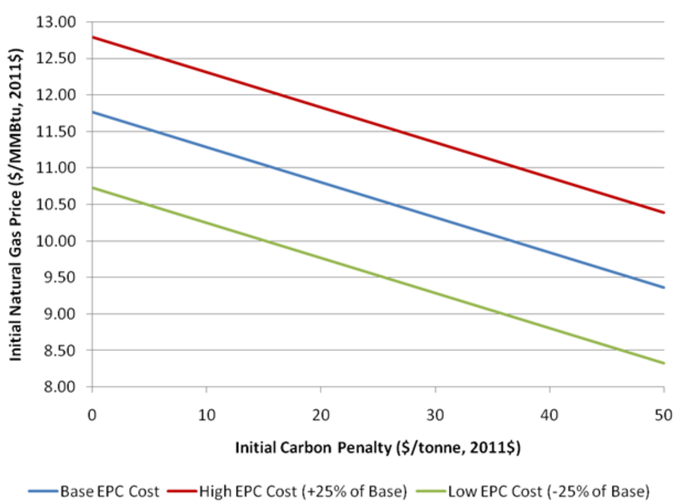
Mature PBR applications can compete with fossil cogeneration units with rising gas prices and CO_2 penalties. Figure 7-5 presents the breakeven natural gas price versus carbon penalty. As the carbon penalty increases, the breakeven natural gas price decreases. Without a carbon penalty the breakeven natural gas price is $11.76/MMBtu and with a carbon penalty of $50/tonne CO_2 the breakeven natural gas price decreases to $9.36/MMBtu.

Figure 7-4: PBR vs. Gas Lifecycle Cost Comparison



Current prices are around $4.50/MMBtu and will have to increase significantly in real terms between now and the start of commercial operation and continue to increase during the plant lifetime for the PBR to be more economical than the gas cogeneration plant alternative. The results are sensitive to variations in power price.

Figure 7-5: PBR Breakeven Gas Price



For higher power prices, the PBR application design can be modified to increase power generation with additional reactors if this is competitive with grid power. The commercial PBR capital costs have a first order effect of the economic comparison, as reflected in the high and low end PBR capital cost cases shown in Figure 7-5. A 25% change in PBR capital cost shifts the breakeven natural gas price by ~$1/MMBtu. With a $50/tonne CO2 penalty and 25% lower EPC costs, the breakeven natural gas price would just over $8/MMBtu.

8.0 PROJECT SCHEDULE (FOAK)

8.1 Approach/Ground Rules

The FOAK PBR Demonstration Plant project schedule establishes a project road map from conceptual design, through construction and startup of the demonstration plant. The schedule identifies various activities and key milestones. Subsequent resource loading of schedule activities provides input for yearly funding profiles starting at the beginning of the design reconciliation task.

In general, the activities presented in the schedule are categorized into project level activities and facility level activities. For consistency with other NGNP-related work, project level activities are identified and grouped in general to be consistent with the Work Breakdown Structure (WBS) provided by Battelle Energy Alliance (BEA)/Idaho National Laboratory (INL). The project schedule integrates the project and facility level activities into a cohesive presentation for the execution of the project.

The schedule of activities for the NSS has been developed with AREVA HTR-Module used as input for scope. The schedules for the ECP and BOP facilities represent conventional scheduling experience.

The Project Schedule integrates the project and facility level activities into a cohesive presentation for the execution of the Project.

The schedule of activities for the NSS has been developed with the project schedule for the HTR-Module schedule used as input and for comparison.

8.2 Assumptions

The schedule bases require the formation of a Public-Private Partnership in September 2011. This partnership will fund the design and licensing prerequisites for construction of the plant. Additionally, the acquisition strategy for each piece of long lead equipment will most likely require material procurement and fabrication long before a final decision to construct (October 2017) is made. To meet the construction schedule, it is assumed that the Partnership will fund the fabrication of long lead equipment in advance of making the final decision to construct.

Limited Work Authorization approval from the NRC is assumed to be in place to support the start of early site work on October 2017. The completion of early site work activities, including mobilization of the civil-structural contractor is needed to permit construction to proceed as soon as the COL is issued by the NRC and the Partnership authorization to proceed is given.

