AREVA NP Inc.

Technical Data Record

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Pebble Bed Reactor Technology Readiness Study

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Pebble Bed Reactor Technology Readiness Study

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TABLE OF ACRONYMS

AGR	Advanced Gas Reactor
AVR	Arbeitsgemeinschaft Versuchs-Reaktor
B&W	Babcock and Wilcox
BISO	Bi-Layer Coating for Fuel Particles used in HTGRs
BUMS	Burnup Measurement System
CANDU	Canadian Deuterium/Uranium Reactor
C/S	Containment/Surveillance
DBA	Design Basis Accident
DDN	Design Data Need
DLOFC	Depressurized Loss of Forced Cooling
ECP	Energy Conversion Plant
EIS	Environmental Impact Statement
EP	Environmental Protection
FHS	Fuel Handling System
FOAK	First-of-a-Kind
HTGR	High Temperature Gas-Cooled Reactor
HTR	High Temperature Reactor
HTR-10	High Temperature Reactor, 10 MW (China)
HTR-MODUL	High Temperature Modular Reactor (Germany)
HTTF	High Temperature Test Facility
IAEA	International Atomic Energy Agency
ID	Identification
IH&S	Industrial Health and Safety
IC	Initial Core to reactor
IROFS	Items Relied on For Safety
KMP	Nuclear Material Key Measurement Point
LEU	Low Enriched Uranium (U% < 20% U-235)
LWR	Light Water Reactor
MAA	Material Access Area
MBA	Mass Balance Area
MPBR	Modular Pebble Bed Reactor

MUF	Material Unaccounted For
NCS	Nuclear Criticality Safety
NDA	Non Destructive Assay
NFS	Nuclear Fuel Services
NGNP	Next Generation Nuclear Plant
NOAK	N^{th}-of-a-Kind
NOG-L	Nuclear Operations Group – Lynchburg
PBMR	Pebble Bed Modular Reactor (South Africa)
PBR	Pebble Bed Reactor
PFD	Process Flow Diagram
PIE	Post-Irradiation Examination
PWR	Pressurized Water Reactor
RBMK	On-Line Refueled Reactor (Graphite Moderated) (Former USSR)
RCCS	Reactor Cavity Cooling System
RP	Radiation Protection
RPV	Reactor Pressure Vessel
SAR	Safety Analysis Report
SER	Safety Evaluation Request
SNM	Special Nuclear Material
SQ	Significant Quantity (of Fissile Material for Making a Nuclear Explosive)
THTR	Thorium High Temperature Reactor
TRISO	Tri-Layer Coating for Fuel Particles used in HTGRs
U	Uranium
U-235	Uranium 235 Isotope
UPRR	Uranium Processing and Research Reactors

1.0 SUMMARY

1.1 Purpose

This report is an evaluation the overall readiness of the PBR technology for deployment based on the design of the German HTR-Module. Emphasis is placed on assessing the critical issues or open questions that might influence deployment of the PBR concept. This report investigates the following major categories:

- Design Status
- Status of Key PBR Issues
- PBR Technology Database
- PBR Fuel Supply
- PBR Graphite Supply
- Constructability and Transportability

A summary of these major categories is provided below.

1.2 PBR Design Status

For each major plant system, an assessment of the readiness of the system for deployment in the NGNP was made. In that assessment, aspects of the system that are appropriate for use effectively as-is, as well as aspects that will require update were identified, including description of the nature and extent of the required update. In addition, the overall status of the HTR-Module design was assessed considering its overall level of completeness, and the effort required to implement that design in the United States, while meeting all key NGNP requirements.

The HTR-Module design is a German design that uses largely proven technology. This design met all necessary requirements of German nuclear regulatory authorities. Review of pertinent German design documents indicate that the HTR-Module was well into the final design stage. Foremost among the challenges of deploying this design for the NGNP is the need for the design to accommodate U.S. regulatory requirements, codes and standards (sometimes called Americanization of the design). It is clear that the NGNP PBR design must necessarily undergo some degree of regression from the near final design stage of the German HTR-Module on which it is based. It has been estimated that an NGNP based on the HTR-Module should be considered to be in the late conceptual design stage. In order to progress to the point of early preliminary design, a reconciliation of the design to NGNP requirements and an initial round of “Americanization” would be necessary. Based on this review, the PBR concept is at a design point that is consistent with the needs of the NGNP program.

1.3 Status of Key PBR Issues

This report describes a series of issues that have been identified by various stakeholders as potentially problematic for deployment of the PBR technology. For these issues, an assessment has been conducted, beginning with a description of exactly what each issue is, and sometimes more importantly what it is not. After the description, an assessment is made as to the potential impacts of the issue on implementation of the PBR technology. As appropriate, design updates recommended or required to alleviate identified concerns are described. Supporting data reviews and/or scoping calculations are provided as necessary to fully explore each issue. The issues assessed were:

- The stochastic nature of the PBR core

- Core compaction during a seismic event
- Graphite dust
- Impact of broken and lost pebbles
- Proliferation
- Shutdown margin issues
- Availability impacts of online refueling
- Tritium

Results of these assessments indicated several areas where the design of the base HTR-Module can evolve during preliminary and final design activities to more fully address certain aspects of the issues. Overall, however, no issues were identified that would preclude deployment of the PBR technology within the framework of the NGNP project.

1.4 PBR Technology Database

The technical database assessment provides a detailed review of existing PBR Design Data Needs identified by the NGNP project for the PBMR technology. The current status of the technology and completion of the actions described within each DDN is assessed. In addition, gaps in the current R&D programs are identified to facilitate generation of any required new NGNP PBR DDNs. New DDNs were identified but not detailed as part of this task.

An assessment of the design data needs for the PBR reactor type, based on the HTR-Module design, was conducted. It was based on an analysis of the DDNs issued by the NGNP project for the development of the NGNP 750°C, steam cycle version of the PBR, excluding DDNs devoted to the hydrogen production process. Specific consideration of the HTR-Module design led to removal of some of the DDNs, which are relevant for parts of the less mature design, but not for a design fully developed and tested. Additionally, some DDNs that could not be found in the list have been added; these correspond to the views of AREVA experts. The result of this assessment is a set of DDNs identifiable as applicable to the PBR technology based on the HTR-Module design. This activity did not generate new detailed DDN documents for the newly identified DDNs.

1.5 PBR Fuel Supply

For the fuel described in the PBR Design Description report, the required fuel qualification programs and fuel acquisition strategies necessary to support NGNP operation consistent with the anticipated schedule are described. An examination of how the current Advanced Gas Reactor (AGR) fuel qualification program, being conducted by INL, could be used as the basis for qualification of the PBR fuel is provided, including identification of any needed qualification activities outside of the AGR program. A description of several possible paths forward to supply fuel for the postulated plant startup in 2021 is also provided. Several scenarios of fabrication equipment usage/upgrades and early commitments to fuel fabrication are included. A key part of this discussion is the identification of needed actions to assure ability to supply fuel for the PBR startup on schedule.

A fuel design and associated qualification strategy was developed based on the current AGR program being conducted by the INL. 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.

Babcock and Wilcox is positioned to support the NGNP program and produce the selected fuel design. With modest capital investment, the capabilities to supply fuel for the HTGR Initial Core can be secured in an approximately 5 year time frame. During this time frame, development efforts to optimize the fabrication process would occur. These efforts could then be channeled into the design and construction of a commercial fuel fabrication facility. The design of the facility would be modular. Additional modules can be added on an as need basis. This modular design allows for efficient scale up of commercial fuel fabrication beyond what is identified within.

As with any project involving the processing of Uranium above 5% ^{235}U, there are risks in securing a suitable Uranium supplier. Beyond that, the risks identified are all manageable. None of the risks identified are believed to be insurmountable.

1.6 PBR Graphite Supply

For the graphite structures described in the PBR Design Description report, the required qualification programs and acquisition strategies necessary to support NGNP operation consistent with the anticipated schedule are described.

The graphite infrastructure is believed to be adequate to produce the quantity of the selected grade of nuclear graphite on the planned NGNP production schedule. This assumes that the required quantity of graphite is ordered in a timely manner. A key issue for PBR graphite qualification is that long life/high fluence is required of reflector graphite. This means the graphite performance may be less certain, and in particular, it means that longer test periods are required to get relevant data. The main issue on graphite acquisition is that every change in raw materials (and more specifically in filler coke origin) will involve the qualification 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 stock for pitch coke graphite, like NBG-18, because pitch coke sources are rare.

1.7 Constructability and Component Transportability

This section provides an assessment of the constructability of the PBR based on the reactor module description provided in the PBR Design Description report. This assessment draws upon previous applicable assessments conducted for the NGNP project as well as new input based on design specific considerations and current NGNP deployment strategies. Part of this assessment considered the transportability of major components to the NGNP site and described potential strategies to mitigate transport concerns.

As with any major construction project, there are risks and opportunities. The key is to identify both the risks and the opportunities early and plan their fate during the design phase of the project. Although there are challenges with the construction of the high temperature gas-cooled reactor (HTGR), we do not believe there are issues that extend beyond those experienced and resolved on similar projects. The current challenges with new-build nuclear projects have and will continue to prepare the industry for the deployment of the NGNP.

2.0 INTRODUCTION

2.1 NGNP Project

HTGRs can provide an important addition to the US and the world's energy supply portfolio. Enabling commercial deployment of the 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 EPAct 2005 provides for a collaborative effort between government and industry to enable the commercialization of the HTGR technology.

To achieve this goal, the NGNP Project must develop and demonstrate the design, licensing, performance, operational capabilities, and economic viability of HTGR and associated process heat technologies. The Project must further enable development of the commercial vendor/owner/user infrastructure, and support the timely Design Certification of the commercial designs by the NRC to help assure subsequent deployment in the commercial market place.

The Department of Energy has been directed by Congress to establish a project to focus on the development, early design and licensing of an advanced HTGR and the associated advanced technologies to transport the high temperature process heat. The basis for the HTGR technology embodied in the NGNP was first developed over 40 years ago in the UK, the US and Germany. Most of the previous work has focused on the generation of electricity. Seven experimental and demonstration reactors have been built world-wide, including a US commercial scale demonstration of a specific HTGR concept for electric power generation at the Fort St. Vrain plant that operated from 1976 through 1989. Other HTGR system-related development efforts exist in South Africa, France, Japan, Russia and China at the design stage or engineering pilot scale. Additionally, a commercial scale demonstration plant utilizing the pebble technology is currently under construction in China.

As currently envisioned, the NGNP Project will result in full scale First-of–a-Kind (FOAK) facilities that demonstrate the commercial potential of the HTGR and associated technologies. Definition of the specific NGNP facilities to be built as part of the Project will be established over the next several years. Two HTGR technologies are being considered as part of the initial phase of the NGNP project. The prismatic design concept is being developed under a DOE FOA funding by the General Atomics design team and the pebble bed HTGR reactor technology concept is being evaluated by the AREVA design team. As the conceptual design and technology assessment work progresses, the facility design is better defined and the costs and the economics of the project are defined with more certainty.

2.1.1 NGNP Project Objectives

The primary goal of the NGNP Project is enabling the commercialization of the HTGR technology across new industrial and commercial markets previously not accessible to nuclear technology. The NGNP Project will create the option for deployment of the HTGR technologies for a range of applications and sites not traditionally served by nuclear energy.

Key objectives for achieving this goal include [1]:

- Fully characterizing the potential market through end-user collaborations and application studies in order to identify a wide range of viable candidate sites, applications and projects

- Providing guidance to design teams regarding the range of site and application requirements that could impact NGNP design and licensing
- Preparing, submitting, and acquiring one [2] or multiple Early Site Permits (ESPs) that envelop the range of potential sites and applications for deployment of HTGRs
- Performing the design activities necessary to prepare, submit, and eventually obtain a Combined License (COL) for one or both HTGR technologies
- Developing the regulatory framework for the licensing of the HTGR technologies
- Enabling the long-lead developmental activities for fuel, high-temperature materials, and methods that support licensing and subsequent construction of the FOAK facilities
- Securing the fuel fabrication capacity needed to support HTGR projects
- Completing the final design activities to allow construction, start-up, confirmatory testing, and operation of the FOAK facilities
- Acquiring the necessary government incentives to make the FOAK facilities economically viable investments for the private sector
- Construction, start-up, confirmatory testing, and completing a commercial operations run for the FOAK facilities
- Enabling the establishment of the supply chain infrastructure necessary for commercial build-out of the HTGR technologies
- Obtaining design certifications from the NRC to support the deployment of the initial fleet of commercial plants
- Capturing the lessons learned from FOAK construction and operations, and validating the assumptions for future plant construction costs and schedule

By meeting the objectives above, it is expected that the NGNP Project will establish an acceptable basis for commercial deployment of the HTGR technology in the broader energy sector. Completing the design, licensing, construction and initial operations of a FOAK plant provides a solid foundation for commercialization and commitment to the extensive deployment anticipated for the HTGR technology, end-user site requirements and hazards, and nuclear-industrial collocation conditions

2.2 PBR Background

In the late 1980s a modular pebble bed reactor concept, the HTR-Module 200, was proposed in Germany. The HTR-Module is a pebble bed modular gas-cooled reactor with a cylindrical core and passive decay heat removal features. The HTR-Module design is considered ready for final design activities, and was reviewed by the German regulatory agency and approved by the German reactor safety commission. This design formed the basis for the subsequent PBR modular reactor designs and therefore has been selected as the reference design for this scoping safety assessment.

The HTR-Module reactor cylindrical core enclosure is constructed from graphite blocks and contains approximately 360,000 fuel spheres (i.e., pebbles). Each fuel sphere contains approximately 11,600 coated fuel particles for a total heavy metal loading of seven grams of low enriched uranium (LEU) oxide. The pebbles are randomly packed in the vessel. The fuel spheres (pebbles) in the pebble bed core are continuously on the move. The direction of the pebble movement is from top to bottom. As the pebbles are removed from the bottom they are examined for their structural integrity and burnup level. Once it's determined that the pebble is structurally sound and not yet reached its burnup limit it is returned to the top of the pebble bed core. A total of 15 cycles are expected for the HTR-Module fuel sphere before it is reached its 80,000 MWd/MT burnup target and it is discarded into the used fuel storage/transport facility and a fresh fuel sphere is then introduced to the top of the pebble bed core.

The reactor uses helium gas as the heat transport media. The cold gas is blown in from the top of the core and forced through the packed bed of fueled spheres to carry off heat generated by the nuclear fission. The heat generated in the reactor core is carried by the gas to the steam generator where it transfers its heat to the water in the steam generator to produce super-heated steam. The primary circuit is then completed as the cooled gas is forced back into the core by the primary gas circulator.

The reactor control is achieved with control rods inserted into the side reflectors. The core diameter is selected such that the geometry provides for sufficient negative reactivity worth in the radial absorbers rods so that in-core reactivity control is not necessary.

A secondary reactor shutdown system is also provided. This system consists of neutron absorber spherical elements that are dropped into the dedicated channels in the graphite reflector to shutdown the reactor. This system is available as a backup/secondary system to the control rods, but it is not used for power shaping or power maneuvering.

The reactor is designed to operate as "base-load" or "load-following" modes. Load following mode of operation is achieved by varying the speed of the helium circulator thus controlling the primary coolant flow.

The plant is configured with 2 x 200 MWt reactor modules each generating super-heated steam for independent turbine-generator sets for electricity production and reboilers to provide high temperature steam for industrial application steam heating.

2.3 PBR Technology Status Assessment

The U.S. Department of Energy (DOE) has selected Idaho National Laboratory (INL) as the lead national laboratory for nuclear energy research. Per the terms of the EPAct, Title VI, Subtitle C, Section 662, INL, under the direction of DOE, will lead the development of the NGNP by integrating, conducting, and coordinating all necessary research and development activities, and by organizing all project participants, including industry. INL will also be responsible for conducting site and project related procurements, and coordinating project efforts within the industrial and international communities.

As required by the EPAct, the Nuclear Energy Advisory Committee (NEAC) will conduct a "first project phase review," when the first phase of NGNP is nearly complete. The first phase of NGNP includes the research and development, technology, licensing, and conceptual design information derived from all Phase 1 activities. Two main technology options are under consideration for the NGNP: the prismatic block core modular HTGR, and the pebble bed reactor (PBR) modular HTGR. The evaluation of these two reactor concepts will form an important part of the Phase 1 review. Conceptual design information for the prismatic reactor concept is being developed under a separate work scope. The purpose of this work is to develop key information to support the review of the PBR technology option.

This effort will provide a limited assessment of the PBR concept that includes the basic design information and various assessments of the design concept needed to evaluate the maturity of the PBR design concept and its technical readiness to advance to the next level. This work did not intend to produce a conceptual design of the NGNP reactor with the PBR technology.

The bases for the PBR technology readiness status assessment is the AREVA HTR-Module design developed in Germany in the late 1980s plus enhancements that support current requirements, safety, and licensing. Adjustments to the referenced plant design would be considered based on HTGR design experience since the HTR-Module was not originally 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.

An evaluation of the readiness of this design is made using trade studies and expert engineering judgments. The results of these assessments are documented in four deliverables:

1) The Plant Design Description report – PDD describes the reference PBR design that is based on the HTR-Module and identifies potential design enhancements. The PDD identifies key system requirements, describes the overall PBR plant and provides a description of each critical structure, system, and component (SSC). Engineering analyses and trade studies, such as a point design and steady-state plant analyses, shall be performed to adapt the previous designs to the NGNP requirements.

2) The PBR Technology Readiness Assessment report (this report) – The technology readiness assessment comments on the readiness status of various technologies necessary to build the NGNP with PBR technology. An existing set of design data needs (DDN) will also be reviewed and potential changes or modifications will be recommended. A study evaluating the overall PBR technology readiness for deployment was performed. This study performed the following: a) examined key PBR technology issues, b) identified technology needs by evaluating the existing design data needs (DDNs) for the PBR design and gaps in the identified needs, c) discussed fuel and graphite qualification and acquisition, and d) discussed the constructability and component transportability of the PBR design concept.

3) The PBR Scoping Safety Study report – In the safety study report the PBR safety case is presented and discussed, the original German HTR-Module accident analysis results are provided and discussion of key technical issues relevant to PBR safety case is presented. The scoping safety study is based on existing analyses; new analyses are not within the scope of this work. This work included review of prior HTR-Module safety analyses. The review included identification and assessment of the PBR plant safety issues and discussion/assessment of the expected outcomes for each major accident sequence. Considerations specific to the PBR technology, such as graphite dust and the requirement for a stochastic approach to the core design and analysis, are reviewed and discussed. The safety study also includes an evaluation and discussion of expected dose at the site boundary (about 400m) for accidents with dose releases using accepted U.S.A. dose calculation methodology and with the original accident source terms.

4) The Cost and Schedule report – This report provides an updated cost and schedule for the PBR FOAK and the NOAK plants. Cost and schedule estimates for deployment of the PBR are developed for the FOAK and NOAK plants. The cost estimate is based on historical information from previous PBR evaluations and similar components as appropriate with scaling, and adjusted as necessary to match the current PBR design concept. The cost estimate addresses a single plant for the FOAK plant and a multiple plant installation for the NOAK. The

plan includes an overall project schedule covering detailed design, fabrication, and construction of the demonstration PBR plant.

2.4 Approach to Readiness Assessment

This report evaluates the overall technology readiness for deployment of the PBR concept through examination of several key topics, including:

Status of the HTR-Module Design - For each major plant system, an assessment of the readiness of the system for deployment in the NGNP was made. In that assessment, aspects of the system that are appropriate for use effectively as-is, as well as aspects that will require update were identified, including description of the nature and extent of the required update. In addition, the overall status of the HTR-Module design was assessed considering its overall level of completeness, and the effort required to implement that design in the United States, while meeting all key NGNP requirements.

Readiness of Supporting Technology Database - This section provides a detailed review of existing PBR Design Data Needs identified by the NGNP project for the PBMR technology. The current status of the technology and completion of the actions described within each DDN is assessed. In addition, gaps in the current R&D programs are identified to facilitate generation of any required new NGNP PBR DDNs. These new DDNs were not generated as part of this task.

PBR Fuel Supply Readiness - For the fuel system described in the PBR Design Description report, the required fuel qualification programs and fuel acquisition strategies necessary to support NGNP operation consistent with the anticipated schedule are described. An examination of how the current AGR fuel qualification program, being conducted by INL, could be used as the basis for qualification of the PBR fuel is provided, including identification of any needed qualification activities outside of the AGR program. A description of several possible paths forward to supply fuel for the postulated plant startup in 2021 is also provided. Several scenarios of fabrication equipment usage/upgrades and early commitments to fuel fabrication are included. A key part of this discussion is the identification of needed actions to assure ability to supply fuel for the PBR startup on schedule.

PBR Graphite Supply Readiness - For the graphite structures described in the PBR Design Description report, the required qualification programs and acquisition strategies necessary to support NGNP operation consistent with the anticipated schedule are described.

PBR Constructability and Transportability Assessment - This section provides an assessment of the constructability of the PBR based on the reactor module description provided in the PBR Design Description report. This assessment draws upon previous applicable assessments conducted for the NGNP project as well as new input based on design specific considerations and current NGNP deployment strategies. Part of this assessment considers the transportability of major components to the NGNP site and describes potential strategies to mitigate transport concerns.

2.5 Key PBR Technology Issues

In addition to the readiness assessments described above, Section 5.0 of this report describes issues that have been identified by various stakeholders as potentially problematic for deployment of the PBR technology. Each section begins with a description of exactly what each issue is, and sometimes more importantly what it is not. After the description, an assessment is made as to the potential impacts of the issue on implementation of the PBR technology. As appropriate, design updates recommended or required to alleviate identified concerns should be described. Supporting data reviews and/or scoping calculations are provided as necessary to fully explore each issue. The issues explored in this section are:

Stochastic Core – In the PBR, the fuel position is known statistically, not deterministically. As such, there may be local hot spots due to packing of fresh fuel pebbles. In addition, it is difficult to predict precise pebble flow, power, and temperature distribution. The experiences at the AVR reactor with anomalous melt wire experiment results are examined for applicability to the stochastic core issue.

Core Compaction - Compaction of the pebble bed during seismic events can introduce a reactivity increase in the core. Impacts of this increase are assessed.

Dust - Graphite dusts are generated in the reactor core and fuel handling system. Fission product radionuclides are adsorbed onto this graphite dust and transported during an accident. In addition, dust in the reactor cavity might interfere with reactor cavity heat transfer and could be combustible.

Lost/Broken Pebbles - Pieces of broken pebbles may get stuck in coolant holes in the bottom of the core, which may lead to long term exposure to high temperature and high burnup. They may also impact helium and pebble flow distributions.

Shutdown Margin - During startup and equilibrium core conditions, alternate means may be needed to assure that sufficient positive reactivity is available for plant operations (starting up, power ascension and at power maneuverability) and sufficient negative reactivity (shutdown margin) is available to stop the nuclear reaction (hot shutdown and cold shutdown).

Online Refueling Fuel Handling System Availability - The PBR online refueling scenario may have significant impacts on overall plant availability, both positive and negative. These impacts need to be assessed to provide realistic comparison to other refueling schemes.

Proliferation – The inability to discretely monitor individual Special Nuclear Material (SNM) items may increase proliferation concerns due to increased monitoring complexity. In addition, though spent pebbles are "bad" for a nuclear weapon, one-pass pebbles are "good" based on plutonium isotopic content.

Tritium - Tritium is considered an important source term because it can permeate into graphite and through metals. Tritium permeation through heat exchanger tube wall may causes problems with steam production cycle and/or downstream process heat applications.

2.6 Document Structure

This document is organized as follows:

Section 1 provides a summary of the PBR Technology Readiness evaluations, assessments and conclusions. This section also summarizes the methodology used to develop these conclusions.

Section 2 provides an overview of the NGNP project and an introduction to the role the AREVA PBR technology readiness status assessment task and the Technical Readiness study.

Section 3 provides a short description of the HTR-Module primary and support systems to promote context and understanding of the technology readiness assessments. More detail PBR system description is provided in the Plant Design Description document, which is a separate deliverable of the technology assessment document.

Section 4 provides an assessment of the state of the HTR-Module design considering primarily the level of compliance of the design to NGNP requirements and an equivalent design phase (pre-conceptual, conceptual, preliminary, final) taking into account required design modernization and Americanization needs.

Section 5 provides detailed assessments of several items that have been determined to be key technical issues for the PBR technology.

Section 6 provides an assessment of technology development activities needed to implement the PBR technology.

Section 7 provides an assessment of current PBR fuel qualification plans and describes a fuel acquisition strategy to supply fuel in accordance with the NGNP deployment schedule.

Section 8 provides an assessment of current nuclear graphite qualification activities and describes a graphite acquisition strategy to supply fuel materials and graphite components in accordance with the NGNP deployment schedule.

Section 9 provides an assessment of constructability of plant buildings and systems and transportability of heavy components.

Section 10 provides a summary of the conclusions of the technology readiness study.

Section 11 lists the references used to support information in this report.

3.0 PEBBLE BED REACTOR DESCRIPTION

In the late-1980s the modular pebble bed reactor concept, the HTR-Module 200, was proposed in Germany. The HTR-Module 200 is a 200MWt pebble bed modular gas-cooled reactor with a cylindrical core and passive decay heat removal features. This design formed the bases for the subsequent PBR modular reactor designs and is the reference design for this assessment [3].

The reactor cylindrical core enclosure is constructed from graphite blocks and contains approximately 360,000 fuel spheres (i.e., pebbles). Each fuel pebble contains approximately 11,600 coated fuel particles for a total of seven to nine grams of low enriched uranium (LEU). The pebbles are randomly packed in the vessel with a packing density of 0.61.