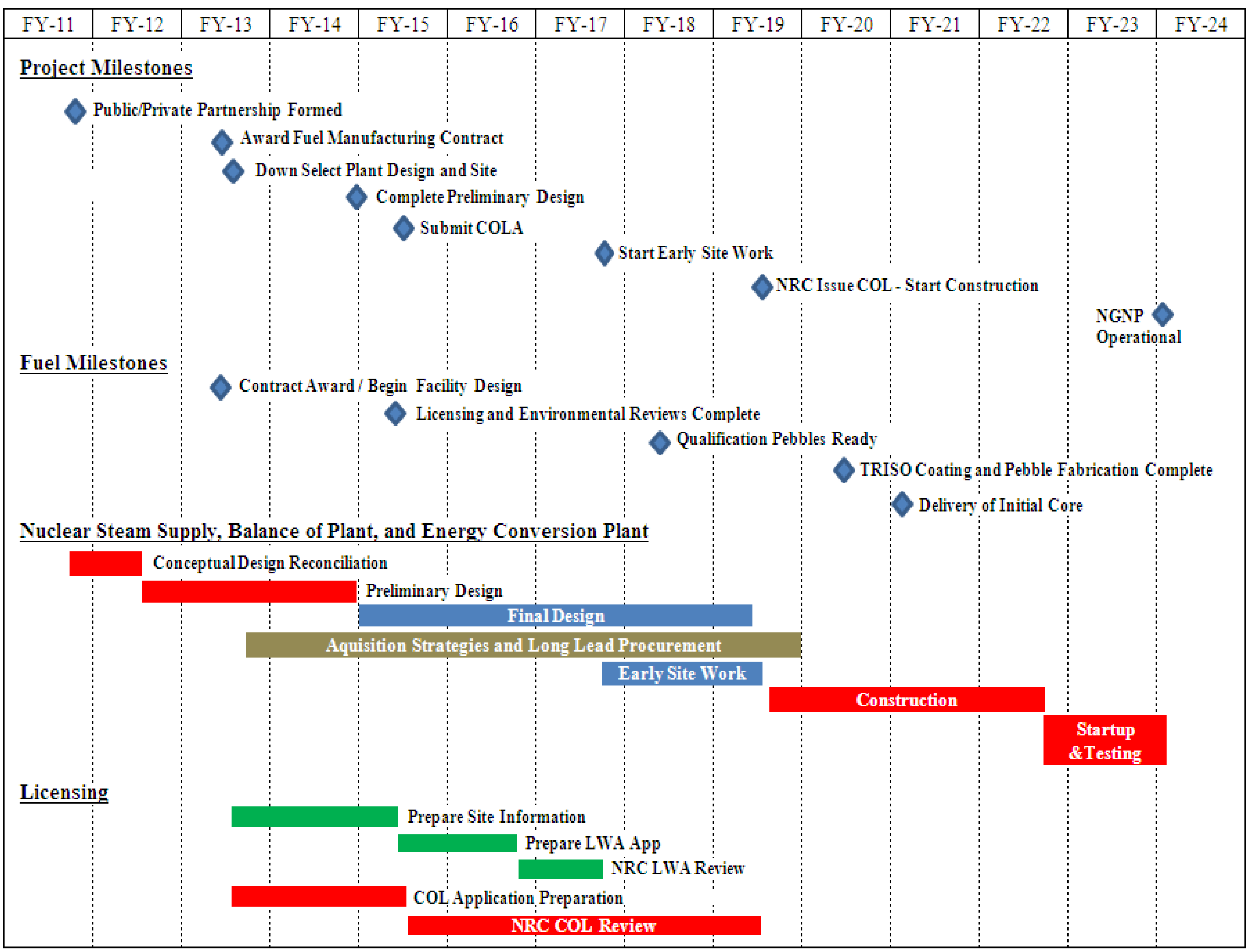
The fabrication and delivery of long lead equipment, such as the reactor pressure vessel will require detailed planning and coordination with perspective fabricators and their suppliers. It is assumed that technology and code compliance issues will not restrain fabrication and delivery of this equipment.

Sub-assembly fabrication on site will be factored into the constructability evaluation and planning process. For this reason, the schedule shows the Acquisition Planning for these long lead pieces of equipment starting as early as practical.

8.3 Results

A summary level schedule is presented in Figure 8-1.

Figure 8-1: PBR Demonstration Project Schedule Summary



8.4 Key Observations

The critical path for this schedule runs through Concept Design Reconciliation, Preliminary Design, Integrated Safety Analysis, COL Application submittal, NRC review and issue of the COL. Critical path continues with construction, startup, and initial operation of the plant. Critical path activities are red on Figure 8-1.

The primary driver for this critical path is associated with receiving a COL from the NRC. The preparation of the application requires site selection and advanced completion of the preliminary design. Implementation of a comprehensive Regulatory Management Plan will be necessary to achieve the COL on schedule.

Upon receipt of the COL from the NRC, Nuclear quality level construction will commence. Construction and startup testing is scheduled to take place over a 5 year period. The successful completion of these activities during this period is dependent on completion of significant early site work activities, long lead equipment deliveries, modular construction and shop testing.

Other potential critical paths could result if any delays occur in the following:

- Fuel Fabrication and Qualification
- Long Lead Item Acquisition
- Early Site Permit Submittal and Review

9.0 FUEL QUALIFICATION AND ACQUISITION

Most past and current pebble bed reactor programs around the world have used UO_2 TRISO fuel as their reference fuel form. A significant experience base exists for UO_2 TRISO fuel, and the potential performance enhancements offered by uranium oxy-carbide (UCO) TRISO fuel, while beneficial, are not mandatory for a pebble bed reactor.