The reactor uses helium gas as the heat transport media. The cold gas is blown in from the top of the core and forced through the packed bed of fueled pebbles to carry off heat generated by the nuclear fission. The heat generated in the reactor core is carried by the gas to the steam generator where it transfers its heat to the water in the steam generator to produce steam. The primary circuit is then completed as the cold gas is blown back into the core.

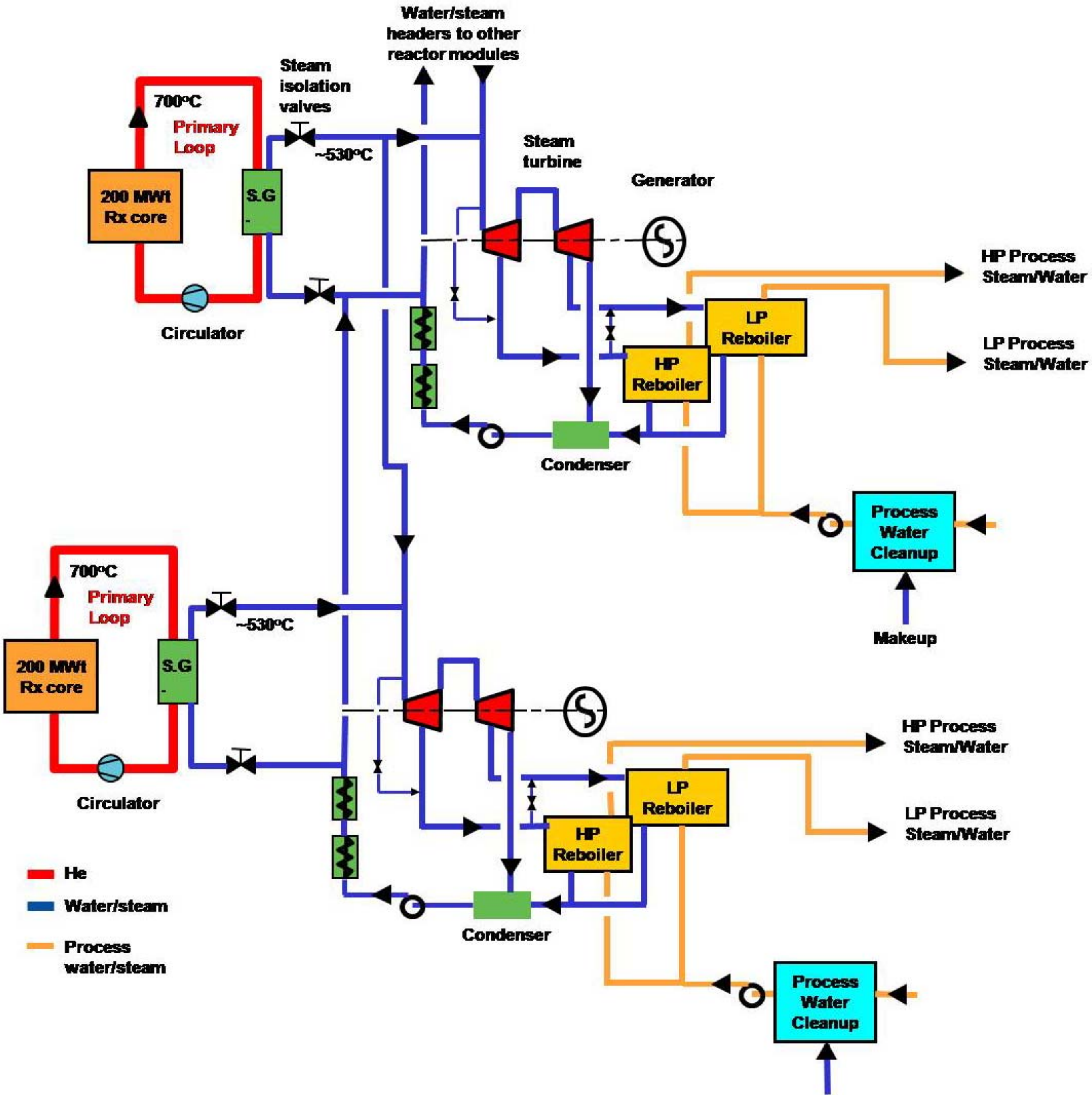
The primary reactor control is achieved with control rods inserted into the side reflector. The core diameter is selected such that the geometry provides for sufficient negative reactivity worth in the radial absorbers rods so that in-core reactivity control is not necessary.

A secondary reactor shutdown system is also available. This system consists of neutron absorber spherical elements that are dropped into the dedicated channels in the graphite reflector to shutdown the reactor. This system is available as a backup/secondary system to the control rods, but it is not used for power shaping or power maneuvering.

The reactor is designed to operate in “base-load” or in “load-following” modes. Load following mode of operation is achieved by varying the speed of the helium circulator thus controlling the primary coolant flow.

A representative plant schematic can be seen in Figure 3-1.

Figure 3-1: HTR-Module 200 Plant System Schematic Representation



Each plant power block consists of two reactor units. Each reactor unit is comprised of one high-temperature pebble bed core, one steam generator and one primary gas circulator. The primary helium transfers the reactor fission heat to the steam generator coils through a concentric gas duct pressure vessel where the reactor inlet and outlet flow in opposite directions separated by an insulated circular duct.

The reactor major operating characteristics and parameters are presented in Table 3-1 and Table 3-2. Values presented in this table are representative of one reactor unit. The basic commercial plant power block, however, is based on a dual unit configuration that combines two reactor units into a single reactor building.

Table 3-1: Main Plant Characteristics

Description	Baseline Value
Nuclear Island	
Reactor	High Temperature Gas-cooled Graphite Moderated Reactor
Construction	Partially underground
Reactor Building	Vented/Filtered
Primary Coolant	Helium
Reactor Core	
Fuel Form	Fuel Spheres (Pebbles)
Configuration	Cylindrical Core
Moderator	Graphite
Reflector	Graphite Blocks
Fuel	
Fuel Design	TRISO Coated Particles in Spherical Fuel Elements
Fuel Kernel	UO_2 (UCO for the Fuel Acquisition Section)
Coating	TRISO
Fuel Enrichment	8 ± 0.5% (14% for the Fuel Acquisition Section)
Basic Fuel Element	A mixture of coated particle fuel in graphite matrix shaped into a sphere with an out layer of fuel free zone
System Configuration	Conventional Rankine Steam Cycle w/Helium to Water/Steam Generation
Primary Loop Configuration	
No. of Loops	One
No. of Cross Ducts	One

Description	Baseline Value
No. of Steam Generators	One
No. of Circulators	One
Heat Transport System	
Primary Fluid	Helium
Steam Generator	
Steam Generator Type	Once-through helical coil
Heat Transfer Medium	Helium to Water/Steam
Refueling System	
Refueling Cycle	Online/Pebble Recirculation

Table 3-2: Plant Physical Parameters – Main Components

Description	Baseline Value
Reactor Parameters	
Core Diameter (m)	3.0
Core Height (m)	9.4
Mean Power Density (W/cm^3)	3.0
Number of Pebbles	~360,000
Number of Control Rods	6
Number of Absorber Ball Channels	18
Average Number of Passes per Pebble	15
Number of Particles per Pebble	around 11,600
Heavy Metal Loading (g/pebble)	~7
Discharge Burnup (MWd/kgHM)	80
Fuel Residence Time (days)	~1000
Reactor Pressure Vessel	
RPV Height (m)	25
RPV Outside Diameter (m)	6.8
RPV Inside Diameter (m)	5.9

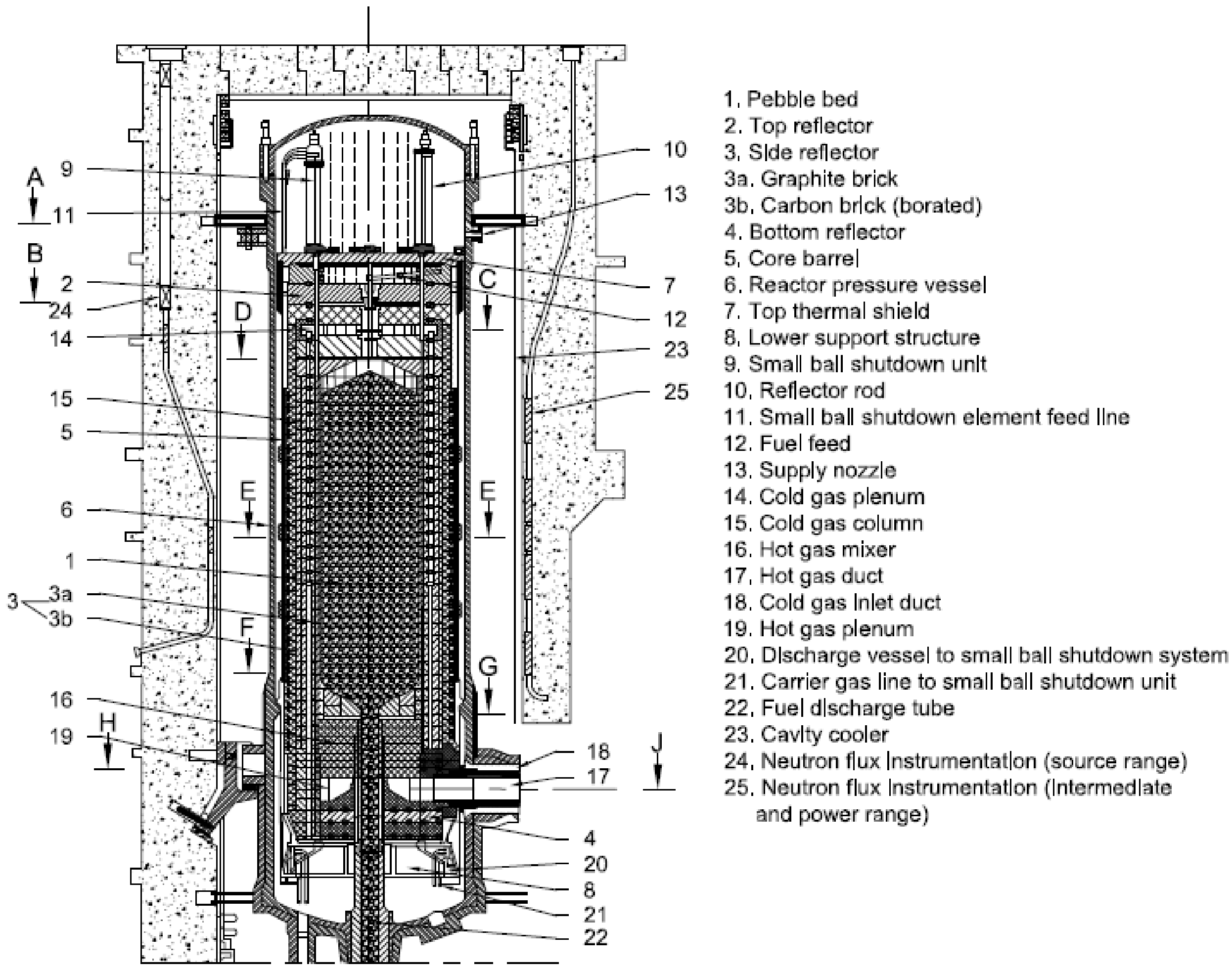
Description	Baseline Value
Circulator	
Circulator (delta P - static head) (bar)	1.5
Steam Generator Pressure Vessel	
SG Pressure Vessel Height (m)	22
SG Inside Diameter (m)	
Upper	3.6
Lower	3.2
SG Tube Outside Diameter (mm)	23
Duct Pressure Vessel	
Duct Pressure Vessel Length (m)	2.9
Duct Pressure Vessel ID (m)	1.5
Gas Duct Inside Diameter (m)	0.75
Gas Support Pipe Inside Diameter (m)	1.0
Insulation Thickness (mm)	100

3.1 PBR Primary System

The PBR reactor, as depicted in Figure 3-2, has a cylindrical core made from graphite blocks forming the core cavity. The graphite blocks also function as the neutron reflector, core heat sink and radial residual heat removal path when the active core heat removal system is not operating. During reactor operation the core cavity is filled with spherical fuel elements. Once critical core neutronic conditions are reached, nuclear heat is generated by uranium fission and transported to the steam generator by the circulating helium.

Each reactor unit is installed in a primary concrete cavity that supports the weight of the primary system pressure vessels. Surface coolers are installed on the inside of the reactor cavity silo to remove dissipated heat during normal operation and decay heat during shutdown and accident conditions.

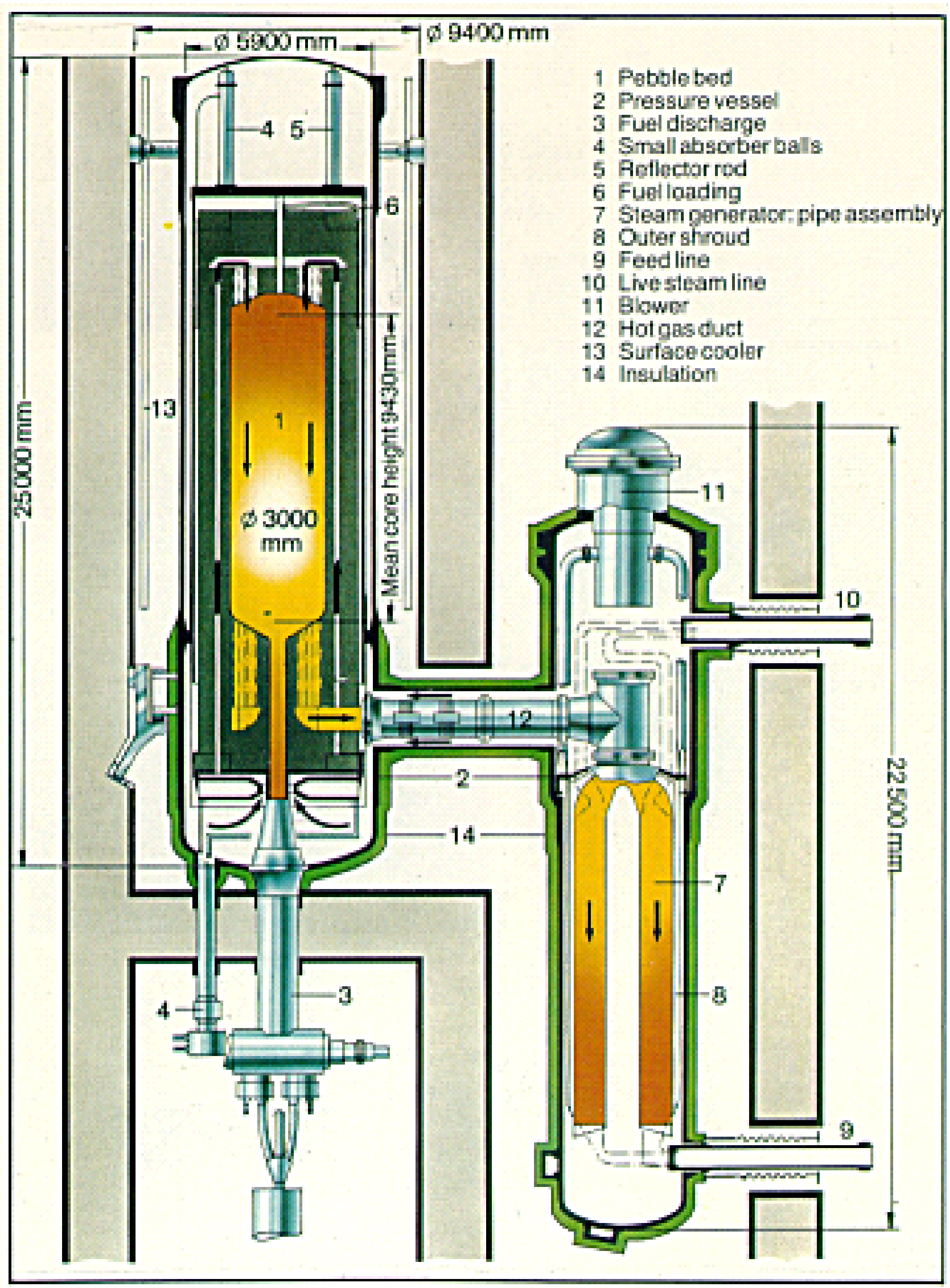
Figure 3-2: HTR-Module 200 Core and Graphite Component Layout



The reactor is designed for continuous refueling operation. Therefore, it operates with a low excess reactivity. Fresh fuel is loaded from the top of the core and discharged through the bottom. Each fuel element generates a small amount of fission heat as it passes through the core. Each fuel sphere makes multiple passages through the core before reaching maximum allowable burnup.

The primary circuit depicted in Figure 3-3 consists of the reactor pressure vessel, with the core internals, steam generator, shutdown systems and facilities for the charging and discharging fuel elements.

Figure 3-3: HTR-Module 200 Nuclear Steam Generating System



3.2 PBR Energy Conversion Plant

The energy conversion plant consists of a steam generator vessel with helical coil steam generator tubes, feedwater inlet and steam outlet nozzles, secondary coolant systems, turbines, feedwater, condenser, extraction steam, reboiler, and connections with process steam system. The steam is either sent to a multistage steam turbine for electricity generation or circulated through one or more re-boilers to produce high temperature process steam for industrial applications.

As shown in Figure 3-3 the reactor and the steam generator are in a side-by-side, staggered position offering the following design advantages.

- After a reactor shutdown, natural circulation of hot helium through the primary circuit is minimized by the thermal-hydraulic decoupling of heat source and heat sink. Therefore, there is no need to cool the steam generator after shutdown.
- The positioning of the steam generator beside and lower than the reactor permits simple, operationally-favorable upward evaporation.
- The substantial separation of the reactor core and the steam generator tubing by the concrete shielding walls of the reactor silo allows easy access to the steam generator cavity after shutdown for inspection and maintenance.

The steam generator for conventional conditions is designed as a once-through helical-coil tube and shell, with water/steam inside the tubes.

3.3 Barriers against Release of Radioactivity

The HTR-Module 200 uses fuel elements in which the uranium fuel is distributed among many small fuel particles. Each 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 edge zone.

The primary characteristic safety feature of the HTR-Module 200 is that 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. This safe confinement of radioactivity is assured by the design of the fuel particle coatings and the inherent upper limit of approximately 1600°C, the maximum possible fuel temperature under accident conditions.

The design of the coated particle fuel also provides key safety characteristics. 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 radiation or accident induced defects. The HTR-Module 200 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 burnup distribution and the core maximum fuel temperature distribution.

Some of the radioactive substances released from these defective particles are retained within the fuel pebble 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. The components of the HTR-Module pressure vessel unit, which includes the reactor pressure vessel, the steam generator pressure vessel, and the cross duct pressure vessel, are designed in such a way that through-wall cracks can be ruled-out. Because of the quality assurance measures taken, un-isolable breaks in the piping which connects to the pressure vessel unit are also highly improbable.

In the event of a break, which is nevertheless postulated, 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 released into the reactor building. Therefore, the HTR-Module 200 design did not place any leak tightness requirements on the reactor building to comply with accident dose limits imposed by the German Radiological protection ordinance. This was possible mainly because of the high radioactive retention capacity of the fuel particles. However, to minimize the impact on the environment, the reactor building is provided with a sub-atmospheric pressure system and a pressure relief system.

3.4 HTR-Module Inherent Safety Characteristics

The engineering configuration and nuclear design of the HTR-Module 200 is such that, even in the event of postulated failures of all active shutdown and residual heat removal systems, the fuel temperature stabilizes at approximately 1600°C. This margin to safety is possible because a temperature differential of approximately 750°K is maintained between the maximum allowable fuel temperature and the maximum operating temperature of the fuel elements in the HTR-Module reactor cores. Due to the negative temperature coefficient for the reactivity, this temperature differential assures that the reactor core shuts itself down before the temperature limit of 1600°C is reached. 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 physical processes. These include thermal conduction, radiation, and convection. The selection 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.

Active residual heat removal systems limit loadings on the passive heat removal components and structures. The design margins are selected such that active systems may fail to operate for several hours without exceeding the allowable limit.

The HTR-Module 200 primary system and core design and material selections reduce the safety requirements on the water/steam cycle and startup and shutdown systems. These systems are designed and operated as purely conventional plant items.

4.0 PBR DESIGN STATUS ASSESSMENT

The purpose of the Design Status Assessment (DSA) is to evaluate the overall HTR-Module design to determine its ability to successfully support NGNP completion. It is the intent of this determination to provide a meaningful representation of the design readiness of the HTR-Module design relative to NGNP requirements and the design maturity of the HTR-Module to be adopted in the U.S.

4.1 System Design Status Assessment Process

The HTR-Module DSA was performed via a systematic approach using quantitative and qualitative methods. Each HTR-Module baseline design area, listed in Section 4.1.2, was compared to a set of key design requirements for the NGNP and evaluated by system cognizant personnel for design compatibility with successful NGNP completion.

Key NGNP design requirements are delineated in Reference [4]. The key design requirements in this document represent those considered by AREVA NP Inc. to be applicable to PBR technology. They were screened from the NGNP Systems Requirements Manual [5] and Key Design Requirements for the High Temperature Gas-cooled Reactor Nuclear Heat Supply System [6]. The selection criteria used for this applicability screen are found in [4].

Qualitative evaluations of the design areas were performed by system experts. The process used was to solicit responses to facilitating questions from system design experts familiar with the HTR-Module baseline design. These experts provided their considered responses based on their knowledge, skill, and experience to qualitatively evaluate the subject design areas.

The questions used to facilitate both the design requirement and design assessment reviews are presented in Table 4-1.

Table 4-1: Questions Used to Facilitate HTR-Module Design Assessment

ASSESSMENT AREA	FACILITATING QUESTION
Alignment with NGNP Requirements	How does the design align with Next Generation Nuclear Plant (NGNP) Requirements?
Maturity/ completeness	What is the maturity and completeness of the design?
Design Gaps	Are there any noticeable gaps in the design?
Topical Area Concerns	Are there any concerns about the design related to certain topical areas of interest (e.g., ALARA, fire protection, internal/external hazards, separation/independence)?

ASSESSMENT AREA	FACILITATING QUESTION
Design Complexity	Is the design complex to the point that we should anticipate design and/or construction challenges?
Americanization of Design	What would be required to Americanize the design?
Design Challenges	Are there technological, manufacturing or procurement challenges to the design, including the need for R&D?
Other Pertinent Information	Is there other pertinent information regarding the design?

4.1.1 Baseline Design

The baseline design of the HTGR, for which this Design Status Assessment (DSA) was performed, is the HTR-Module as described in the Design Description Report. This document is the design baseline that includes plant technical configuration and operational information for the HTR-Module. This baseline design configuration includes plant design parameters, configuration, operation, and performance requirements/characteristics.

The baseline design provides for the following:

- Core power level of 200 MWt (for each of 2 reactors)
- Core inlet and outlet temperature of 250 °C and 700 °C respectively
- Primary Heat Transport mass flow and operating pressure of 85 kg/s and 6 MPa respectively

4.1.2 Scope of Systems Evaluated

The systems and structures were evaluated by the DSA are as follows:

1. Nuclear Steam Supply Facility
    a. Pebble Bed Reactor
        i. Reflector Rods
        ii. Small Absorber Ball System
        iii. Pressure Vessel Unit (RPV, SG Vessel, Cross Vessel)
        iv. Metallic Internals (RPV)
        v. Ceramic Internals (RPV)
    b. Circulator
    c. Hot Gas Duct
    d. Pressure Relief, Pressure Control
    e. Steam Generator (tube bundle)
    f. Reactor Cavity Cooling System

   g. Fuel Handling and Storage
   h. Helium Systems (He-Purification and supporting/connecting Systems)
   i. Reactor Building
   j. Reactor Building Annex
   k. Reactor Auxiliary Building
   l. Spent Fuel Storage (ISFSI)
2. Energy Conversion Plant
   a. Interconnecting piping, valves, vents and drains
   b. Reboilers, vents, drains, chemical injection
   c. Steam turbine generator with extraction ports
   d. Steam turbine auxiliary systems
   e. Condenser and cooling system
   f. TG Control system
   g. Step up transformer and HV electricals
   h. Turbine building, reboiler building
   i. Other auxiliary and support systems
3. Balance of Plant
   a. Administration and control buildings and facilities
   b. Power distribution
   c. Security
   d. Utility Systems
      i. Yard Electrical
      ii. Pipe Racks
   e. Gas Storage and Supply
   f. Fire Protection Systems
   g. Plant Water Systems
   h. Chilled Water Systems
   i. Cooling Water Systems
   j. HVAC Systems
   k. Site Preparation and Foundations
      i. Site Development
   l. Building foundations
   m. Buildings & Common Facilities
      i. Switchgear and Emergency Power Building
      ii. Central Gas Supply System Building
      iii. Control Building
      iv. Security Building and Equipment
      v. Gatehouse
   n. Control & Instrumentation
      i. Plant I&C
      ii. Plant control simulator
   o. Plant Communication

p. Radiation Monitoring
q. Fire Detection
r. Safeguards and Security System
s. Substation and Power Distribution
    i. Plant Electrical Distribution
t. Uninterruptible Power Supply
u. Emergency Power Supply
v. Auxiliary Power Supply

4.2 Assessment of HTR-Module against NGNP Requirements

Based on the DSA, as described herein, it has been determined that, other than exceptions noted below, the HTR-Module baseline design meets the NGNP requirements of [4]. The designators following each requirement area presented below detail the applicable Appendix number, Section and Requirement number within Reference [4].

Regulatory Requirements - *(A.1/2.1 through 2.6A, 3.1 through 3.4)*

The HTR-Module design must be "Americanized" to meet U.S. regulatory requirements (includes federal, state, local and industry codes and standards) and Project specific requirements based location and ownership.