Thus, a strong incentive to explore other options did not exist until recently. The NGNP pebble bed reactor fuel qualification strategy relied heavily on the South African fuel development program. However with the suspension of major fuel development activities by the NGNP pebble bed reactor team, the need for an alternative fuel qualification path must be addressed.

9.1 Fuel Supply Options

Although there is no single, universally accepted ready-to-go solution for providing fuel for a pebble-bed HTGR at this time, there are several options that can be considered.

Option 1- US AGR Program: High quality UCO TRISO particles have only recently been made in the US by B&W. They have been irradiated successfully in the AGR-1 test, but PIE results are not yet available on solid fission product retention, nor are the results from accident conditions heating tests. It is important to note that UO_2 fuel could also be included in the AGR program if deemed advantageous, since the AGR-1 irradiations included this fuel type and indicated acceptable performance.

Option 2 – German Fuel: In principle, you can order an HTGR fuel factory from Germany that would make spherical fuel elements containing high quality LEU UO_2 TRISO particles. This is in essence what was done to support the Chinese fuel program. It may also be possible to order fuel directly from the Germans.

Option 3 – French Program: CEA Cadarache has made UO_2 TRISO fuel and AREVA-CERCA in Romans has made compacts that are presently being irradiated in AGR-2.

Option 4 – South African/PBMR Program: As part of a well funded development program, the Fuel Development Laboratory in Pelindaba, South Africa, has developed high quality UO_2 TRISO fuel. Particles were put into compacts by ORNL and these are being irradiated now in AGR-2.

Option 5 – Chinese Program: China makes good high quality UO_2 TRISO particles and spherical fuel elements completely to German standards and mostly with German equipment.

Though each of these options represents a possible path to production of fuel for the NGNP, there are several considerations that must be examined to determine which is the most appropriate choice to support NGNP deployment.

First, and most important, the selected option must be able to be qualified by the NRC as an acceptable fuel supply that adequately supports the NGNP safety case. Options 1 and 5 are currently the only ones with active, ongoing qualification efforts underway. Option 1 is being conducted in the US and as such, is most compliant with expected NRC requirements. Of the remaining options, Option 4 is the only one with a defined qualification path. The potential for resumption of these qualification activities is unknown at this time.

Other areas that may be considered in further refining the choice of options include the potential for support of advancements in fuel technology and fuel cycle designs, expected ease of implementation in support of the NGNP project schedule, utilization of R&D resources, and security of domestic fuel supply for both the initial NGNP and an eventual fleet of pebble bed HTGRs.

Based on a review of these options, it is concluded that Option 1 represents the best choice for the NGNP. The following section of this report will examine the US AGR Program and identify any recommended changes to optimize it for support of the PBR design.

9.2 AGR Qualification Program

The NGNP Program at INL has established the Advanced Gas Reactor AGR Fuel Development and Qualification Program to address the following overall goals:

- Provide a baseline fuel qualification data set in support of the licensing and operation of the Next Generation Nuclear Plant (NGNP).
- Support near-term deployment of an NGNP by reducing market entry risks posed by technical uncertainties associated with fuel production and qualification.
- Utilize international collaboration mechanisms to extend the value of DOE resources.

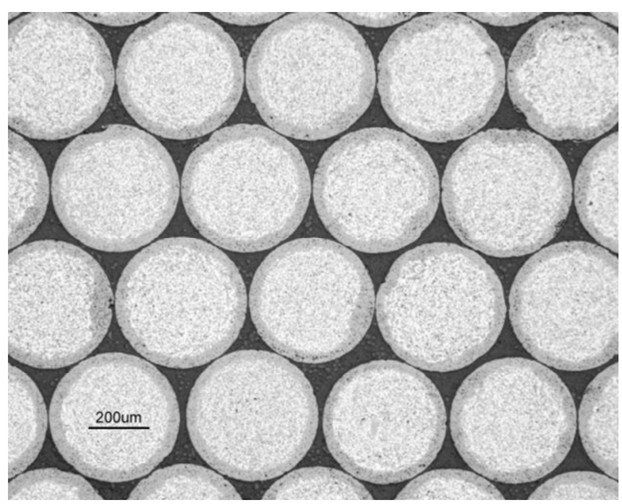
The AGR Fuel Development and Qualification Program consists of five elements: fuel manufacture, fuel and materials irradiations, post-irradiation examination (PIE) and safety testing, fuel performance modeling, and fission product transport and source term evaluation.

An underlying theme for the fuel development work is the need to develop a more complete fundamental understanding of the relationship between the fuel fabrication process, key fuel properties, the irradiation performance of the fuel, and the release and transport of fission products in the NGNP primary coolant system. Fuel performance modeling and analysis of the fission product behavior in the primary circuit are important aspects of this work. The performance models are considered essential for several reasons, including guidance for the plant designer in establishing the core design and operating limits, and demonstration to the licensing authority that the applicant has a thorough understanding of the in-service behavior of the fuel system. The fission product behavior task will also provide primary source term data needed for licensing. An overview of the program and recent progress will be presented.