Reactor Gas Outlet Temperature - *(A.4/2.1-14; B.1.10/2.1-6 and 10)*

The requirement that reactor gas outlet temperature be in the range of 750°C to 800°C is not met. There is also a requirement that supplied steam temperatures be in the range of 540°C to 630°C, which is achievable with the HTR-Module baseline design reactor gas outlet temperature of 700°C. Thus, the HTR-Module reactor gas outlet temperature of 700°C appears adequate. Current PBR technology is judged to be capable of providing 750°C to 800 °C reactor gas outlet temperature based on experience with past and present operational PBR reactors (AVR, THTR, HTR-10) as well as PBR reactor design concepts (HTR-PM, PBMR). AREVA NP Inc. proposes that early in the initial design phase a study be conducted to address this issue. The intent of this study would be to provide design values that would allow adjusting the core outlet temperature to an optimized value, considering all the top-level requirements imposed on the design.

Peak Fuel Temperature - *(A.4/6.1.3-2)*

The current requirement that peak fuel temperature not exceed 1600 °C under accident conditions is not met in an absolute sense. However, while 1600 °C is a widely used guideline for HTGR accident peak fuel temperature, it is not a real limit. The real limit is on fuel performance, which is only marginally dependent on temperature in this range. Significant fuel failure is only seen at much higher temperatures (1800 °C - 2000 °C). To determine the acceptability of a particular temperature transient, a statistical evaluation taking into account the time averaged temperature distribution of the whole core is needed.

Operating Lifetime of SG and Reactor Vessel - *(A.4/2.1-12; B.1.1/2.2-3)*

The requirement that all major components have an operating lifetime of 60 years (except the Main Helium Circulator - 10 yr. requirement) is not met. The HTR-Module was designed for 32 EFPY of operation (operating lifetime of 40 calendar years). All of the key components should be able to be redesigned for a 60 year operating lifetime or appropriate design considerations included to provide for replacement.

Passive Core Heat Removal - *(B.1.1/4/1-5;B.1.3/2.1.1-1; B.1.9/2.1-4)*

The requirement that the reactor system be designed to "provide passive residual heat removal during loss of forced cooling" is not met. The HTR-Module has a passive residual heat removal mode that maintains temperatures within design limits for 15 hours, which exceeded German regulatory requirements at the time. Beyond 15 hours, no unacceptable safety consequences result, but certain (non-fuel) temperature limits are exceeded. The Potential Advancements section of the Design Description Report proposes an alternate design that complies with the NGNP requirement.

The requirement that the reactor system be designed to "provide decay heat removal by passive means from the fuel to the reactor internals without reaching unacceptable fuel temperature during all DBA conditions" is met.

Adequate Shutdown Reactivity - *(A.4/6.1-4 and 5; B.1.7/2.1-2 and 2.1.1-1)*

The requirement that the reactor be shutdown by control elements and remain shutdown during the worst possible reactivity insertion is not met during the first core. During the first core, the temperature coefficient is sufficiently negative such that there is a possibility for the core to return to criticality at temperatures below 100 °C. This situation is discussed in detail in Section 5.6 of this report.

Spent Fuel Storage - *(B.5.1/3.1.1-16A)*

The requirement that there be 10 years of onsite fuel storage is not met. The design can be revised to provide for 10 years of onsite fuel storage. This is not considered a challenging design change and may be easily incorporated in the initial design phase.

Primary Loop Pressure Relief System Designed to ASME Code - *(B.1.1/3.1.5-3)*

Because the HTR-Module design is based on a German design, the primary loop pressure relief system was designed to German standards. For this reason, the requirement that the pressure relief system provide the primary coolant loop’s overpressure protection as required by ASME pressure relief code has not been confirmed. This situation will be rectified during the Americanization phase of design preparation.

NGNP Operation Following Loss of Secondary Heat Process - *(B.1.10/2.3-8)*

The requirement that the NGNP shall be designed to operate following loss of a secondary heat process, such as hydrogen production, and stabilize in the electricity generation phase could not be verified to be met. This is because there is not sufficient design information available. However, such a requirement is fundamental to all steam electric generating plants that accommodate load follow and step load changes such as loss a main feedwater pump. This technology, considered mature and well understood, can be incorporated into the HTR-Module design.

Power Conversion System Cost versus Efficiency/Reliability Balance - *(B.4.2/3.1-2)*

The requirement that steam conditions and cycle configuration be selected to result in a net generation efficiency of at least 42%, balancing cost with efficiency and reliability, is not met. Analysis of the HTR-Module baseline steam conditions and cycle configuration indicates a net cycle efficiency of 40%. The HTR-Module design will be revised in an effort to meet the NGNP requirement of 42%.

Instrumentation and Control Human Machine Interface and Equipment Monitoring - *(B.5.2/2.1-4 and 2.2-1)*

The requirements to a) apply Human Factors Engineering and Operating Experience to human machine interfaces and b) to provide equipment monitoring could not be verified to be met due to unavailability of sufficient design information. These requirements can be adequately addressed during the PBR design process.

Earthquake - *(B.5.3/2.13-8, 9 and 10)*

The requirements for earthquake accelerations given for the free field spectra cannot be verified to be met. The HTR-Module is based on German acceleration values, which are roughly similar to that of a USNRC Reg. Guide 1.60 except for the frequency region above 25 Hz.

Radioactive Waste and Decontamination - *(B.5/3.1.2-5 and 7)*

The requirements to a) provide redundancy to radioactive liquid waste system components and b) provide drying and vacuum capability to decontamination equipment are not met. These capabilities can be included in the design of the HTR-Module.

4.3 Assessment of HTR-Module Design Maturity

This qualitative process, performed by system design experts familiar with the HTR-Module baseline design, provided a broad assessment of the design status based on a set of facilitating questions. These questions were purposely composed at a sufficiently high enough level so as to solicit responses unencumbered by pre-loaded assumptions or low level restrictions. They are provided in Table 4-1.

The HTR-Module design is a German design that uses largely proven technology. This design met all necessary requirements of German nuclear regulatory authorities and has been rigorously reviewed by independent agents. Review of pertinent German design documents, including PBR technical requirements, accident, thermal-hydraulic and structural analyses, equivalent system description documents, process and instrumentation drawings, logic diagrams, and equipment specifications, indicate that the HTR-Module was in late preliminary to early final design stage. It is expected that the design will have to drop back to the end of conceptual design stage in order to complete a reconciliation with NGNP requirements and Americanization. However, the greater maturity of the existing design is still beneficial. It provides a more clear path for the completion of design, avoiding some of the key decisions, iteration, and redesign that are inherent in the normal design process during initial preparation. There will be challenges to mapping the German HTR-Module design directly into an American PBR design.

Foremost among such challenges is the need for the design to accommodate U.S. regulatory requirements, codes and standards and align with U.S. industrial practice (e.g., piping sizes, rebar sizes, pump and motor sizes and bus voltage); such accommodation and alignment is sometimes called Americanization of the design. AREVA NP Inc. has experience with mapping foreign power plant designs into designs that are adequately supported by all U.S. regulatory agencies. Such experience indicates that there is often little synergy between the foreign design regulations, codes and standards and U.S. regulations, codes and standards; this leads to extensive licensing verification efforts and design changes.

Other challenges, to the level of design maturity, include:

- Update to digital controls including control complexities associated with fuel handling system requirements.

- ASME High Temperature Reactor Code not yet final (will also need NRC approval); this may lead to design changes during design maturation.

- Due to insufficient German information, some design gaps may be revealed leading to design changes.

For the reasons discussed above it is clear that the NGNP PBR design must necessarily undergo some degree of regression from the near final design stage of the German HTR-Module on which it is based. Based on the information available to AREVA NP Inc., it has been estimated that an NGNP based on the HTR-Module should be considered to be in the late conceptual design stage. In order to progress to the point of early preliminary design, a reconciliation of the design to NGNP requirements and an initial round of “Americanization” would be necessary.

Though this design would be considered to be in the late conceptual design stage, it has certain advantages over other designs at this stage. Because the HTR-Module had progressed much further in Germany, a defined success path for major design decisions is largely available, which should eliminate or greatly reduce the need for multiple design iterations going forward. This could be of significant potential benefit, in terms of reduced schedule duration, engineering costs, and overall project risk.

4.4 Individual System Design Readiness Assessment Points of Interest

Below are presented several points of interest that were developed during the design readiness assessment activities conducted for each plant system. These items are composed of both design readiness and technology readiness issues. These points do not constitute the entire assessment for each system, nor is each system represented in the below list. For those areas and systems not discussed below, the general assessment results are consistent with the overall plant assessment presented in Section 4.3. The sub-headings under each system heading correspond to the Assessment Area from Table 4-1.

4.4.1 Reactor Core

From the information available for the Reactor Core design, it looks to be beyond the conceptual level and in the preliminary and detailed design phase. The design looks to be in the phase, where most of the operating conditions and requirements have been established.

Most of the operating parameters for the core design have been specified for normal operation. The design for startup has not been finalized and a few options are being assessed currently. Only a conceptual view of operations has been established.

The control and shutdown systems have been conceptualized and some detailed work has been performed, such as material and size. Exact operation and all components have not been identified. Most parameters for the control systems have been defined.

The general layout of the reactor core has been conceptualized and quantified. Most large components (core internals) have been designed to the point of function, capability, position and likely materials. Manufacture and vendor information is not known but for a small portion of the components.

Reactor physics design has been conceptualized and basic calculations and models have been provided. Enough detail exists to perform transient analysis of the core physics. Xenon effects have been taken into consideration. A long term stability analysis and error analysis has been performed for the core physics and thermal hydraulics.

Core thermal hydraulic properties have been calculated and analyzed using methods available at the time of the development of the HTR-Module SAR.

Fuel parameters of operation and dimensions have been calculated, although a final design for fuel has not been picked and further work is being done to evaluate this portion of the design. Burnup and enrichment has not been finalized although the current design has a set burnup and enrichment that was used for the initial calculations using UO_2 fuel.

Reactivity analysis has been done but work may need to be done here to upgrade design to NGNP requirements. This includes shutdown during the initial operation of the core and shutdown margin during operation and accident analysis.

Ceramic corrosion due to fuel element flow should be analyzed to look at effects over the life of the plant (60 yrs). This will need to be combined with helium impurity-driven corrosion of the graphite per NGNP requirements.

Fuel element wear due to interaction with other fuel elements, reactor wall, fuel handling equipment, etc. An analysis will need to be performed to demonstrate that the fuel can handle these loads, combined with operating loads, over the life of the fuel element.

Currently the design calls for shock absorbers at the CRDM that protect the control elements in case of reactor trip. There are also secondary shock absorbers installed at the bottom of the control element channel to minimize damage to the graphite core assembly in the unlikely event of a control rod chain failing. These secondary elements deform plastically and can be replaced as necessary.

Load changes during the first few months of operation will need to be defined. These may be different than normal operation due to low amounts of fission products and the imbalance of fresh fuel vs. spent fuel. This consideration will need to be done in parallel when defining the startup procedure. This may also impact load following capability during the applicable time period.

4.4.2 RPV and Supports

It is not known how much performance analysis has been performed to demonstrate that the plant performs as intended. From a component design point of view, very few of the materials of construction were identified in the documentation, and each of these would need to be changed to meet the rules of ASME III Division 5 once it is published, though it is noted that components that meet the requirements of ASME III NB, it is likely to meet these requirements. Furthermore, it cannot be concluded at this time that the design of the components will satisfy these requirements because no particular details of the component design were provided, and the Construction Code rules against which the design will be evaluated are preliminary.

4.4.3 Steam Generator

The geometrical complexity of the tube bundle (helical part +compensation bundle) together with its supporting system should be taken into account. Design and manufacturing are likely to be difficult due to this complexity, though several steam generators of this type have been built in the past (for THTR and various tests). The SG manufacturer’s helical design experience should be considered.

4.4.4 I&C

For analog control systems, the I&C design is complete to the extent of preliminary design. It is worth noting, however, that: there is only 1 level of protection (operation control system then RPS).

All of the key functional requirements for control of safety systems and the NHS automation are specified. Logic diagrams and P&IDs exist for protection system and NHS system controls.

The design was developed in the late 1980s, and thus does not incorporate the latest digital control system technology; therefore the system design will need to be updated to a primarily digital platform.

The control system functional requirements for the nuclear island BOP are not as fully developed, as would be expected for a project that has not fully completed preliminary design.

The control system design for the energy conversion facility has not yet been developed.

An upgrade to the use of digital controls and automation will impact the licensing of the plants. Digital controls are a key focus of the NRC and gaining design approval is a complex task. However, there is enough precedence in this area to safely manage the associated risks.

4.4.5 Fuel Handling System

The required Burnup Measurement System (BUMS) performances are challenging and have not been fully demonstrated yet. Margins of the BUMS performance (10 s measurement with 5% statistical error required) should be verified.

ALARA requirements are likely to have an important impact on the design in order to make maintenance on the fuel handling system equipment possible during reactor operation. Even though they are not specifically reported in the SAR, these are well covered in the design, supported by THTR operating experience.

4.4.6 Energy Conversion Plant

The integration of a nuclear heat source into a process environment does introduce several challenges that will need to be addressed. Many of these are associated with the chemical releases, explosions and fires that impact on the safety of the nuclear island. For the Energy Conversion Plant (ECP), the part of the plant that takes main steam from the steam generator and produces electricity and process steam, one of the challenges associated with integrating the HTR-Module/ Steam Generator into a process facility is handling upsets in process steam demand. Refineries and other petrochemical plants trip more often than nuclear plants. They also restart more rapidly. Unlike a nuclear plant trip, this would not be a total shut down of steam and feed water flows, this would be a step function impulse into the HTR-Module. A steam bypass system, common to conventional cogeneration systems, should be able to manage steam system shutdowns without tripping steam generators. Steam dump heat balances were run as part of the ECP trade study, which shows what heat sink is needed to avoid impacting reactor operations. The system should be designed to use all the steam in the steam turbine if the process interface refuses steam and condensers sized to handle the process heat load shedding.

The design described in the SAR and that was used as the basis for the FOAK and NOAK cost estimates uses a steam generator outlet pressure of 19 MPa (2755 psia) and a steam turbine throttle pressure of 18.0 MPa (~2600 psia). The AREVA Review Team surveyed manufacturers of steam turbines and found a limited number of suppliers of steam turbines that manufacture small turbines (150 – 300 MWe) that can accommodate high inlet pressures. Since this steam throttle condition is not commonly used at facilities operating in the US, a reduction of the Steam Generator pressure to 17.3 MPa (2500 psia) and a corresponding reduction in the turbine throttle pressure could bring the design closer to an 'American' standard. However, since these high pressure steam turbines have found wider use and acceptability in Europe, and considering the potential for higher cycle efficiencies at the higher temperatures, it would be more economically beneficial to utilize the higher throttle pressure turbines as much as possible. Whether a particular facility wishes to retain the 'American' standard would be made on a case by case basis by the owner/operator of the facility. This decision could be based on a desire to use the same type of turbine throughout a facility.

The other changes that will be necessary to transform the design into one that is compliant with American standards is the application of US codes and standards, as well as the generation of power at 60 hertz instead of 50 hertz as in the SAR design.

The recommendation to use the same HTR-Module ECP configuration (i.e., two HTR-Modules feeding a single turbine) for the FOAK and NOAK cost estimates is based on a conservative assumption about the desire to demonstrate the NOAK-sized steam turbine and to confirm the ability to control any instabilities that could be introduced as a result of two independent reactors supplying a single turbine. A more cost conscious approach for a demonstration FOAK facility would consist of a single HTR-Module with an appropriately sized steam turbine. The concerns with this scaled-back arrangement are that the small-sized turbine that would be needed to match the reduced steam generation would likely be of a unique, and possibly costly, design that would have no future impact on the development of the cycle. In addition, concerns about the ability to control a two module HTR unit supplying steam to a single turbine would still remain.

Additional study should be undertaken in the next phase of the PBR program to identify the capital cost impacts of scaling back the FOAK design to use a single HTR-Module supplying steam to a turbine sized to accommodate the steam flow from two HTR-Modules. While questions about the controllability of the NOAK-design would remain, these questions should not impact the safety case due to the intrinsically safe nature of the pebble bed reactor. Variability in the steam production by two independent HTR-Modules is not expected to be excessive. A sufficiently robust and conventional steam supply control system should be capable of smoothing any irregularities that might arise. This type of control system does not need to be demonstrated in a FOAK.

Unlike the steam supply control system, a high throttle pressure steam turbine to handle the amount of steam generated in an NOAK unit should be included in the FOAK facility, even if it is oversized for a single HTR-Module steam supply. The advantages of manufacturing and operating this smaller-size turbine under these conditions is expected to outweigh the efficiency losses that will result from running at half-throttle. The alternative – using an even smaller sized turbine to match the single HTR-Module steam flow – is counterproductive since it would very likely be a unique design that would not be expected to be utilized in the full-size NOAK facilities.

5.0 EXAMINATION OF KEY PBR ISSUES

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. For these issues, an assessment has been conducted, beginning with a description of exactly what each issue is, and sometimes more importantly what it is not. After the description, an assessment is made as to the potential impacts of the issue on implementation of the PBR technology. As appropriate, design updates recommended or required to alleviate identified concerns are described. Supporting data reviews and/or scoping calculations are provided as necessary to fully explore each issue.

5.1 Impact of Stochastic Nature of PBR Core

The design philosophy behind the PBR core arises from the desire to utilize nuclear fuel in an optimal way such that the fissile and fertile materials in the core are burned up in a highest achievable fashion, leaving behind as little as possible amount of actinides, so that nuclear waste disposal and proliferation issues can be kept to a minimal level. Furthermore, the safety margins in the PBR core are increased further by maximizing the passive heat transfer mode and large core thermal inertia, allowing more time for operator action.

Based on these design philosophies, the PBR fuel is able to achieve superior burnup while minimizing actinides production and fission product releases as advancement over the conventional light water reactor fuels. This is achieved through high-quality engineering and manufacturing processes of the PBR fuel pebbles thus attaining a more homogeneous core with flatter neutron flux and temperature. As results, the power peaking factor and hot-spot temperature in the PBR core are much less pronounced as compared to those in the LWR core.

In order to achieve a maximum and more uniform burnup, each fuel pebble in the PBR is continuously cycled through the core using a fuel handling (or fuel transport) system, until the desired FIMA (Fissions per Initial heavy Metal Atom) level has achieved with a precise measurement technique. Although the location of each of the 360,000 fuel pebble in the HTR-Module core is not known exactly; however, its expected flow path and residency time can be pre-determined statistically with known uncertainties correlated from experimental data as discussed in Section 5.1.1. The uncertainty on the hot-spot and peak fuel temperature in the PBR core using statistical approach have been confirmed with the measurements obtained by the AVR melt-wire experiment, which is discussed in detail in Section 5.1.4.

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 (360,000 in the HTR-Module), only the average behavior can be investigated. The is how well the average properties represent the power peaking and maximum fuel temperature in the pebble bed reactors specifically for the HTR-Module design.

This uncertainty in the physical parameters and the location of specific pebbles results in uncertainties in the core power and temperature distribution are assessed in the following sections.

5.1.1 Uncertainties in Pebble Movement in the PBR Core

A unique feature of the pebble bed HTGR is the continuous circulation of fuel pebbles. The pebbles are randomly packed within the core. This allows for small excess reactivity in the core for power control. The pebble flow behavior of a pebble bed high temperature reactor is important for temperature distribution within the core (both fuel and coolant) and the loading scheme. For this reason, the flow behavior of the pebble bed has been investigated since the initial pebble bed HTGR development. [7] [8]

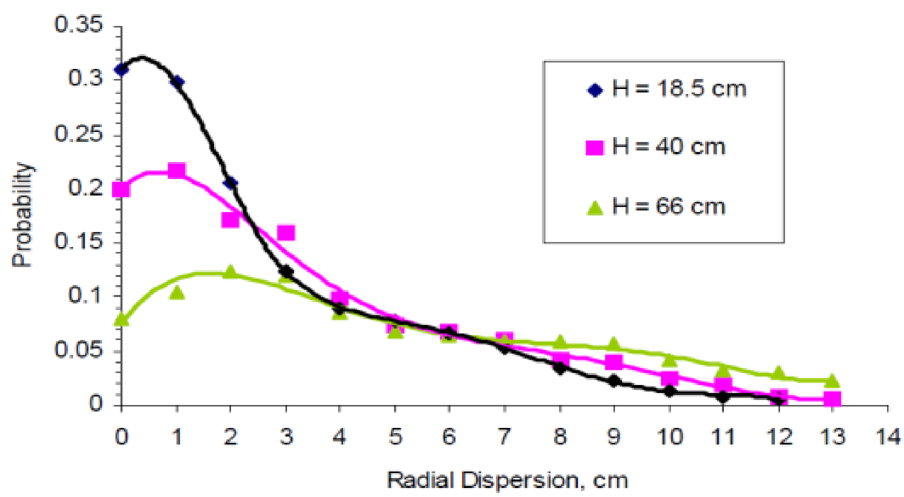
Extensive AVR and model experiments with confirmation from numerical simulations have been performed to investigate the pebble flow behaviors in the PBR. These investigations can be grouped into studies of the feeding of pebbles and in-pile behaviors. Assessments of the pebble movement behaviors for the HTR-Module are discussed in the following sections.

5.1.1.1 Feeding of Pebbles

After landing at the top of the fuel pile, the fuel pebble can potentially but not necessarily roll away towards the reflector from the center of core where it is dropped. Since this is a random process, the initial position of a fuel pebble is not known deterministically, before it is buried by layers of fuel pebbles.

In the MIT pebble-drop behavior study [8] it was determined that the radial spreading of pebble at the top of the fuel pile is a complex function of drop height, dropping rate, location of the dropping point, and the angle of repose of pebbles. In the MIT experiment, pebbles were dropped onto a flat surface made up of several layers of pebbles. The probability of initial location decreases with increasing radial dispersion as shown in the Figure 5-1, which means that it is more likely for a ball to come to initial stop near the collision point. This demonstrates the basic predictability of the pebble drop process.

Figure 5-1: Probability Distribution Function of Pebbles



In an actual reactor, the top surface of the pebble bed will not be flat. Depending on the arrangement of fueling pipe(s) one or more cones will form on the top of the reactor. The shape of these piles evolves so that the resulting distribution of loaded pebbles corresponds to the distribution of pebble flow within the pebble bed.

AVR was the first pebble bed reactor built, which contained one central fueling pipe and four satellite pipes feeding pebbles into the core. Because of this arrangement, in-core inspection in 1984 showed that the surface of the pebble bed asymmetrically contained one larger central fuel pile and four smaller fuel piles surrounding the central one [9].

Since the HTR-Module design uses a single central fueling tube for one fuel zone unlike that of AVR, it is expected to have fuel loading of one cone-shaped pebble pile in the center on top of the pebble bed surface with a uniform pebble dispersion probability.

5.1.1.2 In-Pile Behavior

The important parameters that describe pebble flow behavior inside the fuel pile are pebble path and relative flow velocity through the bed. Experimental data show that general behavior of pebble bed does not obey the granular flow theory, which would have predicted complete mixing of pebbles near the bottom of the core. As the pebbles travel downward, they exhibit little or no lateral diffusion. Although the pebbles flow along flow-lines on a global scale, but on a meso-scale level the movement of pebbles is intermittent in nature. The pebble flow dynamic can be described as follows. Starting from an initial core, a void is created at the bottom of the heap after retracting several fuel pebbles from the discharge tube. A small stable arch may form above the discharge tube, which can remain stable for a longer time. At a sudden moment, after retracting a few more pebbles from the discharge tube, the pebble arch formation becomes unstable and the void is collapsed and filled with pebbles. These fuel discharge voids are relatively small, and the study shows that no large voids can be formed and collapsed thus causing reactivity perturbation during reactor operation [10].

The positions of the pebbles can be determined by the flow lines and the random dropping from the top of core. However, due to the MEDUL fuel strategy [11] chosen for the HTR-Module, where fuel pebbles pass through the core several times, there are many possible combinations of flow trajectories resulting from each core pass with varying probabilities. Because of the stochastic nature of the in-pile flow behavior and pebble dropping, the probability that a fuel pebble goes straight through the hottest or highest power region of core several times is extremely low.

The pebble flow behavior has been studied in many pebble-bed experiments. Residence spectra have been determined for different sets of parameters. The important parameters that have been investigated are the core height, diameter of discharge tube, cone inclination, and ball weight. The major findings of the different studies are: [7][8][12]

- The pebble travels along a streamlines as it moves downward to the bottom of the core. Nevertheless, the motion of individual pebbles is random on a micro-level. The flow is well organized and not chaotic. Therefore, if the initial radial position of a pebble is determined, the residence time can be predicted with high accuracy.
- The average streamlines are related to the radial positions of the pebbles, but they are all parabolic curves except for the central position Figure 5-2 [9].
- The streamlines do not cross each other.
- Pebble flow velocities are slower near the top of the core, but increase sharply towards the de-fueling cone as shown in Figure 5-3 [7].
- The ratio of core diameter to pebble diameter, discharge tube diameter and discharge cone inclination are the main parameters that affect the pebble flow. The pebble diameter and specific density have no impact on the flow. (Note that THTR has a bigger core diameter and smaller inclination angle compared to HTR-Module.)

Figure 5-2: Flow Lines and Potential Size of Stagnant Zone in Outer Core of AVR

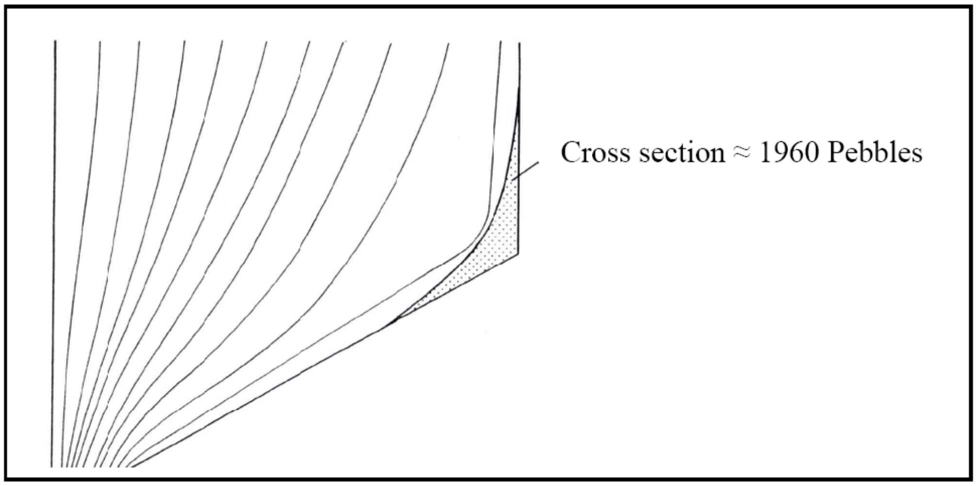
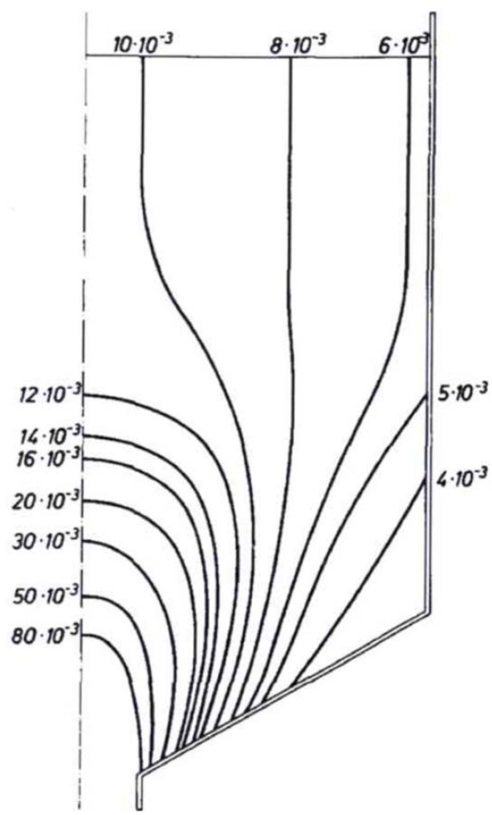


Figure 5-3: Pebble Flow Experimental Results for THTR – Isotaches: Curves of Equal Vertical Pebble Velocity



However, the AVR experiments have shown that the inner core fuel pebbles do not flow as uniformly as compared with those results obtained from the mode-scale experiments [9]. Reasons for this difference could be due to different friction factors in a real HTGR core (due to a radial temperature distribution instead of a constant temperature) and the control rod housings protruding into the core to a depth of half the core radius. Therefore, the pebble flow decreases near the reflector and control rod housing walls..

Pebble flow studies for the THTR in Germany [7] and flow simulations performed for the PBMR annular core design [8] show that pebbles flow is essentially along streamlines in the upper cylindrical section of the core move laterally within a few pebble diameters. Mixing of pebbles is only noticeable in the lowermost de-fueling cone section, in a neutronically insignificant region of the pebble-bed core. Typically, fuel pebbles located within 1½ pebble diameters of the side reflector will come in contact with the wall during their downward flow through the core; and thus, have a longer core residence time relative to the pebbles located farther away from the side reflector wall. [10] Therefore, in order to reduce pebble flow resistance at the reflector wall, indentations were added in the AVR and THTR reflector walls.

The model-experiment studies have shown that the pebble velocity profile is also affected by the defueling cone inclination and core height to diameter ratio (H/D) parameters [10]. That is, pebble flow behavior becomes more uniform with increasing cone inclination. A cone inclination of 30° was chosen to be the optimal value with respect to the neutronic physics requirements and pebble flow behavior for the THTR. Furthermore, the pebble velocity profile becomes more uniform with increasing H/D ratio. However, for core heights greater than 0.8 times core diameter, the influence of the core height becomes negligible. The values of H/D adopted for the AVR and THTR are 0.8 and 0.9, respectively.

Since the HTR-Module core design consists of one homogeneous fuel zone, and meets both criteria that H/D ratio > 0.8 and de-fueling cone inclination angle >30°; a uniform pebble flow velocity profile can be assumed for the HTR-Module core.

5.1.1.2.1 Mixing and Stagnation Zones

Model-scale experiments at the MIT (both 2D and 3D), Figure 5-4 [8], at PBMR in South-Africa, and at INET in China [13] have confirmed that even in a core with two distinct fuel zones, the mixing zone is constrained to a small size (4-5 pebble diameters maximum) due to the laminar flow behavior. These results have confirmed numerical simulations.

Investigation performed at INET in China [13] showed that the size of the stagnant zone decreases as the recirculation continues, which means that is related to running time [13]. Under the experimental conditions at INET, a stagnant zone is present. It is suggested by the authors that “instead of a sharp corner transitions from the rectangle prime section to the cone base section of the vessel, a gradual transition along the boundary curve of the stagnant zone would be favorable for avoiding the stagnant zone” [13]. The stagnant zone is a problem that can be solved by adapting the reflector geometry and is not inherent to the pebble bed design. The Australian Atomic Energy Commission (AAEC) reports that the residence time within the stagnant zone is decreased dramatically with an angular transition between cylinder and cone (5 % of the radius of the cylinder) [10].

Experiments show that a stagnant zone is more likely to occur at the lower cone section of the cylindrical reflector, if the inclination angle is less than 30°. However, these experiments have not been performed in the HTR-Module operating conditions.

The presence of a stagnant zone has also been studied during the ANABEK experiments in AVR [9]. During the first loading of the initial core, several pebbles with differing color where added to the transition zone between the cylindrical part of the core and the cone (inclination 30°) in order to demonstrate that no stagnant zone occurred. The ANABEK experiment result showed only a small reduction of the velocity was detected, even after the last “marked” pebble was retrieved after circulating 18 % of the core volume. The ANABEK experiment demonstrated a very uniform velocity profile existed in the core. The results of the ANABEK experiment have been interpreted using the HTR-Module parameters [14]. It is concluded that in the case of HTR-Module geometry no stagnant zone occurs.

Figure 5-4: Experiment at MIT with 2-Zone Core

5.1.1.2.2 Crystallization

At very slow discharge rates, the pebbles can start to arrange in a perfectly arranged manner called crystallization, which can block pebble flow. Experiments have shown that continued circulation of the pebble bed at a slow rate for a long time results in an almost homogeneous, crystal-like pebble layer throughout the whole wall of the container [10]. Gradually more and more crystal-like layers are formed inside the core, thus disrupt pebble flow through the core. In order to avoid the crystallization of pebbles along the reflector, the reflector is slotted with indentations to enhance random motions of the pebbles near the wall. These perturbations force the pebbles to move slightly radially on the way down. Due to the indentations in the reflectors wall surface, crystallization is completely avoided in the HTR-Module.

5.1.1.2.3 Bridging

For some geometries, the pebble bed can exhibit a tendency of dome void formation bridging over the discharge tube, which causes blockage and disruption of pebble flow. The probability of formation of such a dome is mainly a function of the number of pebbles needed to achieve such arrangement, which depends on the ratio of the fuel discharge tube diameter to the pebble diameter [10]. No domes are expected to form when the fuel discharge-tube-to-pebble diameter ratio is 5.0 or greater [10]. The fuel discharge tube of the AVR-reactor has a diameter of 50 cm, corresponding to a diameter radio of 8.3, thus provides a sufficient safety margin to prevent dome formation [10]. No dome formation was detected to occur during the operation of AVR. The inner radius of the discharge tube for the HTR-Module is 60 cm, which corresponds to a tube-to-pebble diameter ratio of 10. Therefore, no dome formation is expected to occur in the HTR-Module core.

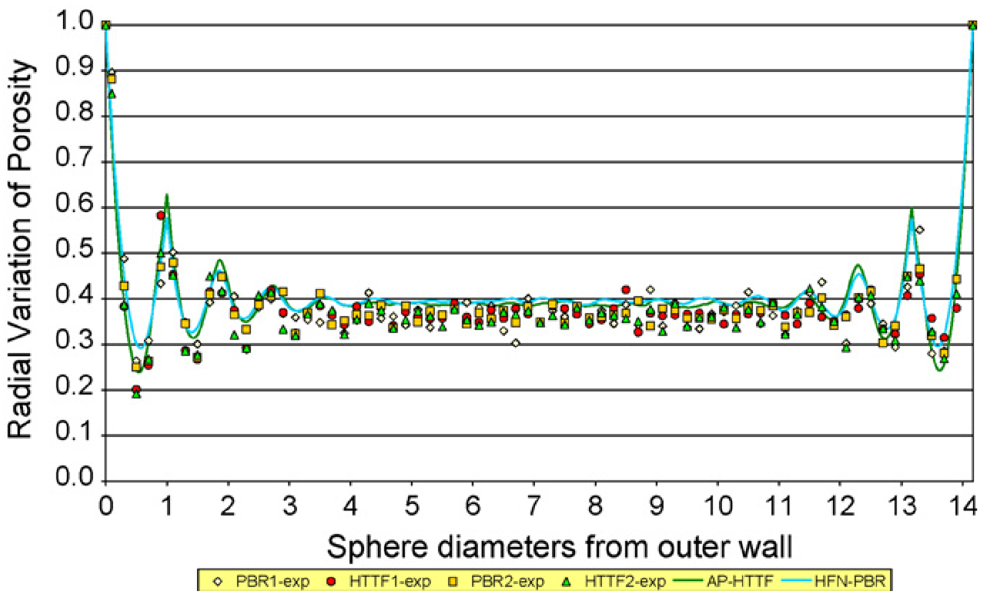
5.1.2 Uncertainty in Pebble Bed Packing Fraction

The in-pile pebble flow behavior results in a statistical packing of spheres, which can be considered as a combination of small regions in which the spheres are packed in different possible ways (hexagonal or cubic). The statistical pebble packing fraction is about 0.61 [10]. The AVR inspections showed a packing fraction of 64-65% [9]. Operational experiences from AVR and THTR have demonstrated that the void factor remained constant during fuel shuffling operation; otherwise, reactivity and power excursions would have occurred and been observed [10].

In a pebble bed reactor, the packing fraction (or porosity) varies sharply near the wall, where the geometry of the packing is interrupted. As a result the coolant flow velocity profile inside a packed bed can be distorted near the wall, reaching a maximum in the near-wall region. This phenomenon is known as flow or wall channeling. Wall channeling may have a significant impact on heat and mass transfer in packed beds. It may also lead to a non-uniform temperature distribution at the outlet of the bed. Thus, knowledge of the porosity distribution within a packed bed is thus important to any proper analysis of the transport phenomena in the bed.

The comparison between the radial variations in porosity of the experimental results from PBMR for the High Temperature Test Facility (HTTF) and the PBR and the results of the analysis of the numerically packed HTTF and PBR are shown in Figure 5-5. It can be seen that the experimental results and the numerical results are in good agreement near the walls and in the middle of the core. [15]

Figure 5-5: Annular PBR Radial Variation in Porosity for Physical and Numerical Experiments



From the comparisons between the experimental and numerical results, it can be concluded that the statistical analysis results concerning pebble flow and porosity distribution in the core are acceptable representation of the actual packed core in the HTR-Module.

5.1.3 Uncertainties in Power and Temperature Profiles in PBR Core

In a pebble bed reactor, fresh fuel is introduced at the top of the core, circulated and reloaded until the desired target burnup is reached; therefore, the core has a slightly lower burnup profile in the upper part of the core.

Furthermore, in the HTR-Module design, the coolant flows from top to bottom so that the upper region of the core is cooler than the lower. This leads to a power profile peaking near the top of the core, away from the hottest gas region in the core bottom. For HTR-Module reactor with cylindrical core, the power density peaks in the outer radial regions of the core next to the outer graphite reflector.

In a PBR, an average core outlet gas temperature is generally specified as the reference parameter for the hot coolant temperature. In the AVR, it is calculated from the core power, coolant inlet temperature and mass flow rate, since direct measurement of the gas temperature in the core is not possible. In the HTR-Module, an average of hot-gas temperature measurements in the hot-gas mixing plenum below the core is used as the reference hot coolant temperature.

Another source of uncertainty in HTGRs is the determination of the helium coolant mass flow rate. Since there is no direct measurement available, the flow is determined by correlating the pressure drop over the circulators using their characteristic curve.

There are a couple of phenomena that may influence the flow distribution in the core. The shape of the core, porosity distribution in the core, and the inlet and outlet conditions are the geometrical factors that may also play a role.

Another source of uncertainty in the PBR parameters is the imprecise knowledge on the fraction of core bypass flow. Due to the complexity in geometry, flows through the gaps and channels inside graphite reflector and support structure can only be analyzed by using the 3D CFD code.

A number of experiments and numerical simulations have been performed in order to quantify these uncertainties in terms of statistical temperature distributions or deterministic hot-spot calculations. Section 5.1.4 discusses the evaluation of AVR melt-wire experiment and its statistical results on temperature distributions. Section 5.1.5 discusses hypothetical simulations of clustered high-reactive fuel pebbles as the deterministic bounding values for the hot-spot analysis in PBR.

5.1.4 AVR Melt-Wire Experiment

In 1986, an experiment was conducted in AVR to determine the radial distribution of the maximum hot gas temperature in the core [16]. The design of the AVR introduced upward coolant flow in the core, and therefore the maximum gas temperature occurred at the top of the core. In the melt-wire experiment, 190 labeled monitor pebbles were selectively loaded into top of core, each containing 20 fusible wires with different melting points ranging from 655°C to 1280°C. The melt-wire pebbles were A3 matrix graphite spheres that did not contain fuel, thus the measured results were characteristic of the fuel surface temperatures. At the time of the experiment, the AVR was operating at full power with a nominal coolant outlet temperature of 950°C.

After discharge and post-examination, the maximum temperatures were determined from the X-ray melting patterns in the monitor pebbles. The radial temperature distribution in the core was then correlated using the measurements. The X-ray results showed that there were a total of 21 out of 144 pebbles detected with maximum temperatures of at least 1280°C. This exceeded the predicted maximum surface temperature of 1150°C for the GLE-3 type fuel pebbles by 130°C under the core coolant outlet temperature operating condition of 950°C [17].

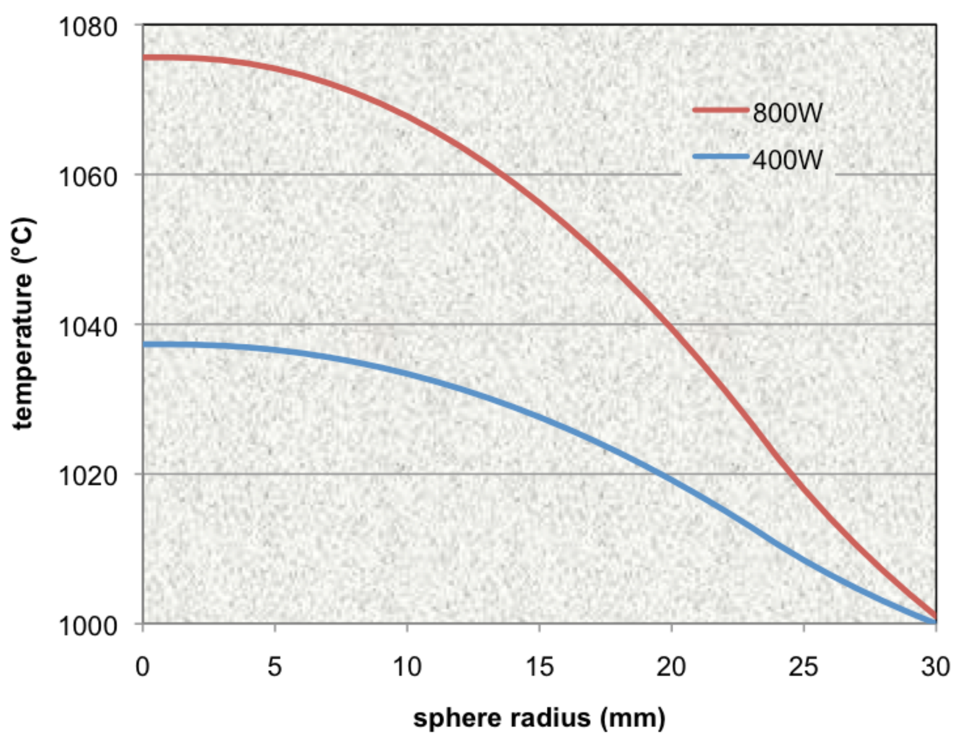
The unexpected differences in the maximum surface temperatures of monitor pebbles and the accepted nominal core outlet temperature can be attributed to the special design features of the AVR, in particular the presence of inner and outer core large core bypass flows, and the inherent behaviors of the pebble bed reactor. The AVR core was consist of the inner and outer regions differentiated by the locations of the five fuel feeding tubes. During normal operation, fresh or highly reactive fuel pebbles were fed into the outer core via the four peripheral tubes; whereas, highly burned pebbles were fed into the inner core via the central feeding tube.

A statistical analysis has been performed to determine the radial fuel surface temperature distributions in the AVR core based on the melt-wire temperature measurements. Furthermore, CFD studies were performed to quantify core flow bypass effects on the maximum fuel surface temperature. The results of these analyses are discussed as follows.

5.1.4.1 Temperature Distributions of Fuel Pebbles in AVR

Since each fuel pebble contains both fissile and fertile material, its temperature profile depends on the power density, heavy metal loading, and burnup. Figure 5-6 shows the fuel temperature distributions calculated based on the assumptions of typical heat conductivity of 25 W/m-K, 400 W per pebble for the inner core, and 800 W for the outer core. As shown, the maximum fuel center temperatures for a typical fuel pebble in the inner and outer core are 37.3 and 75.5°C higher than the respective surface temperatures.

Figure 5-6: Temperature Profiles in a Typical AVR Fuel Pebble of 400W/800W in the Inner/Outer Core

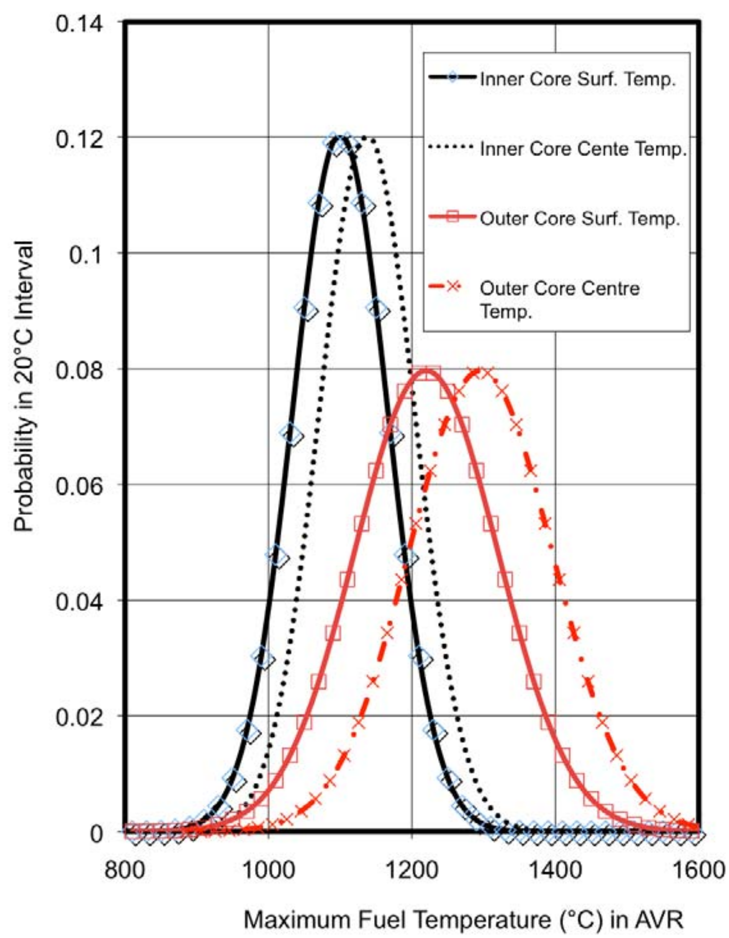


A careful analysis conducted on the AVR monitor pebble discharge times determined that 66% of the 144 melt-wire spheres passed through the inner core of AVR and 34% through the outer core. The underlying fuel temperature distributions were determined by constructing a Quartile-Quartile (Q-Q) plot based on the normal probability distributions. This was done by ordering the histogram data, calculating the probabilities in quartiles, and then determining the underlying normal or Gaussian distributions. The least-square fit of the melt-wire temperature data yields two Gaussian distributions with a mean fuel surface temperature of 1100 ± 66°C corresponding to the inner core and 1220 ± 100°C to the outer core [18].

Note that these temperatures only last for a limited time on the fuel spheres, since the temperatures decrease steadily as they move downward in the pebble bed. The position where each pebble entered the core at the top was determined by distinguishing between the inner and outer core spheres, and by correlating the time each pebble spent in the core with experimental pebble flow data derived for the core.

Using the calculated fuel temperature profiles given in Figure 5-6, the overall Gaussian distributions defining the variations of fuel pebble maximum surface and center temperatures in the inner and outer core regions of AVR have been determined and correlated statistically. As shown in Figure 5-7, the maximum fuel surface temperatures for the inner and outer core regions have the mean temperatures of 1100±66°C and 1220±100°C, respectively. And the maximum fuel center temperatures for the inner and outer core regions have the mean temperatures of 1137±66°C and 1296±100°C, respectively. [18]

Figure 5-7: Gaussian Distributions of AVR Fuel Pebble Maximum Surface and Center Temperatures in the Inner and Outer Core Regions



Therefore, from the probability density distributions shown in Figure 5-7, we can conclude that the probability of AVR fuel temperature exceeding 1600°C is less than 0.001 in normal operation.

Since the HTR-Module core design consists of one homogeneous fuel zone with an average core outlet coolant temperature of 700°C, it can be concluded that the maximum fuel surface temperature would also be a Gaussian distribution with a lower mean value.

5.1.4.2 Core Bypass Flows in AVR

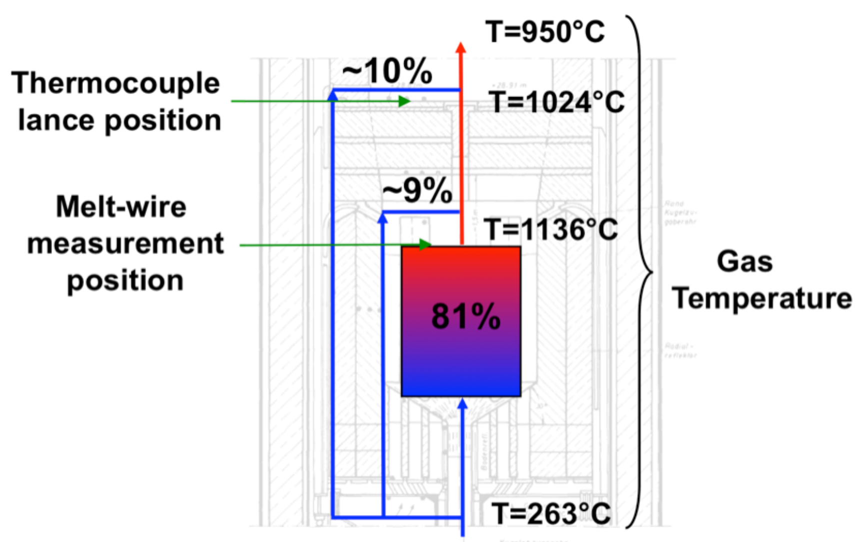
In an ideal case, all the coolant flows through the pebble-bed reactor core in a uniform flow distribution. It is generally accepted that calculations for the AVR were based on the assumptions with no bypass flow and no non-uniform effects. In reality, the flow through the AVR core is directly influenced by two major factors: the flow bypassing the core and the non-uniform flow distribution in the pebble bed itself.

The geometry of the AVR was investigated in detail to identify all the possible flows that could bypass the core. The following main possible bypass flow paths were identified: [19]

- Cooling flow through the control rod guiding boreholes in the reflector noses, exiting through the horizontal holes drilled in the top plugs of these boreholes
- Bypass flow between the outer carbon structure and the metal shroud reintroduced at the locations where the fueling lines end in the borings of the side reflector (below top reflector)
- The engineered bypass pipes that reintroduce flow just below the steam generator (above top of the core ceiling structure)

Figure 5-8 shows the results of coolant temperature distribution based on the CFD studies of coolant mass and heat transport in the AVR reactor taking account of the 19% bypass flow outside of the core [18]. In this calculation, the core average outlet temperature is 1136°C, which is consistent with the AVR melt-wire average temperature of 1140°C.

Figure 5-8: Estimation of AVR Core Bypass Flows From Measurements



In another study by PBMR [20], 3-D neutronics thermal-hydraulics analyses were performed to analyze the AVR melt-wire experiments, utilizing the coupled VSOP99-STAR-CD model, which included both core bypass flow and neutronic feedback effects. The goals of this investigation were to quantify the amount of bypass flows in the AVR and the relative effect they might have on the fuel temperatures in the core and the temperatures measured.

The results obtained by the PBMR analysis are summarized as follows:

- The flows through the control rod holes estimated by this model were in the order of 5-6% of the total flow in the reactor. The effect on the maximum gas temperature is about 44°C.
- The annulus flow was estimated at 10% based on an average outlet temperature derived from the measured temperature of the lance in the AVR. This bypass flow accounts for an additional increase in the maximum gas temperature of about 92°C.
- The wall channeling effect increases the core coolant outlet temperature by 15°C.
- The effect of the radial power distribution, due to two different fuel types loaded in AVR where fuel with the pebbles of UO_2 TRISO particles with higher enrichments were loaded to the outer core, raises the maximum coolant temperature by almost 90°C.