The baseline fuel for the NGNP AGR test series is a low enriched UCO kernel (425µm in diameter at ~15% enrichment), within a standard TRISO particle. UCO was selected because the mixture of carbide and oxide components precludes free oxygen from being released due to fission. As a result, no carbon monoxide is generated during irradiation, and little kernel migration (amoeba effect) is expected. Yet, like UO_2, the oxy-carbide fuel still ties up the lanthanide fission products as immobile oxides in the kernel, which gives the fuel added stability under accident conditions. The choice of kernel enrichment and diameter were chosen by the program based on the anticipated needs of the prismatic HTGR design. This fuel configuration is assumed to be more limiting than that of the PBR design; therefore, qualification results are anticipated to be bounding and applicable for either of the HTGR designs.

In order to complete fuel qualification activities for the PBR, it is anticipated that irradiation and testing of a full-sized fuel pebble will be required. Based on size limitations, use of an additional test reactor will be required. Perhaps the most reliable solution would be to use the HFR reactor, which has a test rig available for pebble irradiation and which had the recent experience of pebble irradiation. The only alternative that appears to be applicable would be in Russia, where some Chinese test pebbles have been irradiated.

Figure 9-1: UCO Fuel Kernels



9.3 Fuel and Core Design Selection

The current AGR program is focused on qualification of a UCO fuel particle for fairly high enrichment and burnup operation, consistent with the stated needs of the prismatic reactor HTGR design. To support pebble bed reactor fuel qualification within the AGR program, either the current AGR particle design could be utilized as is, or an alternate fuel form from a range of possible UO_2 and UCO options could be added to the program. Various forms of UCO and UO_2 TRISO fuel particles that have been tested and/or produced for the prismatic and pebble bed HTGRs were considered as candidate fuel designs.

The UCO particle, with ~14% ^{235}U enrichment and dimensions consistent with the existing AGR particle was selected as the best fuel for the PBR concept. The reasons for selecting this fuel form were:

- UCO minimizes particle internal pressure build-up. This provides less particle failure and less fission product release.
- UCO provides favorable operating performance and higher burnup than UO_2.
- A separate qualification program for UCO particles is not required as this fuel can be qualified under the AGR fuel development and qualification program. However, the AGR program must be modified to accommodate full spheres.
- Higher uranium enrichment and burnup of UCO provides for better fuel utilization and improved cost and economics.
- UCO has potential for future advancements to support higher power and higher temperature operations.

The importance of the ability to support the fuel qualification needs of both prismatic and pebble bed reactor concepts should not be underestimated. The potential cost savings and improved allocation of resources is clear. What is perhaps even more important to keep in mind is the impact of infrastructure bottlenecks on the ability to support the simultaneous development of two different particle designs. It is not clear that there are enough qualified irradiation, examination, and test facilities available to really support two designs at the same time.

Thus, selecting the UCO particle provides a responsible path to provide a qualified fuel form in the US for the PBR concept in the required timeframe for the NGNP project.

Once the preferred fuel particle has been selected, the complete fuel element design must be defined. This determines the final fuel form that the qualification program must support, and it sets the fuel production requirements that the fuel acquisition strategy will have to support.

The basic dimensions of the fuel element are maintained from the reference HTR-Module. The pebble has an inner fueled region 5cm in diameter with a 5mm unfueled outer shell for a total diameter of 6cm. Having selected the AGR fuel particle design with 14% enriched UCO kernels, the remaining pebble design parameter to determine is the number of particles per pebble.

In order to develop the fuel acquisition strategy, the fuel loading design must also establish the total number of pebbles required and a schedule for their delivery. This includes the number of pebbles required for initial reactor startup as well as the number of replacement pebbles required each year during regular operation.

To precisely determine the fuel loading requirements for pebbles, a detailed core design is required. A variety of factors ultimately must be considered in selecting the final pebble design, including overall core reactivity, pebble reactivity swing from beginning-of-life to end-of-life, pebble average power and peak power, pebble and particle temperatures (average and peak), control rod worth, water ingress reactivity worth, and fuel handling system capacity, not to mention fuel fabrication facility capacity and overall fuel cycle economics. Clearly evaluation of all these factors was not possible in the limited scope of the current PBR assessment. Instead, an interim fuel loading design was selected based on scaling from the reference HTR-Module UO_2 core design, input from core design and fuel experts, and limited scoping core analysis.