Table 5-1 [20] summarizes the temperature effects due to control-rod nose and annular bypass flows, wall channeling, and power distribution.

Table 5-1: Maximum Core Coolant Temperature Results due to Bypass Flows and Channeling Effects

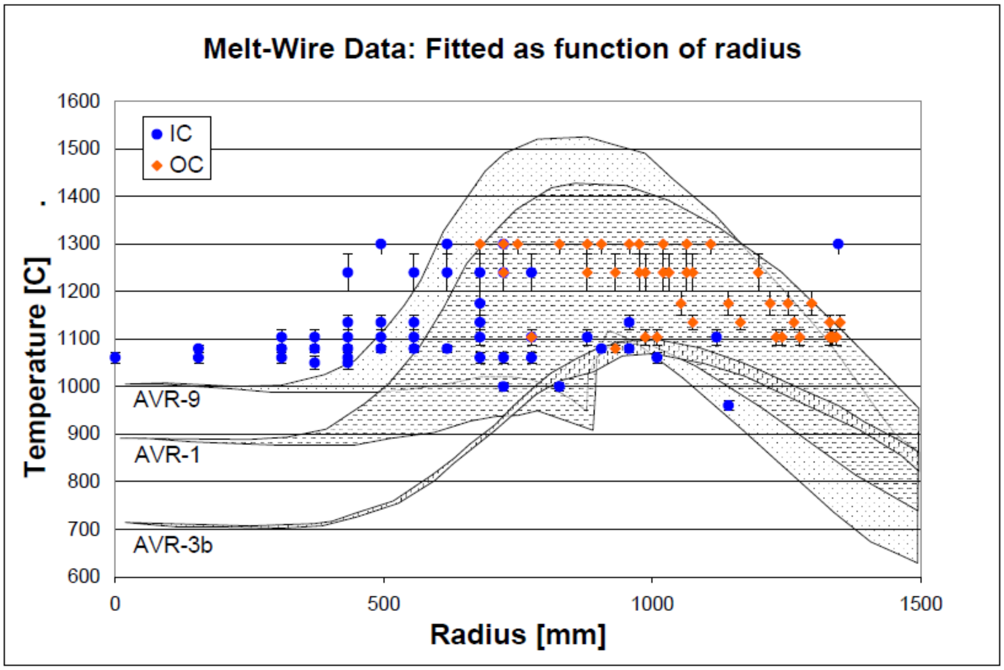
Case	Control-rod Nose Bypass	Annular Flow Bypass %	Wall Channeling Effect	Maximum Core Outlet Coolant Temperature [°C]*
1	No	0	No	1058
2	Yes	0	No	1102
3	Yes	10	No	1194
4	Yes	10	Yes	1209

* Before bypass mixing.

The maximum coolant temperature of 1209°C determined by the PBMR study lies between the AVR mean measurements of 1100°C for the inner core and 1220°C for the outer core. Thus, the PBMR 3D CFD studies quantify the temperature effects due to radial power profile, bypass flows, and channeling effect.

Figure 5-9 compares the measured and predicted maximum fuel temperature distributions, where the radius is backed out from the passage time through the core, using the various assumptions on core bypass flows and power distribution. As shown in the measured and predicted temperature distributions are in good agreement. [19] [21]

Figure 5-9: Radial Distribution of AVR Fuel Temperatures in Comparison to Melt-Wire Test Results



5.1.4.3 AVR Melt-Wire Experiment Assessment

The 1986 Melt-Wire Experiment provides valuable information on the frequency distribution of short-term temperature maxima at the pebble surface when the pebbles are dropped on the top of the core. Based on the analysis of the 1986 experimental data, the following conclusions can be made:

- The least squares fit yields the two Gaussian distributions with temperatures of 1100 ± 66°C for the inner core and 1220 ± 100°C for the outer core and their relative weights correspond to the a-priori known values of melt-wire pebbles inner and outer core.
- The latest evaluation of the AVR melt-wire experiment results provides a clear understanding about the mean values with uncertainties of the AVR maximum fuel temperatures in the inner and outer core regions, and the necessary information for fission product core release 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 AVR core bypass flow.

While AVR melt-wire temperatures were higher than expected, the well-designed experiment has provided reliable information on the maximum fuel temperatures when fuel pebbles going through the inner and outer core. The results from 3D CFD studies confirms that the temperature differences were mainly due to large core bypass flows due to specific AVR design features. In all cases, the maximum fuel temperature never exceeded the design limit of 1600°C.

5.1.5 Hot Spot Issue in PBRs

In pebble bed reactors, the loading and movement of fuel pebbles through the core follow somewhat random processes. These stochastic processes generate concern that "hot spots" may be caused by the clustering of fresh fuel pebbles 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.

5.1.5.1 Hot Spot Simulation in Normal Operation

In a study on the PBMR core peaking power [22], a batch of 20 fresh fuel pebbles were introduced into the region with the highest power where the maximum volume averaged power peak occurs. In practice it is not plausible for 20 fresh fuel pebbles to randomly move into the highest power peak region, which is located about 3 meters from the top of the PBMR core. Furthermore, it would normally take over 50 days of full power operation to move the freshly loaded fuel to the highest power peak region in the PBMR core. By then, typically they would have accumulated over 10 GWd/t burnup. Nevertheless, this scenario represents a simple way to quantify the peaking effect expected from the extreme case as a result of skewed distribution of fresh fuel loading, variations in pebble flow lines, and flow speeds.

A sub-volume containing 1935 fuel pebbles in the highest power peaking region was analyzed, where a batch of fresh fuel was introduced. Since the additional 20 pebbles constitute just over 1% of the region's volume, the fuel pebbles were simply added to the core without removing other spheres. All number densities are volume weighted, which means that the 20 pebbles were actually mixed with the rest of the fuel.

The effect of the introduction of 20 fresh pebbles is very small and no significant effect could be seen on the maximum power (2.97 kW/FS). Even the "1/6th Fresh Fuel" case, which replaces the 1st pass fuel pebbles with fresh fuel, had no effect on the peak fuel power and only a small effect in the volumetric power density. A summary of the results are presented in Table 5-2 [22]. Note that temperature feedback calculations were not performed in all the cases, which would further reduce the effect due to the negative temperature coefficient of the fuel and moderator.

Table 5-2: Variation in Fuel Peaking and Fuel Temperature Data due to Non-Random Loading

Case	Max. Power Density (MW/m^3)	Max. Pwr/Fuel Sphere (kW/FS)	Max. Temp at Peak Power (°C)	Avg. Temp at Peak Power (°C)
Reference	11.26	2.80	1020	842
20 Fresh Fuel	11.25	2.97	N/A	N/A
20 Fresh Fuel with Temperature Feedback	11.22	2.96	1037	853
1/6th Fresh Fuel	11.43	2.97	N/A	N/A

In the table above, the maximum and average fuel temperatures are given in the region where the peak power occurs. The average fuel temperature is calculated for an average pebble with average power and material properties in the mesh. The maximum fuel temperature is defined as the center point temperature of the hottest fuel sphere. Note that the maximum fuel temperature in the reactor normally does not occur at the peak power position but normally at about 6 meters from the top of the PBMR core.

Based on the results of this study, it can be said that addition of 20 fresh fuel pebbles in the region of peak power has little effect on the power density and only increases maximum fuel temperature ~17°C in normal operation.

In order to test the clustering effects, a new model was created with smaller volumes only with fresh fuel and no inter-mixing with other fuel. The results of the cluster model are shown in Table 5-3 [22]. As shown, only small differences can be seen between the two models with the cluster model giving slightly larger power peaking and thus slightly larger maximum fuel temperatures.

Table 5-3: Clustering Effect of 20 Fresh Fuel Pebbles

Case	Max. Power Density (MW/m^3)	Max. Power/FS (kW/FS)	Max. Temp at Peak Power (°C)	Avg. Temp at Peak Power (°C)
Reference	11.32	2.82	1028	857
20 Fresh Fuel	11.41	3.00	n/a	n/a
20 Fresh Fuel with TH feedback	11.40	2.99	1042	864
1/6th Fresh Fuel	11.50	3.00	n/a	n/a

5.1.5.2 Hot Spot Simulation in Loss-of-Flow Cooldown Scenario

A study performed at Idaho National Laboratory (INL) analyzes the consequences of the formation of clusters of different sizes combined with an estimation of the probability for their occurrence in a pebble-bed reactor of an annular core with inner and outer radii of 0.4 m and 1.75 m, respectively, and a height of 9.4 m, producing 300 MW of thermal power [23]. This investigation was performed using the PEBBED code [24], which can model arbitrary pebble circulation schemes with several different pebble types (e.g., fuel pebbles and dummy pebbles).

The PEBBED calculations were performed with a conservative representation of the hot spots by replacing steady-state fuel composition at the location of peak power with composition of different cluster sizes of fresh fuel. The representation of clusters is conservative, because the fresh fuel pebbles will undergo some burnup before they reach the axial core position of the peaks of power and temperature. Table 5-4 [23] shows the PEBBED results for a depressurized loss-of-flow cooldown accident.

Table 5-4: PEBBED Results for DLOFC Peak Temperatures

Case	Number of Pebbles in Cluster	Peak Power (W/cm^3)	Peak DLOFC Temperature (°C)	Probability of Fresh-Fuel Cluster
Nominal	NA	7.38	1580	NA
1	2	10.0	1613	8.26E-3
2	4	10.1	1617	6.83E-5
3	18	10.1	1636	1.80E-19

This study leads to the following results: [23]

- The peak fuel temperature in all cases is unchanged from the nominal unperturbed value in normal operation.
- Clusters of two fresh pebbles are likely to be present in the hot region of the core, but the peak fuel temperature in a DLOFC event only slightly exceeds (by about 13 ºC) the established limiting value in a two-pebble cluster. This figure has significant uncertainties as well due to the assumptions and simplicity of the models employed.
- The cluster of 4 pebbles produces a small excess temperature (17 °C) over the nominal fuel temperature design guideline. However, its probability of occurring at the peak power location is very small.
- A large agglomeration of 18 fresh fuel pebbles is extremely unlikely. Even if it did occur, the peak fuel temperature during a DLOFC event would only be about 60 °C above the nominal value.

Based on the INL calculations, it can be argued that the formation of clusters of fresh fuel causes only minor increases on the hot spot temperature during the course of a DLOFC accident.

5.1.6 Stochastic Core Conclusions

Based on experimental and analytical results, the stochastic nature of fuel pebbles movement in the PBR core is well understood. Due to the simple continuous geometry of the HTR-Module and the chamfered edge at the upper end of the 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. Further analyses of the AVR data using statistical approach show that fuel temperature in the outer core region is higher than the inner core, and pebbles move with propensity more towards the inner region as they flow downward. Although the AVR temperature measurements appeared to be higher than expected, detailed 3D CFD studies show that the temperature differences were mainly due to coolant bypass flows and radial power and temperature distributions, which were not included in the original analysis. Moreover, analytical results also show that packing fraction of pebbles is slightly lower near the reflector walls due to higher flow resistance.

Studies on power peaking effect due to artificially introduced fresh fuel spheres serve as a simple and conservative way to study statistical variations in fuel loading, flow speeds and clustering of more reactive fuel pebbles. The studies by PBMR and INL show that although the maximum power delivered in a fuel pebble may increase due to 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 Significance of Core Compaction Scenarios

Core compaction during seismic events is a problem because of reactivity changes that accompany changes in pebble packing fraction. Nominally, the core operates with a pebble packing fraction of about 0.61, which is less than the theoretically possible value of approximately 0.64. During an event that shakes the reactor, the pebbles could settle closer together and cause a quick increase in reactivity. The increased packing fraction will also impact the core thermal-hydraulic behavior due to a decrease in core porosity and an increase in bypass flow.

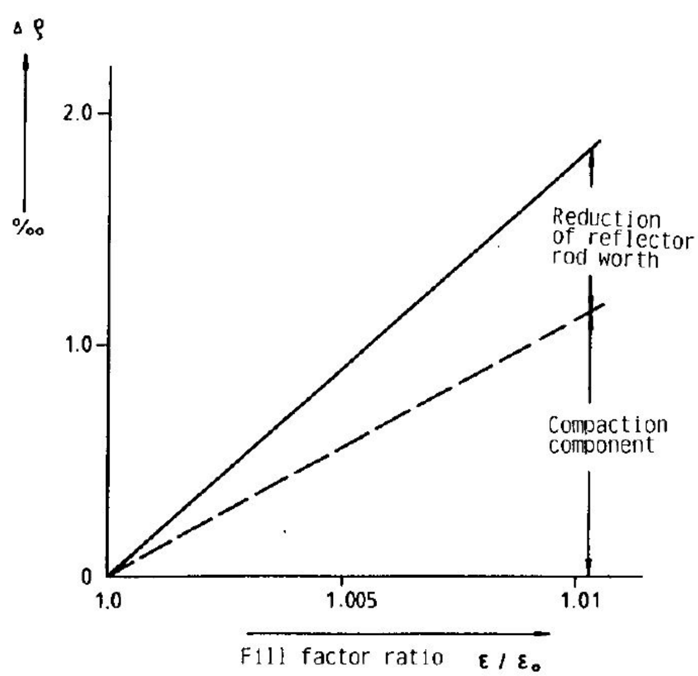
This subsection considered two different levels of core compaction – a partial settling due to a postulated seismic event (likely a design basis accident) and a compaction to the theoretical maximum. Some of the thermal-hydraulic impacts are also discussed.

5.2.1 Impact of Seismic Events on PBR Reactivity

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 the following:

- Reducing the neutron leakage from the pebble bed
- Movement of the pebble bed surface relative to the reflector rods

Figure 5-10: Reactivity Change Due to an Earthquake as a Function of Fill Factor



For a postulated horizontal earthquake acceleration of 0.5 g (g = 9.81 m/s^2), the fill factor increases from 0.61 to 0.614 within approximately 6 seconds at constant excitation. 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 5-10 [3] shows the reactivity increase due to reduced reflector worth and component compaction as the pebble bed fill factor increases.

The initiating criteria for the reactor protection system are as follows:

- Period less than or equal to 20 seconds
- Thermally corrected neutron flux greater than or equal to 120%

Figure 5-11: Relative Power vs. Time Curve for Core Compaction due to Earthquake

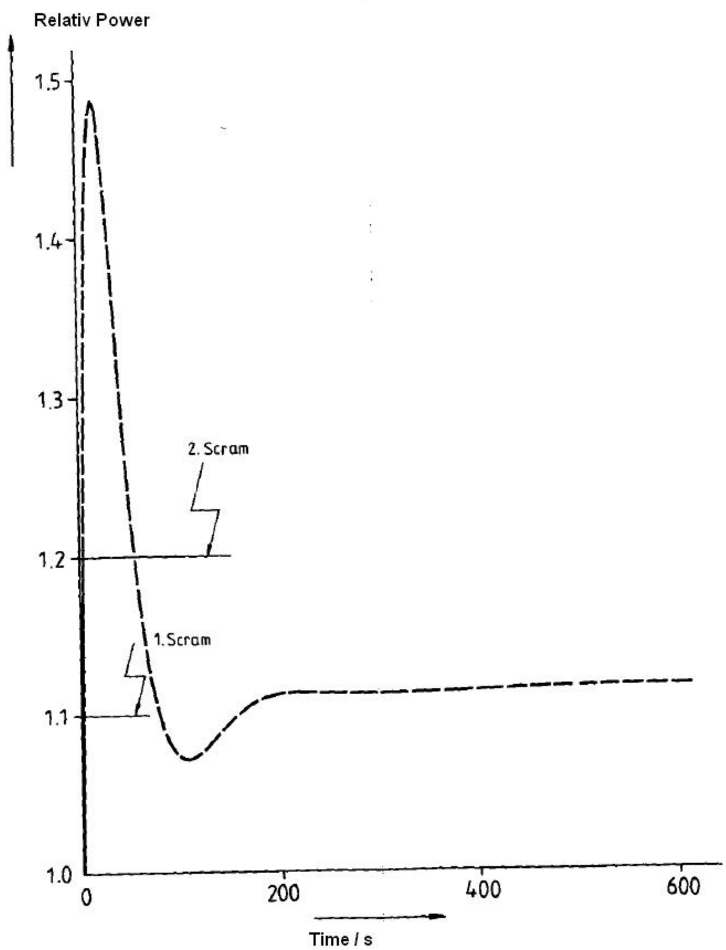


Figure 5-11 [3] shows the reactor power-versus-time curve for this accident. In the associated calculation, actuation of countermeasures by the reactor protection system was not postulated. The points at which the scram initiating criteria are met are marked in the above mentioned figure.

The first of the above initiating criteria is reached after approximately two seconds, the second criteria is reached after approximately four seconds. The figure also shows that, even if countermeasures of the reactor protection system failed to be actuated, the long-term power increase would be extremely low. The hot gas temperature increase would be about 50 K after approximately 500 seconds. The resulting maximum fuel temperature is in a temperature range that has been validated for fuel elements by operating experience.

5.2.2 Reactivity Changes Due to Hypothetical Maximum Core Compaction

The reactivity change due to core compaction to the maximum density was 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 44 cm 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 300 pcm when the core is compacted. These results are summarized in Table 5-5.

Table 5-5: Reactivity Change Due to Core Compaction

Packing fraction	K_{eff}	Standard deviation (pcm)
0.61	1.07467	14
0.64	1.07740	14

5.2.3 Thermal-Hydraulic Changes Due to Core Compaction

Compaction of the pebbles has important consequences for the flow of helium in the core. Because they have less space between the pebbles for gas to flow, regions with high packing fractions exhibit greater hydraulic resistance to the flow. Thus, the helium gas tends to flow around these regions. The reduced coolant flow in these regions, which are already experiencing an increase in the local power density due to the compaction, contributes to the magnitude of the hot spots that result.

In the event that the entire core is compacted, the pressure drop across the core is increased, which results in more helium bypassing the core by flowing through the gaps and spaces in the side reflectors. With less coolant flowing through the active core, the temperature rise across the core increases, resulting in significantly higher temperatures at the bottom of the core. This situation also aggravates issues associated with hot streaks that can appear downstream of the core exit and persist until the coolant exiting the core mixes with itself and the bypass flow.

5.2.4 Core Compaction Conclusions

The following assessments can be stated about the HTR-Module:

- The mechanisms of compaction during seismic events are understood

- Reactivity is expected to increase due to pebble movement and reflector rod worth
- The reactivity transient resulting from a seismic event is understood and manageable
- Thermal-Hydraulic impacts are understood and of lesser consequence than the reactivity impacts

In conclusion, core compaction of the HTR-Module is an understood and manageable phenomenon. No significant consequences are anticipated during normal operations or following a design basis accident.

5.3 Graphite Dust Evaluation

Though graphitic dust is expected to be present to some degree in all HTGR plants, it has been recognized as a potential issue for the PBR due to the higher expected dust quantities. 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 favorable air environment, potential dust ignition/explosion following a depressurization accident is another safety concern for the PBR.

Special design features in the HTR-Module are evaluated and assessed concerning their adequacies addressing the graphite dust issues in the following sections.

5.3.1 Graphite Dust Generation Mechanisms

As a part of the German HTGR program, experimental investigations in the AVR (Arbeitsgemeinschaft Versuchs-Reaktor, 46 MW-thermal) [25] and THTR (Thorium High-Temperature nuclear Reactor, 750 MW-thermal) [26] 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.

In fact, the most interesting and relevant experimental results are those produced in the AVR, because of the real operating conditions and the long period of operation. However, the difficulty here is that direct measurements of total amount of generated graphite dust were impossible and must be estimated.

The AVR reactor was built for the purpose of pebble fuel development tests. A total of 289,789 fuel pebbles from 26 different fuel batches (see Table 5-6 [27]) were fed into the AVR core, out of which 3000 were taken out for general inspection and 600 went through detail examination [27].

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. Another production mechanism for graphite dust in PBR that has been suggested is the reverse Boudouard reaction over metal carbonates (such as Li, Na, K, Cs, Sr, and Ba) [28][29] , due to presence of coolant impurities and fission products.

Besides the graphite dust, small amounts of metallic dust may be present from the construction of the reactor, erosion of circulator blades, and from metallic components during fuel cycling. These metallic dust particles may become activated in the reactor core neutron field. The quantity of metallic dust is expected to be very small in comparison to that of the graphite dust. Up to 15% of dusts collected in the AVR experiments were metallic dust, which can be considered as a portion of the graphite dust when calculating the transport of radionuclides.

The experimental determination of dust generation rate is complicated due to the fact that the friction coefficient between graphite parts is much lower in the free atmosphere than under HTGR conditions. This is because of the adsorption of oxygen at the graphite surface, which can only be avoided by using high purity helium and requiring evacuation of the equipment prior to experiment [30].

Table 5-6: AVR Fuel Loads in the Years 1966 to 1987

Reload Number	Insertion Date (mo/yr)	Fuel Element Type	Number of Fuel Pebbles	CP-Kernel Type	Coating Type	U-235 Enrichment (%)
1	7/66	UCC	30,155	$(Th,U)O_2$	HTI BISO	93
2	10/68	T	7,510	$(Th,U)O_2$	HTI BISO	93
3	4/69	GK	17,770	$(Th,U)O_2$	HTI BISO	93
4	7/70	GK	6,210	$(Th,U)O_2$	HTI BISO	93
5-1	11/70	GK	25,970	$(Th,U)O_2$	HTI BISO	93
5-2	12/71	GO-1	20,825	$(Th,U)O_2$	HTI BISO	92
7	1/73	GO-1	7,840	$(Th,U)O_2$	HTI BISO	93
6-1	10/73	GO-1	11,000	$(Th,U)O_2$	HTI BISO	92
6-2	12/73	GLE-1	2,446	UO_2	LTI BISO	15, 0.7
8-1	5/74	GFB-1	1,440	UO_2,ThO_2	LTI BISO	93
8-2	5/74	GFB-2	1,610	UO_2,ThO_2	LTI TRISO LTI BISO	93
9	9/74	THTR-1	5,145	$(Th,U)C_2$	HTI BISO	93
10	12/74	THTR-2	10,000	$(Th,U)O_2$	HTI BISO	93
11	12/74	THTR-2	5,000	$(Th,U)O_2$	HTI BISO	93
12	3/76	GO-1	11,325	$(Th,U)O_2$	HTI BISO	93
14	11/76	GO-1	9,930	$(Th,U)O_2$	HTI BISO	93
13-1	12/77	GFB-3	6,077	UC_2,ThO_2	LTI TRISO LTI BISO	90
13-3	12/77	GFB-5	5,354	UCO, ThO_2	LTI TRISO	92
13-2	7/80	GFB-4	5,861	UC_2,ThO_2	LTI TRISO LTI BISO	90
15	2/81	GO-2	6,087	$(Th,U)O_2$	LTI TRISO	93
18	7/81	GO-3	11,547	$(Th,U)O_2$	HTI BISO	93
19	7/82	GLE-3	24,615	UO_2	LTI TRISO	10
21	2/84	GLE-4	20,250	UO_2	LTI TRISO	17
20	10/85	GO-2	11,854	$(Th,U)O_2$	LTI TRISO	93
22	9/86	THTR	15,228	$(Th,U)O_2$	HTI BISO	93
21-2	10/87	GLE-4	8,740	UO_2	LTI TRISO	17

The dust produced during normal operation is circulated through the primary circuit by the coolant until it is deposited on a surface. The dust concentration in the AVR during normal operation was very low and amounted to 5 μg/m³(STP), as reported.

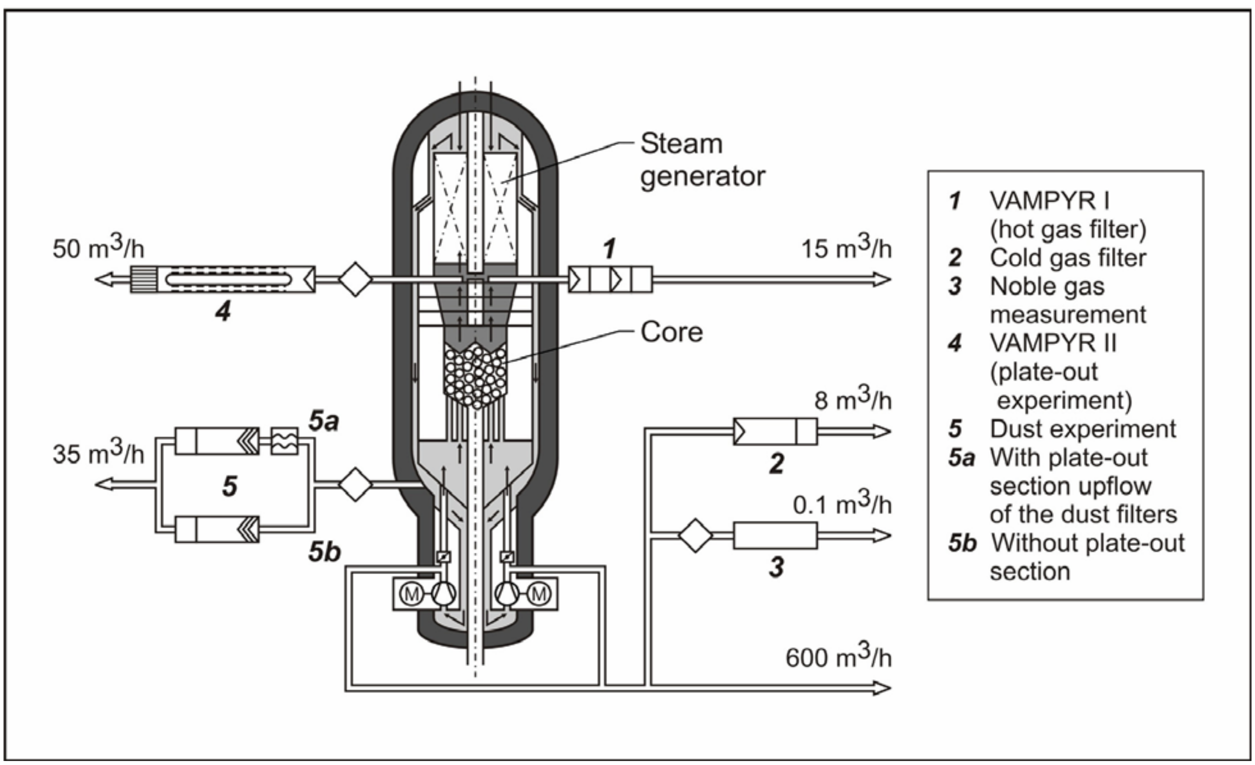
The relevant dust deposition mechanisms are: inertial impaction, diffusion, thermo-phoresis turbulent deposition and gravitational settling. Several successive deposition/re-suspension steps may occur before the dust particle reaches its final location. This effect ensures that all particles except for 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 as described in [31]. If the dust particles have been on the surface for a long time at high temperatures, sintering may also contribute to this hardening effect.

Based on the findings in the DEACO project [32], it has been confirmed that dust is indeed deposited in areas subject to high gas velocities, and that the dust in those regions comprises a significant fraction of the overall amount of dust. The remaining dusts are found in the stagnant helium regions, such as pipe bends in the FHS and the sides of steam generator tubes facing into the flow direction.

5.3.1.1 Graphite 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. There were three major series of dust experiments carried out in the AVR specifically to determine the activities of radionuclides adsorbed on dust. In the first experiment, a set of filters was placed in a cooled line located downstream of the helium circulator. In two subsequent series of experiments, VAMPYR-I and VAMPYR-II, special probes were placed in the main coolant stream at the core exit, to draw off dust and aerosol fission product contamination [33].

Figure 5-12: Schematics of AVR and Its Fission Product Related Experiments



These experiments are summarized as follows:

1. VAMPYR-I hot gas filter: Determined the activity of condensable nuclides in the hot gas via a plate out section and on dust in a subsequent dust filter.

2. Cold gas filter: Determined the total activity (except noble gases, but including dust borne activities also by a separate dust filter) in the cold gas range.

3. Noble gas measurement: Analyzed the total noble gas inventory in the coolant.

4. VAMPYR-II plate out experiment: The plate out section consisted of metallic materials representative for the THTR, and was equipped with dust filters upflow and in certain experiments downflow. VAMPYR-II was operated only for short periods at the end of AVR life.

5. Dust experiment: Examined the dust behavior more in detail using dust filter equipment in the cold gas with several different filters in parallel and series. A plate out section in front of a dust filter for the reduction of the molecular activity did not work properly, and respective evaluations are difficult. The dust experiment operated from 1984-88 only.

In addition to those experiments, data on dust was obtained by wiping down the primary circuit components after decommissioning of the plant. The AVR experiments showed that the dust mass in the AVR primary circuit varied considerably with time.

Because matrix graphite (A3) dust has a very high sorption capability for fission products like Cs and Sr, the following observations have been concluded from the AVR experiments:

- The sorption capability increases with decreasing temperatures.
- Dust present in pebble bed HTGRs is able to absorb all Cs released from the core.
- Overlying dust layers on surfaces may contain the majority of Cs and other fission products.

Table 5-7 summarizes the characterizations of all the AVR dust measurement results.

Table 5-7: Characterizations of All 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, 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: [34]

- The ingress of some 100 m^3 (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.

5.3.1.2 Recent AVR Experiments

Between 2008 and 2009, several dust experiments on the AVR components were performed at FZJ for the PBMR called the DEACO project [32]. The purpose of the DEACO project was to gather as much experimental data as possible on dust and fission products produced in the AVR. This information allows the reduction of uncertainties in nuclear safety calculations for the PBR. In the experiments, two sections of piping taken from the FHS were cut and carefully examined for dust contents and activities. Pipe A was taken from a feed line to the pre-purification of the Purification Unit, coming from the outlets of the circulators of the primary circuit. Pipe B was taken from the two connection lines of a 2 train valve station between the feed line to the pre-purification and the exit line to the containment and other auxiliary systems. Pipe A was operating under the turbulent condition and Pipe B under the laminar condition.

The main results concluded from the first phase of the DEACO experiments are as follows: [32]

- 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; for example, ^{154}Eu is located nearer to the surface whereas ^{60}Co and ^{137}Cs are mainly located in the mechanically removable (by scraping) dust layers.

- The removable dust layers have high iron content, as result of a water ingress accident.

- The dust layers consist mainly of very small particles with an average size smaller than 1 µm.

- The de-convolution of the particle size distribution indicates different kinds of particles.

- The mean value of the surface scraped-off dust mass from Pipe A was 2.36 mg/cm^2. Given the surface areas of the steam generator of 1762 m^2 and of the pebble bed of 1130 m^2, this yields the estimated total mass of dust amounts to 70 kg.

- The mean value of the average thickness of the dust layer is 16 µm.

Furthermore, a separate experimental work performed in a special test apparatus determined that the major source of dust generation in AVR is caused by the fuel handling system, in particular the lift lines. The respective tests showed that graphite dust produced in the pebble bed core plays only a minor part.

5.3.1.3 THTR Dust Experience

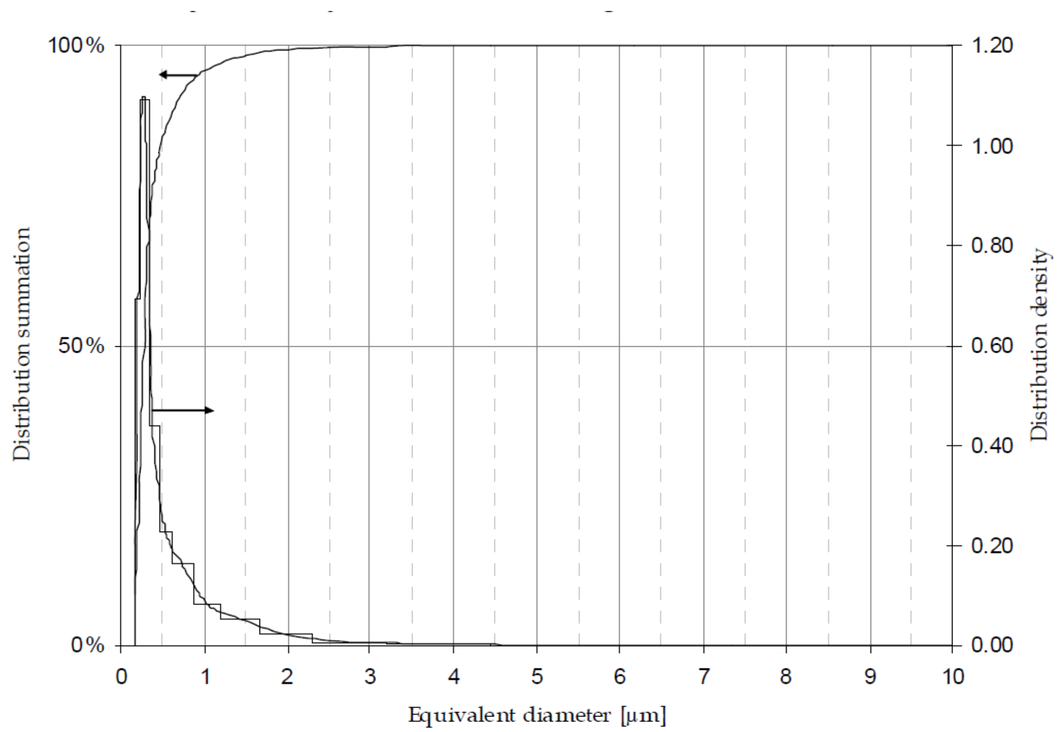
Graphite dust was not a particular issue for THTR, since it had dust filters built into the helium purification system, fuel handling system, and moisture monitors. Nevertheless, in the THTR a graphite dust layer a thickness of about 5 to 10μm was found on all metallic surfaces, after an equivalent full-power operation time of 16 months [35].

5.3.1.4 Graphite Dust Particle Size

Analyses from the AVR and THTR plants showed that the dust occurred primarily in a range between 0.2 μm and 10 μm. Some random particles from wipes were found to be as large as 200 μm. The cumulative distribution of dust mass, as measured in AVR, was found to be 99% of less than 10 μm. Analyses of AVR dust experiment performed at the end of AVR life revealed a mean grain size of 0.76 μm with distribution range of 0.5 to 40 μm.

In the recent measurement performed by the project DEACO IB [32], the de-agglomerated scraped-off dust from Pipe A was analyzed with respect to the size of the particles. The scraped dust from mechanical decontamination was dispersed in ethyl acetate and de-agglomerated in an ultra sonic bath. A de-convolution of the particle size distributions reveals at least three types of particle sizes, which indicates that three types of particles are present in the scraped dust. These include those that originated from the matrix material, the larger carbonaceous dust due to decomposition of oil, and those due to corrosion of graphite. Figure 5-13 shows the number-weighted probability distributions of the dust size independent of the origin of dust.

Figure 5-13: Number Weighted Distribution of Removed De-agglomerated Dust Material



5.3.1.5 Estimated Graphite Dust Generation Rate in Commercial HTGR

In the commercial scale HTGRs, 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 HTGR will increase dramatically compared to AVR. However, all future HTGRs will likely be supplied with a dust filter in the fuel handling system to filter out fine dust particles of size 0.3 to 0.12 μm diameter [36].

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: [37]

1. The total mass of dust generated per full-power year (fpy) is 22.7 kg/fpy.
2. The total mass of dust is 727 kg after 32 full power years.
3. The mass of Helium-borne dust at stationary normal operation is 210 mg.
4. The mass of Helium-borne dust at Depressurization Phase is 45 g.

The representativeness of AVR-data, especially concerning the dust source term, has to be proven by appropriate experiments because a number of avoidable dust inputs were experienced at AVR.

In the pre-conceptual study of a 500 MW-thermal NGNP pebble bed reactor [38], the graphite dust generation rates for the NGNP were calculated for two sources of graphite dust generation: the abrasion of fuel pebbles in the core and in transit through the Fuel Handling and Storage System (FHSS). These sources were calculated independent of the removal mechanisms within the reactor primary circuitry. The estimated values for the generation rates of dust in the NGNP systems are given in Table 5-8.

Table 5-8: 500 MWt NGNP Dust Generation Rates

Graphite dust source Dust generation rate,	kg/fpy
Core (pebble bed, side reflectors and bottom reflector)	28
Fuel Handling and Storage System (dust before filtering)	20
Total	48

5.3.2 Radionuclide Absorption on Graphite Dust

The total amount of activity carried by the dust in an HTGR 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 radio-nuclides 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.

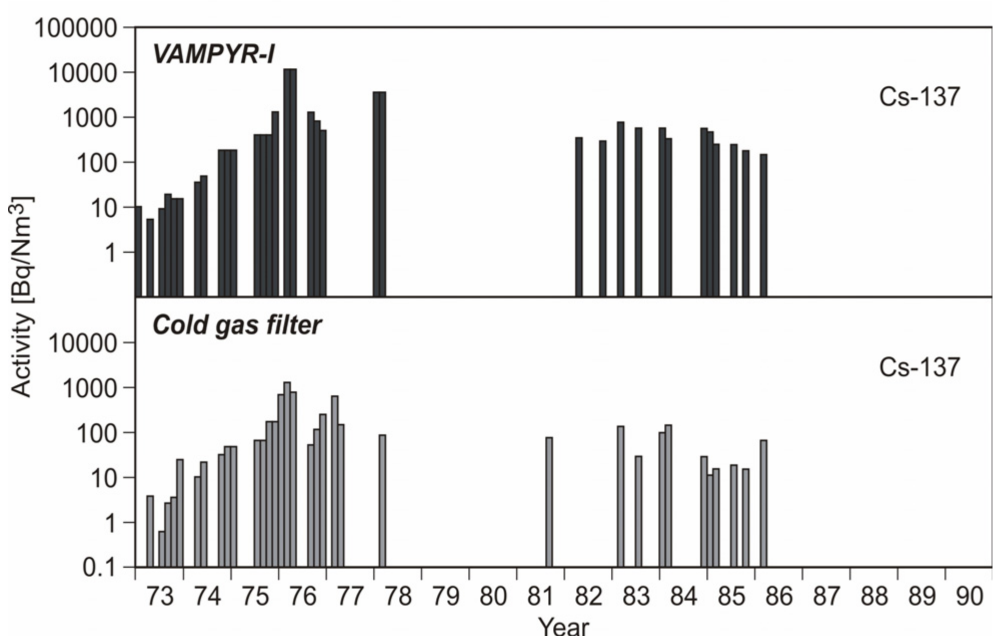
The process of fission product adsorption competes with the plate out on colder metallic structures and also with adsorption on solid graphitic structures such as the side reflectors. Therefore, not all of the free activity of the primary circuit can be considered dust-borne. Moreover, adsorption of coolant-carried fission products can only take place on surfaces that are in contact with circulating helium. Therefore, the activity of dust deposited in regions in which helium is stagnant during normal operation will be limited to the radio-nuclides present in the fuel matrix material due to abrasion of fuel pebbles.

The AVR pebble bed reactor was operated 1967-1988 at coolant outlet temperatures from 750°C up to 950°C. The following conclusions were made concerning fission product releases in the VAMPYR experiments performed in the AVR:

- Dust is deposited primarily on the surfaces or in the wake flow areas of the primary loop under steady-state operating conditions [39].
- The radionuclides found in the primary loop were largely attributable to initial defective fuel particles and uranium contamination in fuel 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 [40].

The activity concentrations of dust depend on the location in the primary circuit and also vary with time. Radionuclide concentrations on dust were extremely low as illustrated by AVR experiments. Figure 5-14 shows the hot and cold gas activities for Cs-137 between the years 1973 and 1986 [33]

Figure 5-14: Comparison of Total Cs-137 Activities in AVR Hot Gas (VAMPYR-1) and Cold Gas



5.3.3 Graphite Dust Remobilization

Deposited graphite dust particles can be remobilized wherever boundary conditions arise in the primary loop, due to sudden changes in flow, leading to an abrupt increase in shear forces acting on the particles.

A number of dust remobilization experiments were performed at AVR in 1986. The reactor was shut down during the experiments. The experimental facilities Cold Gas Filter, Dust Experiment and VAMPYR-I were used for the collection of the dust. In the experiment, the Helium flow transient was initiated by a quick increase of the circulator speed from 1500 rpm by a factor of 2 or 2.5 to 3000 or 4000 rpm in one minute, around 40 rpm per second. The circulator was kept constant at the final speed until the dust accumulation in the dust filters reached a steady-state value. The decrease of the activity collection in time in the dust filter was measured online and inferred as a measure for the dust concentration.

Results of the "Dust Remobilization" circulator transient experiments in AVR are shown in Figure 5-15. As shown the maximum concentration of Helium-born dust for the upper curve (final circulator speed=4000 rpm) was 1050 μg/m³ (STP), and the half-time of dust depletion was 80 minutes. Whereas, the maximum concentration of Helium-born dust for the slower transient (final circulator speed=3000 rpm) was 280 μg/ m³, and the half-time of dust depletion was 31 minutes. The results show a strong dependence of remobilized dust concentration on the final flow velocity. The maximum remobilized dust concentration was at least two orders magnitude greater than the normal dust concentrations. Figure 5-16 shows the total steady-state dust concentration in the coolant, as measured by the dust filter of the cold gas experiment. Moreover, the total mass of remobilized dust was less than 0.1% of the total dust deposit inventory.

Figure 5-15: Dust Concentrations in AVR in Circulator Transient T3

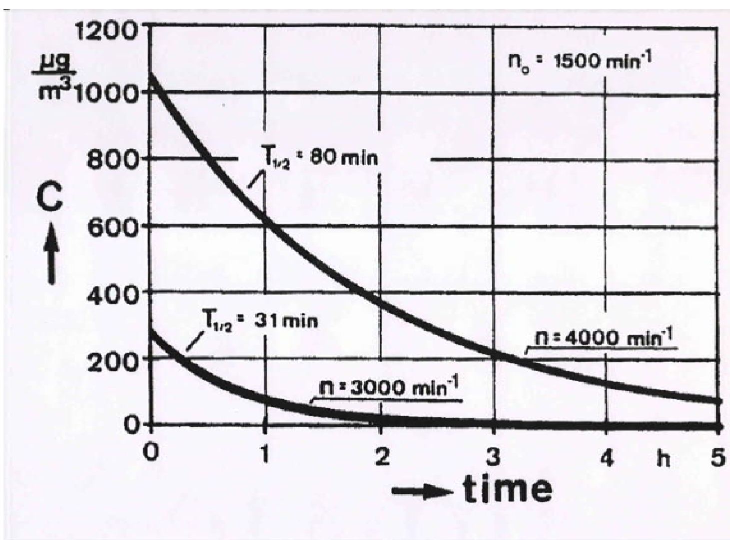
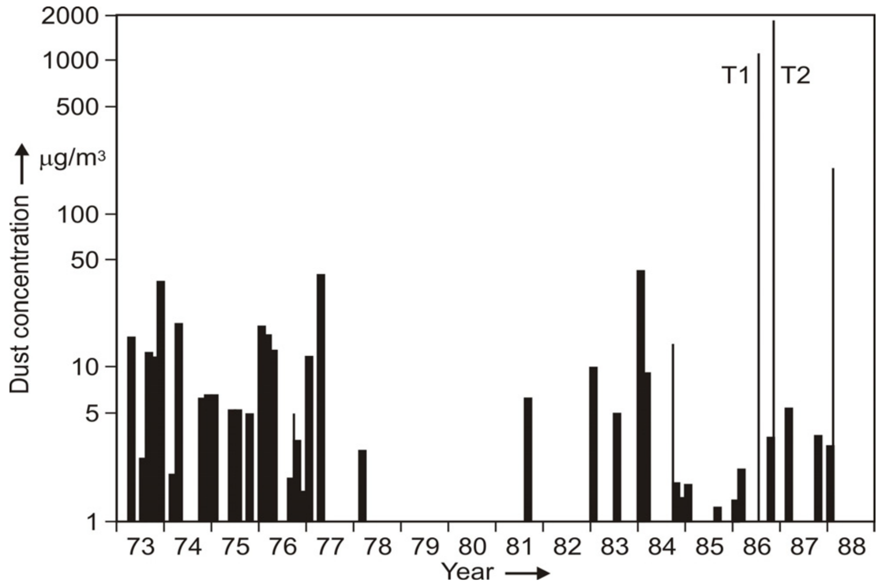
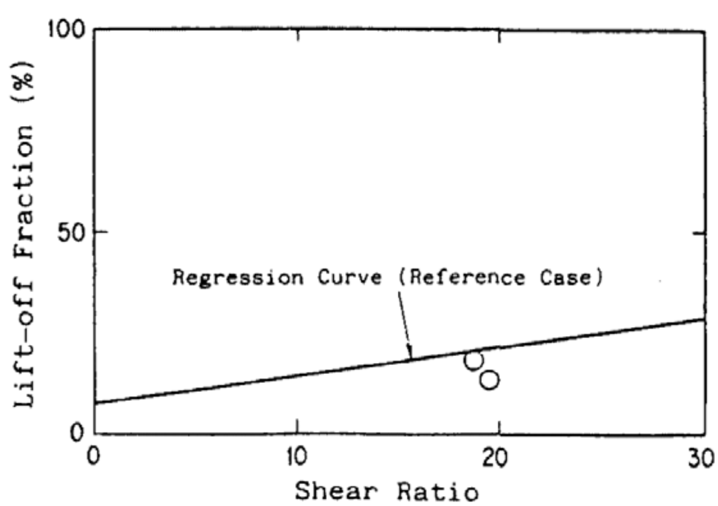


Figure 5-16: Dust Concentrations in AVR Operation



In another experiment investigating dust behavior during depressurization for HTGR [41], the liftoff fractions of graphite dust were obtained. In this experiment, graphite powders of 10 μm size were artificially attached to the test section and blown down under high-speed gas flow to simulate flow condition during the depressurization accident. The liftoff fraction test results were correlated to the shear ratio, which is defined as the ratio of wall shear stress in the blowdown condition to the normal condition. Test results showed that the liftoff fraction increased linearly with the shear ratio, as shown in Figure 5-17. The maximum liftoff fraction obtained was about 25% using linear regression on the data. However, the graphite powders used in this experiment were not representative of the characteristics of dust deposited on the pipe walls inside AVR, which were found to be in the form of strongly bound layers and very hard to remove except by force. Therefore, the shear ratio concept does not correctly describe the dust remobilization behavior during the depressurization phase of the design basis accident in the HTR-Module.

Figure 5-17: Liftoff Fraction of Dust versus Shear Ratio



5.3.4 Impact of Dust Remobilization on HTGR

The limiting design base accident (DBA) for the HTR-Module is the break of the DN65 pressure equalization (PE) line in the primary circuit, which connects the annular space between the RPV and core barrel to the gas purification unit filled with stagnant Helium.

In the Depressurization Phase of the DBA, the calculated maximum gas flow rate out of each break-end is 15 kg/s. Since the flow rate of the helium in the primary circuit is 85 kg/s during normal operation, this break flow rate does not constitute a major change in the flow pattern in the primary circuit. Hence, dust concentration results obtained from the "Dust Remobilization" experiments in AVR are representative of expected dust behavior during the depressurization phase of the DBA in the HTR-Module. Therefore, the following statement can be confirmed and supported by the experimental results:

> *"An enveloping value of 1 kg was assumed for the quantity of dust that is taken up by the primary coolant from dead flow zones during blow-down and is subsequently released from the primary system. For long-lived radio-nuclides, the specific activity of this dust is equivalent to the specific activity of the graphite on the surfaces of the fuel element."*

5.3.5 Combustion of Dust Cloud in 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 in the DBA.

A similar concern for dust explosion hazard also exists in ITER. A series of experiments using a standard method with a 20-liter spherical combustion chamber have been conducted to measure the explosion indices of fine graphite and tungsten dusts and their mixtures. The indices include maximum overpressure, maximum rate of pressure rise, and the minimum explosion concentration limit. The effect of dust particle size on the explosion behavior of graphite dusts was investigated in the range of 4 to 45 µm. The graphite dust particle size is shown to have a profound effect on the explosion characteristics. The finest dust (4 µm) showed the highest maximum overpressure and rate of pressure rise, and the lowest ignition concentration as shown in Figure 5-18 and Figure 5-19 [42].

As shown in Figure 5-18, the ignition threshold obtained by ITER experiments for fine graphite dust of 4 µm is found to occur at a minimum concentration of 70 g/m^3 in air [42]. Since the maximum re-mobilized dust concentration is in the order of 1050 $\mu g/m^3$ following the depressurization phase of DBA, it is more than three orders of magnitude smaller than the ignition threshold. It can be concluded that dust combustion/explosion in HTR-Module would be quite impossible.

Figure 5-18: Minimum Explosive Concentration of Graphite Dust vs. Dust Particle Size

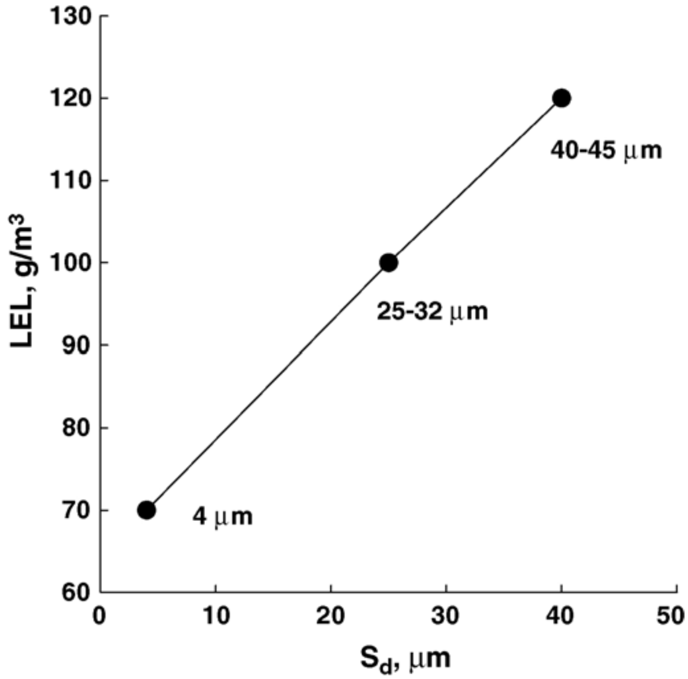
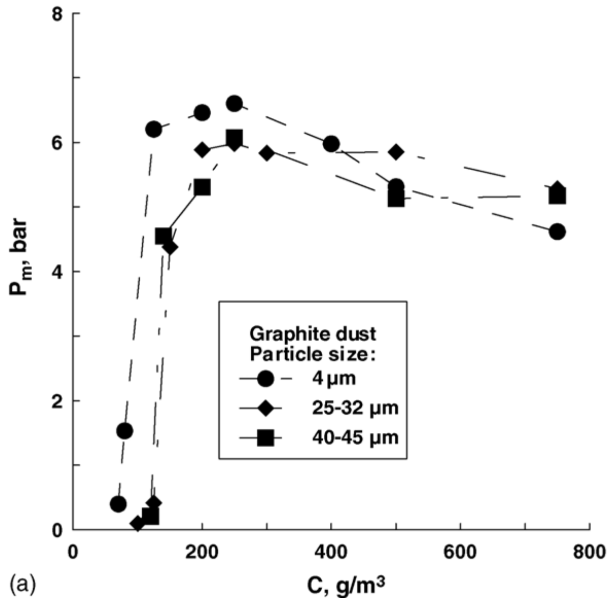
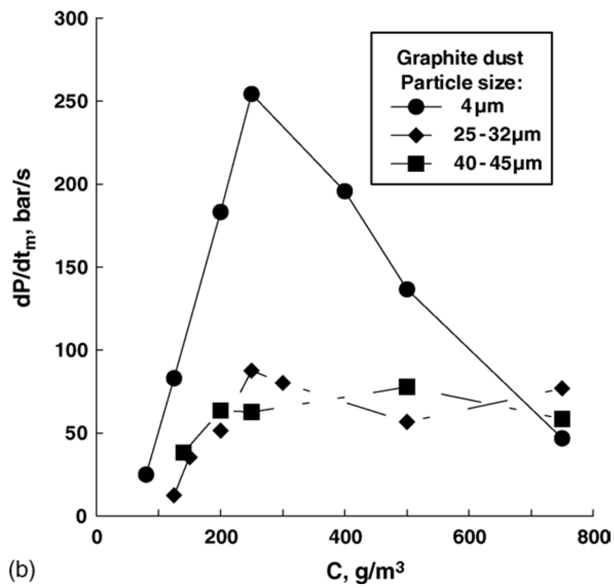


Figure 5-19: Explosion Indices of 4, 25-32, and 40-45 µm Graphite Dust vs. Dust Concentration: (a) Maximum Overpressure, (b) Maximum Rate of Pressure Rise





5.3.6 Dust Cloud Interference with Radiant Heat Transfer to RCCS

The absorption and scattering of thermal radiation due to presence of particulate or dust in the reactor cavity atmosphere plays an important role in the overall energy transfer between the RPV and RCCS, especially under a DBA scenario.