The reference HTR-Module UO_2 core design has 7g of heavy metal per pebble and approximately 0.5g ^{235}U per pebble (for equilibrium core new pebble). This requires about 11,600 kernels (500μm) per pebble. For the selected UCO particle with 14% enrichment, the initial fissile content per gram of heavy metal increases, but the fertile content decreases. Moreover, the UCO particle has a significantly higher target burnup (140 GWd/MT) compared to the UO_2 particle (80 GWd/MT). These factors mean that the net reactivity swing for an average pebble will be significantly different for the 14% enriched UCO core than for the 7.8% enriched UO_2 core.

Two simplistic pebble loading assumptions were postulated in an attempt to bracket a reasonable range of potential equilibrium pebble loadings. One assumption was to simply keep the ^{235}U content of each pebble the same as the reference UO_2 pebble. The alternate assumption was to keep the total heavy metal content of each pebble the same. Initial scoping reactivity analyses comparing these two alternatives suggested that the 7g heavy metal case (second alternative) was preferred. More importantly, this alternative provides a more conservative bounding assumption for fuel pebble qualification, since it leads to a somewhat higher packing fraction and potentially a higher initial peak power.

Based on this assumption of 7g heavy metal per pebble, the equilibrium pebble requires 17,800 UCO kernels (425 μm). The kernels are 14% enriched UCO.

The number of new replacement pebbles required per year of operation is determined from the heavy metal content per pebble and the target burnup. The plant is conservatively assumed to run at 200 MWt per reactor with 100% availability. This high availability is a necessary conservative assumption for a plant with on-line refueling. For 140 GWd/MT and 7g heavy metal per pebble, the annual replacement pebble requirement is 150,000 new pebbles per year for a two reactor HTR-Module plant.

A somewhat different approach was used to determine the required pebble inventory for initial plant startup. Because the pebble bed reactor is designed for a relatively narrow reactivity swing, a special approach is followed for initial criticality and the run-in process leading to equilibrium core operation. Upon initial startup, all pebbles have no burnup and no fission products. This requires a special mixture of unfueled and partially fueled pebbles to be used during this period. Over time, the initial pebbles are gradually replaced until eventually the core is fully fueled with equilibrium pebbles at various stages of burnup.

In the reference UO_2 HTR-Module core, the fueled pebbles in the initial core have reduced enrichment, which results in 0.27g ^{235}U per initial fueled pebble. Since the initial core has no burnup or fission products, the approach taken to estimate the required UCO pebble loading was to preserve the initial pebble ^{235}U content. Assuming a reduced enrichment of 8%, the heavy metal loading for the initial core UCO pebbles must be reduced to about 3.4g. Therefore, 8,600 UCO particles are required for each initial pebble. With lower enrichment and reduced packing fraction, these pebbles will be bounded by the qualification of the equilibrium core pebbles.

The total number of initial core pebbles is 360,000 per reactor or 720,000 for a two reactor HTR-Module plant. Of these, about half would actually be loaded along with unfueled pebbles to reach criticality, and the remaining initial core pebbles would be used during the first part of the run-in phase to replace unfueled pebbles and to compensate for burnup of the previously loaded pebbles.

Of course, the above selections are only initial estimates to be used for scoping purposes in developing the fuel qualification and acquisition strategy. They provide a reasonable estimate for establishing program strategy and setting the near-term course for the project. Most importantly, they provide a good basis to judge the overall feasibility of a single particle UCO strategy, which can support the pebble bed HTGR concept as well as the prismatic concept. As reactor design progresses and more detailed core analysis is performed, final UCO pebble requirements will be available to support full pebble element irradiation and to begin production of initial core pebble production.

9.4 Fuel Supply and Schedule

A plan has been developed for establishing and operating coated-particle fuel manufacturing facilities in support of an initial PBR based on the HTR-Module and for a follow-on fleet of commercial HTGR modules all utilizing the PBR technology.

For the initial HTGR utilizing PBR technology, one initial core and 11 re-loads are scheduled over a 13 year period. This equates to approximately 14 MT Uranium. Commercial deployment up to 10 HTGRs is possible over the same 13 year period.

Fuel supply for the NGNP plant will be provided by B&W based on the prototype fuel fabrication facilities in Lynchburg, VA and Erwin, TN. Modifications required for the NFS facility include the addition of sintering capabilities beyond those which currently exist. While procurement of this equipment can occur concurrently with the safety and licensing reviews, the installation cannot occur until the licensing approvals are complete.

In order to meet the initial core delivery for the NGNP on schedule, a contract award of 4/1/2013 is required. Fuel delivery to the NFS site starts in the first quarter of 2016 with quarterly shipments ending in 2018 to meet the production capability of the fuel facility. An estimated 220kg U is delivered to the NFS site per quarter of fuel production It is assumed that enriched uranium for the NGNP demonstration plant is supplied from down-blended weapons material.

At the Lynchburg site, equipment procurement would occur concurrently with the safety and licensing reviews. Installation of equipment, including additional TRISO coating furnaces, would occur following licensing approval. Shipping container design and approval would need to begin in 2015, approximately four years prior to shipment of pebbles. This allows for three years of design and approval followed by one year for fabrication.