Within a suspension of dust particles, thermal radiation can be either enhanced or attenuated, depending upon the size and concentration of particles, the temperature distribution, and the radiative properties of the matter [43]. When a beam of radiation is incident upon a dust cloud, some of it is transmitted, some absorbed and some scattered. The scattered radiation includes that diffracted, refracted and reflected by the particles. The absorbed radiation is retransmitted at a wavelength corresponding to the temperature of the particle and not the source [44].

The attenuation of thermal radiation within a dilute cloud of pulverized coal and ash has been investigated experimentally and theoretically in the ranges of 1.6 to 30 µm particle size. An empirical expression has been developed for obtaining the absorptivity and emissivity of a coal/ash cloud. According to the empirical nomogram chart shown in Figure 5-20, the dust effect (Dp) is found to be negligible in the ranges of dust particle size and concentration of the HTR-Module [44].

Figure 5-20: Absorption or Emission Coefficient of a Dust-Laden Stream

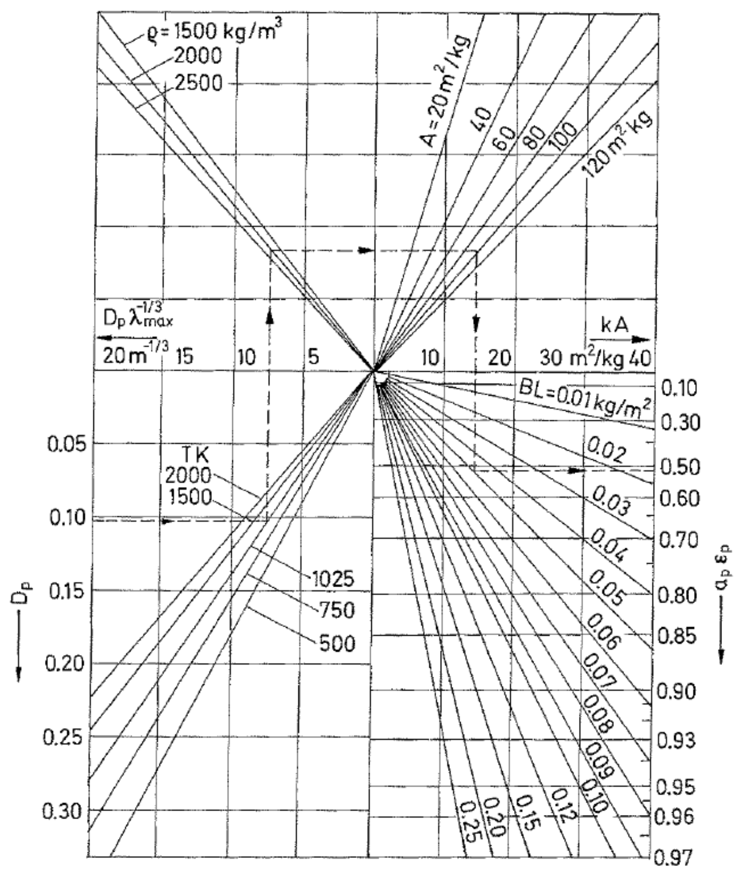


Fig. 13. Nomogram for finding the absorption or emission coefficient of a dust-laden stream (Example: for ash particles $D_p = 0.103$, $T = 1025$ K, $\varrho = 2000$ kg/m³ $A = 100$ m²/kg, $B = 0.005$ kg/m³ and $L = 10$ m then $\varepsilon = 0.51$)

5.3.7 Dust Assessment Conclusions

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

1. The operations of AVR and THTR demonstrated that dusts did not cause problems affecting the reliability of PBR.

2. The AVR provided a valuable experimental data base on dust issues, including steady state conditions and dust remobilization in accident conditions.

3. The dust at inner surfaces of the primary circuit forms a closed layer with strong binding on the surfaces.

4. The estimated amount of remobilized dust in the primary circuit of the HTR-Module in the DBA Depressurization Phase supports an enveloping value of 1 kg in the safety evaluation of the HTR-Module.

5. Dust explosion scenario in the RCCS as a result of the Depressurization Phase in HTR-Module is unrealistic due to the extremely low dust remobilization concentration.

6. 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. Graphite dust is not a release problem; rather it should be regarded as a decontamination task.

5.4 Impact of Broken and Lost Pebbles on PBR Operation and Safety

Broken fuel pebbles in HTGR operation could lead to destruction of coated fuel particles if they get stuck in the core and experience excessive burnup, resulting in the release of actinides that could be adsorbed in dust. During the operation of the THTR, there were far more numbers of scrapped pebbles discharged than originally planned for.

Two additional concerns also arise, if pieces of broken pebbles stuck inside the coolant channels in the core bottom structure could have adversary effects on the reactor operation:

1. Stuck pieces of broken pebbles standing out from the bottom reflector could affect the pebble flow; and

2. Coolant flow through the channels could be reduced.

5.4.1 Broken and Lost Pebbles in AVR

Due to the continuous movement of the pebbles in a PBR core, the possibility of damaged or broken pebbles cannot be excluded. Therefore, in any PBR pebble transportation system components for scrap separation and collection are indispensable.

During the initial loading of AVR, shell-type fuel pebbles were used for the first core. Although less sensitive to surface damages compared to the later pressed-type pebbles, nevertheless the shell-type pebbles showed a higher rate of breakage. This was explained by the formation of a gap between the inner graphite fuel sphere of 40-mm diameter and the outer shell, caused by irradiation densification of graphite in the fuel sphere [45]. However, after introducing the pressed-type fuel pebbles through the pebble transport system, the scrap rate went up. A portion of the pebbles experienced surface damage when falling on sharp-edged ribs of the scrap separator. A re-design of the scrap-separator waltz solved this problem.

The other type of pebble surface damage was due to the jamming of pebbles before the so-called dosing wheel, which acts as a pressure lock in the pebble discharge line between the scrap separator and the pebble distribution and lifting device. A redesign of the dosing wheel and reducing the number of pebbles in the loading column from 10 to 5 reduced the scrap rate and stabilized the rate of damage to approximately 1 to 2 damaged pebbles per 100,000 circulated until the end of AVR operation. The total number of scrap pebbles produced in 21 years of AVR operation is determined to be between 300 to 400 pebbles among the approximately 300,000 fuel pebbles and 70,000 non-fuel pebbles used, for a total circulation of 2.4 million pebbles through the AVR fuel handling system.

The total amount of scrapped pebbles removed from the scrap can is equivalent to some 220 pebbles. The rest of scraped or broken pebbles were "lost" by falling through the coolant channels and retained in the coolant-circulator dome of the RPV.

5.4.1.1 Impact on Pebble Flow Behavior in AVR

In the AVR core, all pebbles in the inner core were loaded by the central pebble feeding line, and all outer core pebbles were loaded by the 4 circumferential pebble feeding lines. During the later years of AVR operation, pebble flow data on the minimum and the average passage time through both inner and outer core could be obtained with the employment of the precise pebble measurement.

All the data on pebble movement revealed a very stable and predictable pebble flow behavior throughout the AVR operating history. This observation was confirmed by the melt-wire pebble experiments conducted, which began in September 1986. The data obtained from the melt-wire pebble experiment indicated that the average passage time for the inner-core pebbles was slightly faster by several percent.

Therefore, there were no indications that broken pebbles in the AVR core had in any way impacted the pebble flow behavior in the core.

5.4.1.2 Impact on Coolant Flow in AVR

Concerns on pieces of broken pebble stuck inside the coolant channels of the core bottom structure, however, were real in the AVR. Inspections of the AVR core during decommissioning revealed that the core bottom was cracked and some broken pebbles had stuck in the cracks [46].

In 1999, after the AVR had been defueled, the bottom reflector was video-inspected. Although no standing-out scrap pieces could be detected, there were indeed pebble scraps stuck inside the coolant channels as seen on the AVR video pictures. The video pictures showed there were 12 such stuck inner spheres visible. The "scrap" pebbles were of a uniform kind, consisted exclusively of the 40-mm inner spheres of the shell-type pebbles loaded in the initial core, and were pressed into the 35-mm wide slits on the bottom reflector. Furthermore, the video pictures also revealed damages in the bottom reflector structure. There were various cracks in the upper graphite-block layer and blocks had shifted sideways. Consequently, some coolant channels were narrowed or even closed, while others became wider so that the intact inner spheres of the broken pebbles could be trapped.

However, the impact of stuck broken pebbles and damages on the bottom structural had negligible effects on coolant flow through the core. This was confirmed by an always stable relation between the coolant circulator speed and measured core pressure drop.

5.4.1.3 Impact on Fission Product Release in AVR

Because the stuck fuel scraps in the AVR were the inner spheres of the shell-type pebbles belonging to the first core, they were irradiated for the whole AVR lifetime. However, the burnup the stuck fuel spheres achieved is probably not extremely high because of the low neutron flux in the bottom reflector. Their BISO-coated particles did not show any important fission product release, since the coolant activity went down to low values after 1981 and remained low for the rest of the AVR operation.

Yet, there were extreme burnups possible in the AVR. For example, the TRISO-coated fissile particles of the GFB (feed/breed) type pebbles achieved a burnup of some 77 % FIMA. The 17,290 pebbles of this type showed excellent fission-product retention, despite the high AVR temperatures.

5.4.2 Impact of Damaged Pebbles in THTR

During the operation of THTR, an unexpectedly high number of damaged fuel pebbles had occurred. The surface-damaged pebbles were sorted out by the helical scrap separator during discharge of fuel pebbles from the reactor core. A total of 10 scrap casks have been filled with approximately 17,000 damaged pebbles. The percentage of damaged pebbles among the total pebbles withdrawn was about 1.5% in the beginning of the refueling operation, and then continuously decreased to 0.6% [47].

It has been assumed that the damage was mainly caused by frequent and deep insertion of the in-core control rods during the THTR operation. This assumption has been confirmed by agreement with experimental data; the damage in most cases only affected the graphite shell, which did not contain fuel particles. Hence, the coated fuel particles inside the damaged fuel pebbles remained intact, and the retention of the fission products was secured. Furthermore, there was no evidence that the pebble flow behavior in the core and the insertion of the in-core rods were impaired by the damaged pebbles.

Extensive downtimes were needed to exchange the scrap casks containing damaged pebbles, since the process required complete depressurization of the pre-stressed concrete reactor vessel. Therefore, the issue of damaged fuel pebbles is more of an economic concern rather than safety concern.

5.4.3 Addressing Broken and Lost Pebble Problem in PBR

From the operating experiences of AVR and THTR, a number of design changes that will keep the scrap pebble production rate during operation to a very low number have been made in the HTR-Module,.

5.4.3.1 Improvements on Graphite Support Structure

One of the key features of the high-temperature reactor is the graphite structure enclosing the reactor core. This structure accommodates the forces imposed from the weight and movement of pebbles in the core. It acts as neutron reflector, radiation shield, pebble bed container, and coolant flow channel guidance.

A modification has been made in the graphite structure of the HTR-500, where the 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 [48].

A second modification made is that the bottom graphite reflector structure is 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.

For design optimization and verification purpose, seismic tests must be performed with the core structure models, in order to determine the forces upon the reflector side walls and the core bottom structure due to dead load, weight of pebble bed, and pressure drop.

5.4.3.2 Improvements on Control Rod Design

Experiences in AVR and THTR have shown that the in-core control rods and control housing protruding into the core were primarily responsible for most of damages on fuel pebbles and slowdown of normal pebble flows. Therefore, in both proposed HTR-Module and HTR-500 designs, control rods have been moved to entirely inside the side graphite reflectors, thus eliminating the possibility of pebble damages caused by control rod movement and preserved normal pebble flows.

5.4.3.3 Improvements on Fuel Handling System

A number of design modifications have already been made in the fuel handling systems of AVR and THTR to minimize damages to the fuel pebbles. These modifications include eliminating sharp edges in the fuel discharge line, jam-free dosing wheel configuration, re-routing of the coolant gas bypass flow through the fuel discharge pipe which impeded the rolling out of pebbles, and increasing the scrap cask capacity.

5.4.3.4 Improvements on Pebble Fuel

Many improvements have been made in the manufacturing process of the fuel pebbles to increase quality. The pressed-type fuel pebbles with higher annealing temperature of 2000ºC have improved corrosion resistance and increased strength, thus better retention of fission products [49].

5.4.4 Broken and Lost Pebble Assessment Conclusions

The experiences learned from the operation of AVR and THTR, have shown that the scrap pebble production rate can be kept to a very low level and that it does not represent a safety problem in any way for the operation of PBR.

The experiences have also shown that pebble damage occurring in the PBR can be minimized with optimal design of the graphite bottom support structure and fuel handling system, and keeping the control rods inside the graphite reflector. Furthermore, improvements in pressed-type fuel pebbles (with higher annealing temperature of 2000ºC) have drastically limited the broken pebbles issue in HTR-Module. Therefore, 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

There are specific features of the PBR design that might appear as weaknesses in terms of proliferation resistance when compared with other types of reactors presently operated in the world:

- The on-line refueling, possible when the reactor is operated at full power, extends the period when the fuel can be handled and possibly diverted to the full lifetime of the reactor. This is contrary to the reactors with batch reloading—for which fuel handling is possible only when the reactor is in cold shut-down, with the vessel open for refueling—which is more compatible with the discontinuous nature of IAEA controls.

  Moreover, in reactors with batch reloading, undeclared opening of the reactor vessel can be easily detected through satellite surveillance. Conversely, in the PBR design, refueling and possible related fuel diversion would remain unnoticed by such remote control methods. Even direct visual control of the fuel from the arrival of the fresh fuel on the reactor site to the storage of spent fuel in storage bins, which is a usual practice for safeguards purpose, is difficult since the fuel remains enclosed in inaccessible zones.

- The fuel elements (spherical pebbles 6 cm in diameter) are small, which makes their hiding for diversion much easier than that of the long fuel elements of LWRs. They are numerous, which makes errors in their counting possible.

- Though still being LEU, the fuel of PBRs has a higher enrichment than the fuel of LWRs, which makes its use for further enrichment attractive. Furthermore, if irradiated fuel diversion is possible, it is no more difficult to divert the fuel after only one pass in the reactor than after its full irradiation. The plutonium of this fuel would have a high fissile content, whereas the isotopic content of the plutonium in fuel that could be diverted after one cycle in an LWR would already be significantly degraded. The only way to obtain weapon grade plutonium in an LWR would be to anticipate the opening of the vessel, which would not remain unnoticed as explained above.

Modular PBRs appear more adapted than large LWRs to the needs of many developing countries with limited nuclear infrastructure because of their smaller size and because of their alleged robustness and forgiveness. The question, then, is if these reactors have a large deployment in such countries, could they be a significant vector for nuclear proliferation?

The enhancement of proliferation resistance is one of the major goals fixed to the development of Generation IV nuclear systems [50]. Could PBRs contribute to the fulfillment of this goal or would their deployment contribute on the contrary to deteriorate the proliferation risks in the world?

5.5.1 Issues to Examine for Assessing PBR Proliferation Resistance

The sale of nuclear facilities is authorized only to countries committed to submit these facilities to IAEA safeguards provisions.

Due to its specific features, could it be easier to divert, unnoticed by IAEA safeguards controls, fissile material from PBR than from other types of reactors, whether the diversion is performed in the frame of a national undeclared program for production of fissile materials or in the frame of clandestine terrorist activities unknown to national authorities?

Would the fissile material diverted from a PBR be more or less attractive for a direct or indirect weapon use than the fissile material diverted from another type of reactor? A direct use could be the manufacturing of a nuclear explosive with the fissile material or of a “dirty bomb” (that is, a conventional explosive surrounding irradiated nuclear fuel that could be exploded for spreading radio-contaminants). An indirect use could be the feeding of an undeclared plutonium production reactor or enrichment plant.

Political changes can lead the authorities of a country that previously accepted IAEA safeguards, to reject them. Would PBR technology allow such a "rogue state" to obtain weapon grade fissile material more easily than with other nuclear technologies?

In assessing proliferation resistance of PBR, pointing out its proliferation potential in relation to an ideal norm of absolute proliferation resistance, for which there is no actual reference, should be avoided:

> *"Over the past 15 years, about 100 states reported 1340 incidents dealing with nuclear materials, 303 of these involving unauthorized acquisition (i.e. by theft), which correspond to about 21 attempts/year. Considering that exactly in two thirds of the cases the materials reported lost or stolen either from a specific location or during transportation, were not recovered, this means the success rate of these malicious acts is of 14 successes/year."* [51]

Though the statistics mentioned above also include the theft of radiological sources, not only of fissile materials, they show that there is still a lot of progress to be made in terms of proliferation resistance. Therefore, the proliferation resistance of PBR is to be assessed in comparison with existing nuclear systems: will it be at least equivalent to existing reactors in terms of proliferation resistance, or even will it bring some progress?

5.5.2 Safeguards Provisions for PBR

The objective of the safeguards provisions is to have a timely detection of possible diversion of a significant quantity of fissile materials. A significant quantity (SQ) of fissile materials is the quantity required to fabricate a simple atom bomb, accounting for the losses in chemical conversion and machining. One SQ of LEU is 75 kg of U-235 in the form of LEU. One SQ of plutonium is 8 kg [52]. To obtain one SQ of U-235 from a PBR, 136,000 fresh pebbles are required, which represents more than one third of the HTR-Module core. To obtain one SQ of plutonium, 52,000 spent fuel pebbles or 104,000 core pebbles are required. Nevertheless, special target pebbles with high loading of depleted uranium could reduce this number to about 20,000 pebbles [52][53]. By comparison an SQ of plutonium can be obtained from only 2 PWR fuel assemblies and an SQ of U-235 from 4 fuel assemblies.

The low quantity of fissile materials in a pebble and the large number of pebbles required for obtaining an SQ is an argument put forward in favor of the natural proliferation resistance of PBR. But without rigorous safeguards measures able to detect any significant diversion of fissile material, this argument cannot be credible

In most of the present industrial reactors, fissile material safeguards are based on a combination of redundant controls:

- Item accountancy:
    - Visual inspection: confirmation of the serial number of fuel assemblies,
    - Random verification of fissile content (U-235 content of fresh fuel, checking of Cerenkov glow of assemblies in the reactor when the vessel is open, to detect possible undeclared substitution of fuel assemblies with dummy assemblies, measurement of plutonium content in spent fuel through non-destructive assay (NDA) or, if needed, through detailed isotopic analysis of samples in hot cells).
- Containment and surveillance (C/S).

In the HTR-Module core, there are 360,000 pebbles, while the core of other types of reactors with batch reloading is only composed of a few hundred fuel elements. About 5,000 pebbles are extracted from the core every day and then, after burn-up measurement, either re-circulated to the core if they did not reach the target burn-up yet, or sent to a spent fuel storage bin. Each pebble is circulated on average 15 times in the core before being stored as spent fuel. About 300 fully burnt pebbles are sent every day to the spent fuel storage bin and an equivalent number of fresh pebbles are introduced in the core to replace them. The pebbles are not identified individually by serial numbers.

The very large number of fuel elements circulated, and the absence of individual identification of these elements, prevent item identification pebble-by-pebble. The IAEA considers LWRs an "item facility" because the fuel is discrete and can be verified as an item. PBR is more like a "bulk material facility" in which the fuel is in bulk form when stored in fresh fuel drums, in the core, and in spent fuel storage bins.

Therefore, though C/S is still possible, due to the fact that item accountancy is not practicable, PBR safeguards cannot be based on the approach used presently for LWRs. A novel approach must be developed.

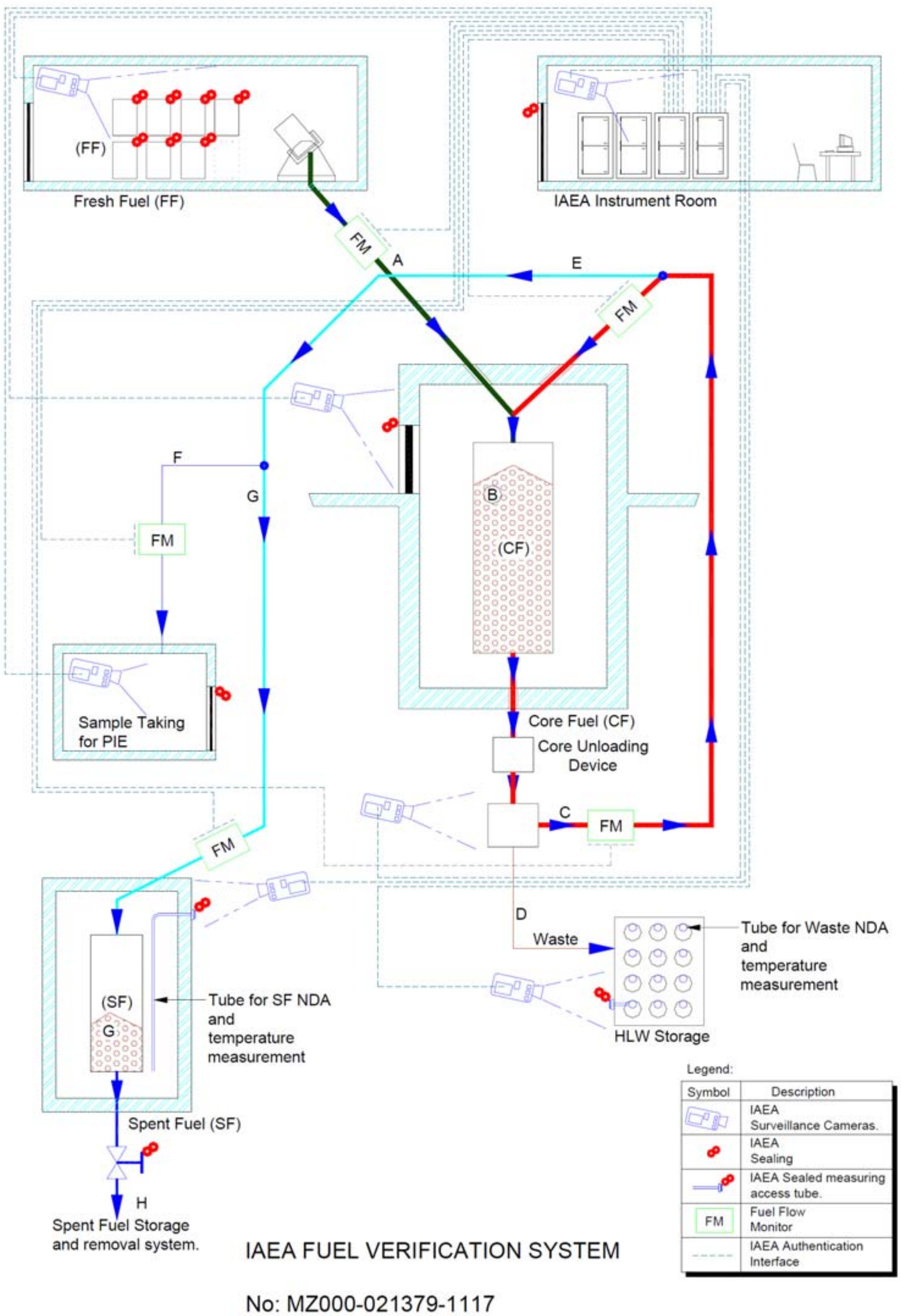
As there are bulk facilities in the fuel cycle (enrichment and conversion facilities, reprocessing plants, etc.) and as some reactors like CANDU reactors are also close to bulk facilities, the IAEA already has significant experience in safeguarding such facilities. But for the time being, the IAEA safeguards approach for PBR, as well as the specific criteria applicable in this approach, is still in development. This makes it difficult to make an assessment of the robustness of safeguards provisions for PBR very conclusive.

Nevertheless approaches have been suggested for PBR safeguards. PBMR has proposed a method [54] based on item accountancy limited to the account of pebble flow on circulation paths of pebbles and on dual C/S provisions based on the redundancy of video surveillance and seals (Figure 5-21). Though applying the continuity of knowledge principle, this method is not immune from faults that can lead to a loss of the continuity of knowledge:

- "The failure of video-surveillance without loss of flow monitors does not affect the conclusions drawn from the number balance, but the failure of flow monitors without failure of the surveillance does not necessarily provide adequate assurance of the lack of a diversion, because identification and categorization of the spent pebbles cannot occur. It would be difficult to differentiate and account for targets, graphite balls, and spent fuel pebbles sent out to the wrong path." [52]

- Counting hundreds of thousands of pebbles continuously during several decades will most likely lead one day or the other to some unbalance in pebble flows and therefore to a non-zero "Material Uncounted For (MUF)" value. This happened in HTR-10 in spite of the relatively small core size (28,000 pebbles) [52]. Of course the safeguards for this type of reactor, as for other types of nuclear systems, have to deal with uncertainties. However, the risk here is that cumulative uncertainties lead to a situation in which IAEA would no longer be in a position to assure that an SQ of fissile material has not been diverted without extensive controls that could possibly require reactor shutdown.