Potential technical and schedule risks associated with production of fuel for the NGNP plant include:

- Qualification of fuel to meets NGNP performance requirements – Medium Risk

- Design and licensing of pebble shipping containers – Medium Risk
- Pebble fabrication development – Medium Risk
- Development of efficient automated inspection – Medium Risk

Fuel supply for the fleet of follow-on PBRs will be based on new facilities constructed at the Lynchburg, VA site. Following contract award, key critical staff and process/facility design to support NRC licensing application occurs within three months. Building design and site preparation occur after successful NRC license application approval. The UF_6 conversion activities, along with the TRISO particle fabrication line and Pebble fabrication line are installed with pebble fabrication beginning in 2022. This equipment is sufficient until the quantity of Uranium throughput exceeds 8 MT U/year (2026 and beyond). To support throughput beyond 8 MT U/year, a second TRISO particle line capable of 4 MT U per year is added. Compacting equipment does not change with the increase in Uranium throughput as the pebble loading increases with the shift from pebbles to support initial cores to pebbles supporting re-loads. Enriched uranium for the follow-on plants is assumed to be supplied by new commercial source.

Potential technical and schedule risks associated with production of fuel for the PBR fleet include:

- Availability of enriched uranium – High Risk
- Licensing of dry UF_6 conversion facility - Medium Risk
- Coating technology upgrade to 10” coaters – Medium Risk

10.0 GRAPHITE QUALIFICATION AND ACQUISITION

10.1 Graphite Qualification

Graphite is extensively used in HTGR concepts, in particular for reactor internal components. These graphite components are relied upon to establish core geometry, serve as the moderator in support of the nuclear heat generation process, and direct the flow of helium coolant. In order to fulfill these roles, the graphite must provide neutron moderation and reflection, provide structural support to the fuel and coolant passages, remain dimensionally stable during fast neutron irradiation, and resist corrosion due to the coolant gas and impurities.

These graphite components also serve as a path for passive removal of heat in the case of certain licensing basis events, passive heat removal capability being fundamental to the HTGR safety concept.

The grades of graphite that were used for previous HTGRs are no longer available. New grades of graphite have been developed based on the strengths and weaknesses of those previous grades.

Graphite qualification programs are underway in the US and Europe (AGC and RAPHAEL) to determine irradiation behavior of graphite in support of eventual HTGR deployment. These qualification programs are focused on developing a data set of physical properties and irradiation behaviors able to support development of predictive behavioral models and associated Codes and Standards.

In order to support design and licensing needs, these programs must also verify that the behavior of the graphite grade, under operating conditions of fluence and temperature is compatible with the data and assumptions used by the designers. This compatibility is then expressed in requirements that concern only characteristics before irradiation, which are measured just after production. They must also verify fabrication processes to insure that graphite characteristics, in every point of the billet, correspond to the requirements and that the graphite material properties are reproducible during the whole production run of the components.

Current testing and data acquisition schedules are supportive of deployment of the NGNP.

10.2 Graphite Acquisition

Today, there are only a few graphite suppliers actively considered as suppliers by potential NGNP reactor vendors, including GrafTech in the US, SGL Group in Germany and France, Toyo Tanso in Japan, and Mersen (former Carbone-Lorraine) in France. These graphite vendors are experienced at producing graphite for nuclear applications. They understand and are able to meet the quality requirements for nuclear components.

Information was gathered from two of the graphite vendors (SGL and Toyo Tanso) to assess graphite availability. Based on the information collected, the graphite infrastructure is believed to be adequate to produce the needed quantity of the required grades of nuclear graphite on the planned NGNP production schedule. This availability assumes that the required quantity of graphite is ordered in a timely manner. Graphite availability for a fleet of HTGRs is expected to be acceptable, though development of additional manufacturing facilities and advanced procurement of key feedstocks may be advisable

The main issue related to graphite acquisition is that every change in raw materials (and more specifically in filler coke origin) will involve at least some requalification of a new grade. After qualification, in order to secure graphite supply, it may be useful to stock all the raw materials necessary for the manufacturing of all the graphite parts. It would be particularly necessary to consider this feedstock for pitch coke graphite, like NBG-18, because pitch coke sources are rare.

11.0 R&D NEEDS

An assessment of the design data needs for the PBR reactor type, based on the HTR-Module design was conducted. The main focus was on the needs for designing, licensing, building and operating the NGNP, however, DDNs were also identified that would support further development of this technology.

The DDN assessment was based on an analysis of the DDNs issued by the NGNP Project team for the development of the 750°C, steam cycle version of the PBR NGNP, excluding DDNs devoted to the hydrogen production process. Specific consideration of the HTR-Module design led to removal of some of the proposed DDNs, which are relevant for parts of the less mature design, but not for a design fully developed and tested. DDNs that had been developed for the AREVA prismatic steam cycle design were also considered as appropriate. Additionally, some required DDNs that could not be found in the existing lists have been added. The result of this assessment is a set of DDNs identifiable as applicable to the PBR technology based on the HTR-Module design.

Beyond the identification of the needs strictly necessary for an HTR-Module based NGNP, called Enabling DDNs, two other categories of DDNs were identified:

- Enhancing DDNs describe data that could be taken into account in NGNP design and could improve it in terms of performance, cost, or in other areas.
- Long Term DDNs describe the potential for further development of PBR technology for improving performances and cost and for enlarging its market, mostly through higher temperature applications.

This activity did not generate new DDN documents for those that were identified as new DDNs.

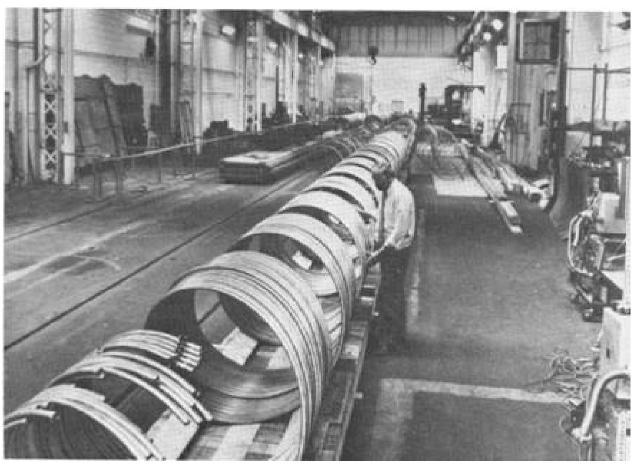
11.1 Required R&D to Support Current Concept

The main focuses of enabling R&D are the fuel and the graphite, which are clearly important R&D efforts to support the construction and operation of the NGNP. The main reasons for these needs are well known: the newly fabricated fuel has to be re-qualified; the graphite grades used for old HTGR projects, in particular for the HTR-Module, are no longer available, and therefore a new appropriate commercial grade has to be selected and qualified. There are also a few needs for developing some modeling capabilities in existing computer codes (system transient analysis, fuel and structural mechanics) and complementary data needs concerning well known metallic materials (SA-508 and Alloy 800H), as well as some limited testing needs concerning a few critical systems (fuel handling system, steam generator and RCCS).

An important issue could not be addressed properly in this study, though it might have a significant influence on the volume of R&D required is the impact on design data needs of the necessary Americanization of the design. This concerns most particularly the materials. Large databases have been developed in Germany under KTA standards on the materials used in the HTR-Module. Will these databases be acceptable for use on the NGNP, will it be necessary to reconstruct them fully or simply to check that a few new data obtained following US requirements are consistent with the German data and possibly to complement them for some missing or doubtful data? Following the answer to this question the magnitude of the R&D program on materials will be quite different. A general answer cannot be given to this question. It will have to be addressed on a case by case basis, examining the quality assurance information available on each set of data, and involving the ASME in the discussions.

A similar question exists for the qualification of the critical components: if they are built strictly following the design defined for them in the HTR-Module, will it be necessary to re-qualify them, in spite of the fact that they have been fully qualified in Germany, admittedly not following US standards. This concerns more particularly, but not exclusively, the steam generator. Its design was justified not only by calculation, but many tests were performed concerning heat transfer, flow distribution, bundle vibratory behavior, and fabricability including integral tests at full scale (but with a reduced number of tubes) at full temperature, pressure, and chemical conditions (helium with controlled impurities) in the 10 MW KVK loop. Duplicating these tests would cost hundreds of millions of dollars, including the construction of a large loop similar to KVK. What is recommended here is to rely upon the existing tests, which are fully documented, except for fabricability tests and tube bundle inspection tests. Even if fabricability was proven in the 1980s with a German manufacturer, a manufacturer selected for NGNP will have to be qualified with its own methods, which will likely require a significant number of tests, as a helical bundle steam generator of 200 MW, which is moreover a nuclear component, will certainly not correspond to daily industrial practice. On the other hand, as the steam generator tubes will have to be inspected with present inspection methods, and not with methods existing in Germany in the 1980s, these methods will have to be adapted to the particularities of the HTR-Module design, taking into consideration other experiences gained on inspection of helical tube bundles, and qualified.

Figure 11-1: Helical Heat Exchanger Tube Test in KVK Loop



11.2 R&D for Future Advanced Concepts

Enhancing DDNs are not numerous, which shows that the HTR-Module technology is relatively mature and requires only few developments to be optimized. The main developments considered here are relative to the fuel and to the circulator. There are also significant needs concerning the integration of the progress made in the last few decades on instrumentation and the development of modern radio-contaminant transport models.

For the fuel, the use of UCO instead of UO_2 will generate new data needs, including the need to qualify reactor physics codes for higher burn-up, in order to be able to use the burn-up margins obtained with UCO. On the other hand AREVA considers that there is still a significant effort for understanding and mastering the methods for fabrication of HTGR fuel before being able to perform large scale stable industrial fabrication of this fuel in good quality and economic conditions. Moreover the development of modern non-destructive quality control methods and their integration into the fabrication process will support significant decrease in the cost of the fuel, while maintaining its high quality level, or even improving it if needed.

The change of oil bearings for magnetic bearings will significantly affect the design of the whole circulator, requiring a significant program of tests, including integral tests of the whole component.

As could be expected, there are many more R&D needs for long term development towards higher temperature and direct heat supply than for 750°C steam supply. They are focused on materials development and on the IHX development. AREVA agrees with the current assessment on the effort to be made on the plate IHX, which appears to be a more economic solution than tube IHX. The main challenge with a plate IHX is to obtain an acceptable lifetime. Therefore the tube IHX solution should not be neglected as a back-up. However, this type of design has been the object of extensive developments in Germany and thus no significant R&D needs have been identified.

On the other hand, AREVA is rather skeptical of the possibility of developing a large IHX in ceramics within a reasonable period of time. Moreover AREVA considers that current estimates underestimate the length and the complexity of the experimental program necessary for supporting the development of a plate IHX, which could likely not be integrated in the schedule of NGNP for a starting of the reactor in the early 2020s. AREVA proposes a step-by-step approach, which would have the merit of allowing screening different plate IHX concepts for selecting the most appropriate and giving a preliminary answer on its feasibility for VHTR operating conditions before launching a very costly qualification program.

Another difference is relative to the range of applicability of Alloy 800H and of Nickel base alloys. AREVA considers that the range for applicability of Alloy 800H cannot go very far beyond the range of operational conditions considered for the steam cycle NGNP and that Nickel base alloys will find their limits, due to the degradation of their mechanical properties as well as to the enhancement of corrosion, between 850 and 900°C.

A question that remains open is relative to the need to develop an advanced fuel for higher operating temperatures. Will UCO be sufficient or will additional developments be necessary, in particular on advanced coatings if the behavior in accident conditions is to be enhanced? A DDN has been identified to answer these questions and give the orientations to a possible R&D program. Presently it is too early even to outline the R&D needs for such a possible program.

Finally it should be noted that apart from hydrogen production, there is a large area of development that is not addressed here: the area of process heat applications, because the processes will have to be adapted to the heat supply from a nuclear reactor. Contrary to the case of steam supply, which can be imagined as a plug-in substitution of a nuclear boiler to conventional boilers into an existing steam network, convective heat supply will replace radiative heat transfer from combustion of fossil fuel directly around the process chamber or even internal combustion inside the process chamber. The conditions of the processes will drastically change, and therefore the processes will have to be re-optimized or even fully modified. New components for heat transfer and for process will have to be developed as well as technologies for heat transport, which is not a common industrial practice at temperatures above 550°C. But this is an area where end-users of process heat have to be involved and development needs have to be identified with them on a case-by-case basis.

12.0 CONCLUSIONS

The HTR-Module design provides a solid basis for evaluation of the current status of PBR technology and a solid foundation for future development of a PBR design for US deployment. The HTR-Module had progressed to the early Final Design stage in Germany and review by the regulatory authorities was essentially complete.

Reconciliation of the HTR-Module design with American regulatory requirements, construction and design standards, and market requirements will be needed. But the basic elements of the concept would not be affected, and a clear path is available to complete the final design without excess design development or iteration.

The existing HTR-Module design does not fully comply with all current NGNP Project requirements, but the intent of the NGNP requirements is satisfied. Reconciliation with the NGNP requirements can be easily addressed as part of the overall Americanization process.

12.1 Conclusions on PBR Technology Issues

The PBR technology assessment examined several issues that are specific concerns for PBR concepts. These are hypothetical "showstoppers" that are sometimes raised as objections to pursuing PBR technology. While some issues were complex and others were relatively simple, it was determined that none are fundamental roadblocks to further development of PBR technology.

That is not to say that none of the issues pose challenges. Some of the issues involve new paradigms. Many of the issues will require substantial dialogue with regulators and other stakeholders before they are resolved. But the conclusion of the PBR technology assessment is that no issues were identified which cannot be resolved successfully based on the assessment's consideration of the fundamental technical considerations for each issue.

12.2 Overall Conclusions

The overall conclusions of the PBR technology assessment are:

- Modular PBR well-aligned with NGNP Project objectives
- PBR technology is well established based on demonstration projects in Germany
- Infrastructure can support PBR demonstration and deployment
- PBR safety characteristics are well understood
- Substantial investment required to support FOAK Project
- Mature concept provides a long-term alternative to fossil fuels with rising gas prices and CO_2 penalties

Modular PBR technology provides a viable option to replace fossil fuels in meeting long-term industrial process heat needs.