Figure 5-21: Schematic of Proposed Measures for the Application of Safeguards at PBMR

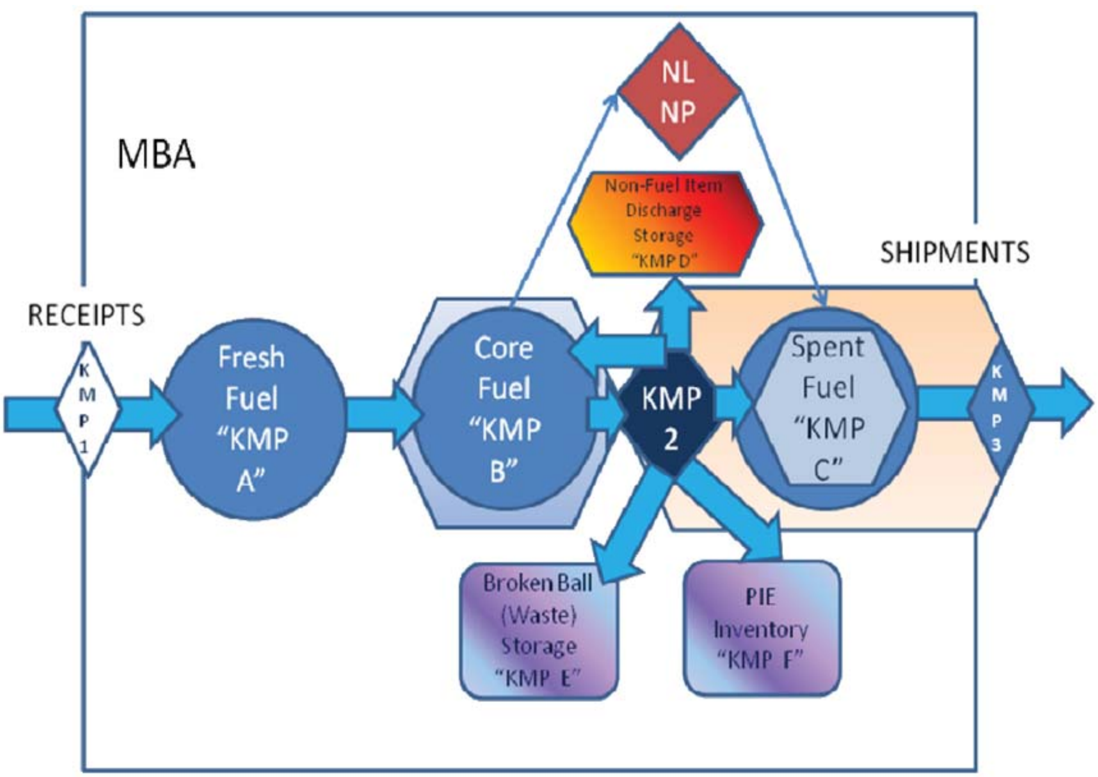


In order to recover the assurance of absence of diversion that would result from a loss of continuity of knowledge that cannot be excluded with the method envisaged by PBMR, independent bulk measurements of quantities and radiation attributes in different areas of the facility have been considered in addition to pebble flow measurements [52]. For that purpose the whole facility is divided into several mass balance areas (MBA), the fresh fuel storage area, the reactor core area, the spent fuel storage, the broken pebble storage, the non fuel item (graphite pebble) storage and the hot cell for post-irradiation examination (PIE), and the following additional controls are performed:

- In each storage area, the bulk mass/volume and fissile content are measured:
  - In the fresh fuel area, the fresh fuel drums ID numbers are verified, the drums are weighed and their U-235 and U content is randomly checked using an Active Well Coincident Neutron counter.
  - In spent fuel bins, the spent-fuel fill-height and the gross radiation attributes are measured. For that purpose, instrument tubes should be installed in these bins. In addition, sampling spent fuel pebbles for verification of their burn-up and plutonium content in the reactor hot cell is possible.
- No direct bulk measurement can be made in the reactor core area, except possibly pebble sampling for hot cell examination, but, based on the power history, the evolution of the core fissile inventories can be estimated using a burn-up code.

Therefore, a hybrid approach, using both the pebble flow accounting and C/S measures envisaged by PBMR and bulk measurements in the different MBAs is proposed [52][55], as illustrated in Figure 5-22, where the "Key Measurement Points (KMP)" are identified. Though not eliminating any MUF, such redundancy in the establishment of mass balances should allow maintaining them at a low level for the whole reactor lifetime.

Figure 5-22: The Mass Balance Areas (MBA) and Key Measurement Points (KMP) of a PBR



In addition to the controls described above, the IAEA should be able to have access to the design information required for identifying potential diversion paths right from the design phase and for tracing their possible evolution until the extended shutdown phase. It should also be able to verify them directly in the facility, in accordance with the safeguards agreements signed by countries operating nuclear facilities.

5.5.3 Assessment of Proposed Safeguards Provisions for PBR

As noted above, safeguards approaches are still in development and are not integrated yet in the detailed design of any PBR plant. Moreover IAEA has not formally endorsed specific criteria for PBR safeguards. Therefore only a provisional and qualitative assessment of proposed (but not approved) safeguards can be made.

One should not give too much importance to the arguments about inherent proliferation resistance of PBR. These arguments should be put into perspective with some inherent weaknesses of PBR:

- The very large number of fissile elements required for obtaining one SQ makes it difficult to divert a significant fissile mass unobtrusively. But the limited size of pebbles makes it easier to hide them than LWR fuel assemblies. The number of fuel flow paths to be controlled is also larger in a PBR than in an LWR.

- Though the presence of strong coatings on fuel particles adds a difficulty for retrieving the fissile content of the fuel, it is not impossible: this is done currently by mechanical processes for recovering fabrication scraps, admittedly not with irradiated fuel, and new processes in development (for instance disintegration of TRISO particles with pulsed currents) might make it even easier [56]. Moreover the difficulty can be bypassed if target pebbles highly loaded with natural or depleted uranium without coating are used [57].

- The very high burn-up that can be achieved in a PBR makes the irradiated fuel improper for weapon use. This is not necessarily true. If made with very high burn-up HTGR fuel, a nuclear explosive will be less efficient than with weapon grade plutonium, but it will work [51][58]. The IAEA discards plutonium from safeguards controls only if it contains at least 80% Pu-238, which is far from being the case with the plutonium produced in an HTGR. Moreover if target pebbles or normal pebbles are diverted after one pass in the core, their plutonium has a high fissile content, much better than the plutonium that can be retrieved from batch reloading reactors (including block type HTGR).

Therefore the PBR is not inherently proliferation resistant. But its proliferation resistance can be reinforced by design. For instance:

- The accessible flow paths should be minimized. From that point of view, the HTR-Module design, with a single pebble bed mostly inside the reactor and with the only part outside in a zone with very difficult access (high radiation zone), is rather optimized.

- Unavoidable critical possible diversion paths should be made accessible to controllers (and therefore sufficiently shielded) or at least adequate coverage should be provided for video surveillance. But these paths should be enclosed in areas that can be safely sealed between inspections.

- Storage bins should be designed in an appropriate manner, to make that the bulk measurements proposed for controlling their content be possible (pre-installed instrument tube with one of sufficient size to accommodate a small neutron generator).

- Locations for measuring the radiation attribute of the fuel should be provided in such a way that differentiation between dummy fuel elements or target pebbles and normal fuel elements on the other hand should be possible.

- Utilities required by IAEA instrumentation should be planned by the designer.
- Storage areas should be protected against intrusion.

It is clear from these examples that safeguards requirements should be integrated in the design for the safeguards of future PBRs to be really effective.

Now, if the proliferation resistance of a PBR can be reinforced by design, it can only be assured through effective safeguards controls. How effective can the controls envisioned for PBR be, especially with the hybrid approach proposed above? If it appears that there is no obstacle in principle for achieving a level of confidence in the detection of possible fissile material diversion at least equivalent to the one obtained today for LWRs, it is presently difficult to go beyond this qualitative assessment because detailed information and analyses are still missing:

- A full assessment of the hybrid approach would require this approach to go beyond a basic conceptual scheme and to be defined in details and to implemented in an actual reactor detailed design in order to test its robustness with scenarios meant to challenge it. Moreover the specific criteria to be used for safeguarding a PBR should also be defined precisely for such assessment to be possible.
- It is clear that the uncertainty of the safeguards controls should be reduced by adopting the hybrid approach in comparison to the approach proposed by PBMR. But is it sufficient to maintain for sure the unavoidable MUF below the SQ level? To answer this question, the uncertainty of all the measurements should be known, which seems not to be the case for the time being. In particular the uncertainty in counting a large number of pebbles should be evaluated and the impact of broken pebbles on this uncertainty evaluated.
- The safeguards of PBR will be more complex than for LWRs, if only due to the larger number of fuel flow paths. On top of this, the redundancy of the hybrid approach makes it even more complex. Will this complexity become a safeguards nightmare, with a multiplicity of error sources, or can it be handled smoothly, within the scope of usual IAEA practices. Only IAEA can answer this question, on the basis of its inspectors' experience.
- Some measurements required for the hybrid approach still require development: for instance, for controlling the spent fuel for safeguards purpose, it is not sufficient to measure its burn-up, its plutonium content should also be evaluated. This requires the use of methods (for instance X-Ray Fluorescence) that have still to be developed and qualified in PBR conditions. More generally the IAEA is developing a new generation of safeguards tools (new measurement techniques, electronic seals…). The adaptation of these tools to PBR safeguards needs and the evaluation of the benefit that can be drawn from their application are still to be addressed.
- The burn-up codes, which are needed to verify the mass balance of the core MBA, are not qualified up to the very high burn-up reached by PBR fuel. Therefore the uncertainty in the calculation of the core mass balance is not very well known.

From this list of open issues, it appears that gaps in the determination of uncertainties in some measurements and calculations are critical for obtaining a more precise assessment. But without waiting for additional data on uncertainties, sensitivity studies could already give precious indications. Unfortunately they are rare.

At least for an important issue, the detection of fertile targets inserted for a single pass in the core, a sensitivity study is available [59]. In this study the effect of introducing, in addition to normal fuel, target pebbles loaded with natural uranium particles is assessed: target pebbles are loaded mixed to the fuel pebbles in order to have a homogenous distribution of these pebbles in the core. Calculations have been performed for two densities of target pebbles: one that represents an additional uranium mass of 0.1% (in natural uranium) to the total enriched uranium mass of the fuel core and one that represents an additional mass of 0.4%. The added natural uranium increase the number of neutron captures in the core, which have to be compensated by increasing the number of normal fuel pebbles added every day (and consequently sending the same increased number of spent fuel pebbles in a spent fuel bin) in order to keep the same reactivity margin than in the initial core in order to be able to continue its operation in a sustainable way. The result of the calculation is that the daily number of fresh fuel pebbles must be increased of 21% in the first case, and of 95% in the second case. Moreover, the quantity of plutonium produced in the target pebbles is very low and it would take in the first case 92 years to get an SQ and in the second case 23 years. This shows that

- Even with a very low plutonium production, the introduction of target pebbles would be easily detected, would it only be by the control of fresh fuel consumption (there will also be abnormal signals for the core flux and burn-up measurements, but their sensitivity has not been determined).

- If a proliferator freed from safeguards controls wants to get more quickly an SQ by inserting targets pebbles in a PBR core, he will load these pebbles with more natural or depleted uranium (likely not in the form of coated particles to avoid the difficulty of having to break the particle coatings for separating the kernels). For keeping the reactor critical in such conditions he will need to increase drastically the fuel supply of the reactor (more than twice) and will certainly be faced sooner or later with a fuel shortage. This is inherent to the very small reactivity reserve kept in a pebble bed – this at least can be qualified as an inherent proliferation resistant feature of PBR.

In conclusion, the introduction of fertile material targets in a PBR is not a very attractive way to divert fissile materials: it is easily detectable and very costly in fuel consumption.

5.5.4 Attractiveness of PBR Fissile Materials for Proliferation Purposes

Whatever the difficulty of diverting fissile materials is, we will suppose here that a proliferator has been successful in diverting significant quantities of fissile material from a PBR, which covers the case of a rogue state that would bypass the obstacles of safeguards. Would this fissile material be attractive for this proliferator?

As already stated, the difficulty of retrieval of fissile materials from a PBR is not a very strong argument for discouraging possible proliferators.

Certainly fresh pebbles represent a more attractive source of U-235 than LWR fuel due to their higher enrichment. A way to limit the risk of proliferation with PBR enriched uranium is to maintain in the country operating a PBR a stock of pebbles with a U-235 lower than one SQ, which represents more than a year of fresh fuel supply for the HTR-Module. As long as the country is accepting IAEA safeguards, the controls limit the risk, and once the country rejects the safeguards, the fuel supply is stopped and county is left with a quantity of U-235 less than one SQ.

Concerning plutonium, the most favorable isotopic composition for proliferation purpose would be obtained from targets inserted in the fuel for a single pass, or from normal fuel withdrawn from the core after only one pass. Figures are given for in reference [60] for normal PBMR fuel: after the first pass in the reactor, the burn-up is 10 GWd/tHM and the plutonium contains 81% of fissile isotopes, which is not sufficient to qualify it as weapon grade plutonium, but which is certainly sufficient for making a nuclear explosive. With the HTR-Module, which has shorter passes, the isotopic quality of the plutonium at the end of the first pass would even be better. Now the quantity of plutonium obtained after one pass is 0.047 g per pebble. For making one SQ of plutonium 170,000 pebbles would be required. The withdrawal of a part of the fuel after a single pass will be possible only if additional fuel is supplied for keeping the reactivity balance. Even if sufficient quantities of plutonium could be generated in a few years if sufficient additional quantities of fuel are supplied, this will likely be prevented by fresh fuel supply shortage, if the supplier puts limitations in the fuel supply flow rate.

The main concern should therefore be with the spent fuel—if it is stored in the country where the reactor is located, without return to the supplier country—because a large number of pebbles are accumulated in a few years in the storage bins, and because its diversion has no impact on the reactor operability and on the fresh fuel availability. But is the fissile content of the spent fuel valuable for a proliferator?

The uranium that can be retrieved from spent fuel is still with a small enrichment, higher than the enrichment of the uranium retrieved from an LWR, but with a similar reactivity penalty due to the presence of U-234 and U-236 [61]. Nevertheless, if used to fuel reactors with highly thermalized neutron spectrum like CANDU reactors or RBMK, the capture penalty due to the epithermal resonances of U-234 and U-236 would be minimized. Therefore the uranium retrieved from PBR spent fuel could be used as a fuel for a plutonium production reactor.

The isotopic quality of the plutonium that can be retrieved from a PBR is not very good (~ 60%) [58][60], but in the same range as LWR plutonium. As mentioned above, it can still be used for making a nuclear explosive, of course with poor efficiency, and would not be exempt from IAEA safeguards.

In conclusion, the quality of fissile material that can be retrieved from PBR spent fuel is attractive for proliferation purpose, but no more than with other types of reactors. Moreover, due to the difficulty of dismantling the fuel before reprocessing and to the large quantity of pebbles to be handled to recover significant quantities of fissile materials, this is not the optimal way to produce nuclear explosives.

There is another risk that is rarely addressed: the risk for a few pebbles to be diverted from a PBR and used in a “dirty bomb.” As pebbles are small, they can be more easily hidden and then enclosed with conventional explosive in a bomb than other types of fuel. As the focus of safeguards is the detection of fissile material SQ, and as in a single pebble there is an extremely small fraction of an SQ, the risk of such a diversion should not be neglected. The consequences of the explosion in terms of radio-contamination should also be assessed. A single reference addressing this question could be found [57]. The author bases his assessment on an analogy with tags used in conventional explosives, which are small particles inserted by manufacturers in explosives. These particles keep their integrity in an explosion and are used in investigations to identify the origin of the explosive. As coated particles remain intact at the very high temperature during a PBR accident, the author claims that they will survive unbroken to an explosion. In terms of temperature this may be true (though the temperature transient will be much faster than in accident conditions), but will the particle survive the mechanical stresses due to the shock wave and possible impact of particles on hard surfaces? It should not be forgotten that they are ways to break particles.

One could object that the diversion of a few pebbles should be more a matter of security, to be addressed by the national authorities of the country operating a PBR than a matter of safeguards under IAEA responsibility. But what about a government deciding to remain openly under IAEA safeguards, continuing operation of its facility, while secretly sheltering the dissemination of “dirty bombs”? This risk should therefore be addressed more in depth, in terms of detection and in terms of consequences.

5.5.5 Conclusions

Though PBR concept has some inherent features that make proliferation of fissile materials from this type of reactor more difficult, it would be imprudent to rely too much on these features to consider the overall proliferation resistance of PBR. There are also some inherent proliferation issues in this reactor concept to be faced, for instance the multiplicity of fuel circulation paths. The proliferation resistance of a PBR should be built from the integration of safeguards concerns in the details of the design in order to minimize the possible diversion paths and to facilitate safeguards inspections and measurements and from the development of a robust safeguards approach adapted to this type of reactor.

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, though more attention should be paid to the residual risk of “dirty bomb.” But more analyses involving in particular the IAEA and more R&D and qualification work should be performed to confirm this assessment.

The PBR design is no 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, apart from the small size of each fuel element, which should be a concern for the “dirty bomb” risk. Moreover, using a PBR is not the optimal solution for production of fissile materials for nuclear explosives.

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 indeed be drastically increased to maintain the reactor critical. Therefore, in order to be able to force a diverted reactor to stop long before accumulating a significant quantity of plutonium, the fuel supplier, who should not be local (reactors can be exported, but not fuel manufacturing), should have a policy of just-in-time fuel delivery to avoid building up large fresh fuel stocks in the country where the reactor is operated.

Now does the PBR concept address the Generation IV goal of improved proliferation resistance? This cannot be assured at the level of the concept, but similar conclusions could likely be drawn from the analysis of other Generation IV concepts, with specific concerns for each of the concepts. It is mainly by taking into account proliferation concerns in the details of the design and by developing advanced safeguards measures that proliferation resistance can progress.

5.6 Shutdown Margin

This section provides an assessment of the pebble bed reactor (PBR) available shutdown margin throughout its operating phases with specific examples from the HTR-Module design. This assessment is based on the information provided in [62]. PBRs normally operate with a small amount of excess reactivity due to their on-line refueling nature. Nevertheless, the reactor must possess sufficient shutdown-down margin in the main control rod and reserved shutdown elements to render the reactor core sub-critical during all modes of reactor operations from full power, partial power, at temperature zero power and cold-shutdown at ambient conditions.

Typically, a PBR has a primary reactivity control system (i.e., control rods) that are used to maneuver the reactor through its operating power range and shutdown the reactor if unsafe conditions are detected. In addition, PBRs have a secondary shut-down system as a backup or diverse mode of reactor shutdown.

In our assessment key elements of a typical PBR core reactivity contributions and their functional dependence on temperature, burnup, and core location are discussed. The assessment draws on a typical reactivity balance and shutdown margin available during equilibrium core power operation and first core initial cold start-up phase.

The geometry of the PBR core, the dynamic characteristics of the fuel, the graphite moderator, and the reflector neutronic behavior cast the question of the operability and controllability. In other words – Does the PBR core design have sufficient control and shutdown margin?

The individual components of the reactivity coefficient temperature dependencies are then examined. Finally, the long term impact of xenon distribution and stability are discussed and the power density distribution and oscillations and reactor maneuverability are examined.

This assessment is based on the HTR-Module reactor design and safety analysis [3].

5.6.1 HTR-Module Reactivity Balance and Shutdown Margin

Core reactivity and its relative value is a global characteristic of the reactor. Core reactivity is equivalent to the rate of change of neutron flux in the core. In general, its value and behavior in time and space depend on the reactor fuel, its physical location, distribution and temperature in the core. The neutron physics of the moderator and the position of the reactor shutdown elements also play an important role in core reactivity determination. On the other hand and unlike light water reactors, the PBR primary coolant helium; due to its neutronic characteristics does not have any significant effect on core reactivity balance.

The relationship between reactivity and the effective multiplication factor keff is expressed in the following correlation:

$$\rho = (1 - 1/k_{eff}) \times 100\%$$

Excess reactivity is at its highest in a cold and xenon-free core. To compensate for excess reactivity, to ensure an adequate positive shutdown margin, and to control changes in reactivity that occur during operation, two features are provided for the equilibrium core and three for the first core (running-in phase), which each have a high neutron absorption capacity:

- 6 reflector rods
- 18 small ball shutdown units and
- Absorber elements (only in first core and running-in phase)

These features serve both to assure sufficient reactor sub-criticality in the cold, zero-power state and to compensate for the following reactivity effects:

- Reactivity differential between "zero-load cold" and "full-load hot" operating states
- Steady-state and transient xenon/samarium poisoning
- Reactivity changes as a consequence of accidents (e.g., increase in moderation due to in-leakage of moderating substances such as water or compaction of the core as a consequence of induced vibration)

- Burnup-induced and fuel loading-induced fluctuations in reactivity, especially during the initial operating period of the reactor

For example the HTR-Module equilibrium core reaches its highest excess reactivity of approximately 7.8% in the cold condition about 3 to 5 days after downtime. This excess reactivity is composed of the following fractions:

- 1.2% excess reactivity at 100% full-load operation for a long period as the reactivity reserve for the 100% - 50% - 100% of thermal power load cycle

- 3.0 % reactivity gain due to cooling the core from operating temperature to 50°C

- 3.6% maximum reactivity gain added by decay of fuel isotopes (including Np 239) and decay of fission products (including Xenon 135) at 50°C within about 3 to 5 days

In the event of long-term shutdown during which core temperature falls to the above level over a long period, this reactivity must be bound by the shutdown systems.

Since the temperature coefficient is especially negative during the first months at full load, shutdown margins for the first core are smaller than for the equilibrium core. Under accident conditions, on loss of function of one shutdown system and assuming most adverse uncertainties in the reactivity balance calculations, it would consequently be possible for the reactor to return to criticality at temperatures below 100°C.

This condition does not present a problem from the safety stand point, since the core can only generate an output corresponding to the small power loss (less than 100 kw) but it represent a serious issue for plant maintenance activities at low core temperatures. Nevertheless by controlling the criticality and core loading with absorber spheres, it is ensured that this situation will not occur during maintenance operation. If the overall core temperature decreases too much, the core temperature will be raised. This can be done by coolant flow heat-up or by a short increase in power operation.

If the plant operation has to be interrupted for a longer period, the reactor can always be cooled down to the subcritical cold state using partial core off loading or introduction of absorber spheres.

The load cycling range for the first core is smaller during the first months at full load than that for the equilibrium core and the reflector rods are predominantly used in this phase to compensate for the of reactivity coefficients.

Absorber elements

In the first core and during the "running-in phase," absorber elements are used to compensate for the excess reactivity necessary for the fuel loading scheme. This simplifies fuel management and reactivity adjustment during the first period of operation. Hafnium and boron are used for reactivity binding. Approximately 2% reactivity is linked to the absorber elements. It is also possible to use special absorbers that undergo a predetermined loss of reactivity binding capacity during one cycle through the core (e.g., burnable poison absorber balls).

Control and shutdown systems

The six reflector rods and the eighteen small ball shutdown units ensure safe shutdown of the reactor from any operating condition. The shutdown worth of both systems is designed to assure shutdown even on failure of the highest-worth element to drop.

Under normal operating conditions, the reflector rods compensate for changes in reactivity such as those which come about due to load changes, or they induce rapid reactivity changes (e.g., reactor scram).

The small ball shutdown system is used for long-term shutdown and to compensate for slow reactivity changes (compensation for lack of xenon poisoning on start-up or, in rare cases on load changes, e.g. long term operation at part load of 50% or lower).

Reactivity balance of the equilibrium core

The shutdown systems as a whole have the following safety-related functions:

- To transfer the reactor to a subcritical, zero-power state
- To keep the reactor subcritical in the long term under the worst combination of conditions to be taken into consideration

The hot shutdown system (6 reflector rods) is designed to be able by itself to transfer the reactor to the zero-power and subcritical condition under all operating conditions and anticipated operational occurrences sufficiently quickly and to keep it subcritical for a sufficient length of time. This can be accomplished assuming a failure of the highest-worth control component (i.e., highest worth rod fails to trip).

The cold shutdown system (18 small ball shutdown units) is designed to be able by itself to render the reactor subcritical under all normal operating conditions and anticipated operational occurrences that require no rapid changes in reactivity, and to keep it subcritical in the long term.

Both systems as a whole are capable of rendering the reactor subcritical from all normal operating conditions, all anticipated operational occurrences and postulated accident conditions and of keeping the reactor subcritical in the long term, even if single failure occurs (i.e., failure of the highest-worth component)

Table 5-9 and Table 5-10 show the reactivity balances for the equilibrium core at hot and cold shutdown conditions.

Table 5-9: Reactivity Balance of Hot Shutdown System for Equilibrium Core

Necessary	Reactivity (%)
Control reserve for 100% - 50% - 100% incl. control margin	1.2
Reactivity compensation at part-load (xenon, temperature)	0.4
Maximum accident reactivity	0.5
Part-load/zero-load reactivity (hot)	0.1
Total	2.2
Available	**Reactivity (%)**
6 reflector rods	3.2
5 reflector rods (one reflector rod not considered)	2.6

Assuming worst-case calculation uncertainties, a shutdown margin of 0.4% is available if no credit is taken for one reflector rod.

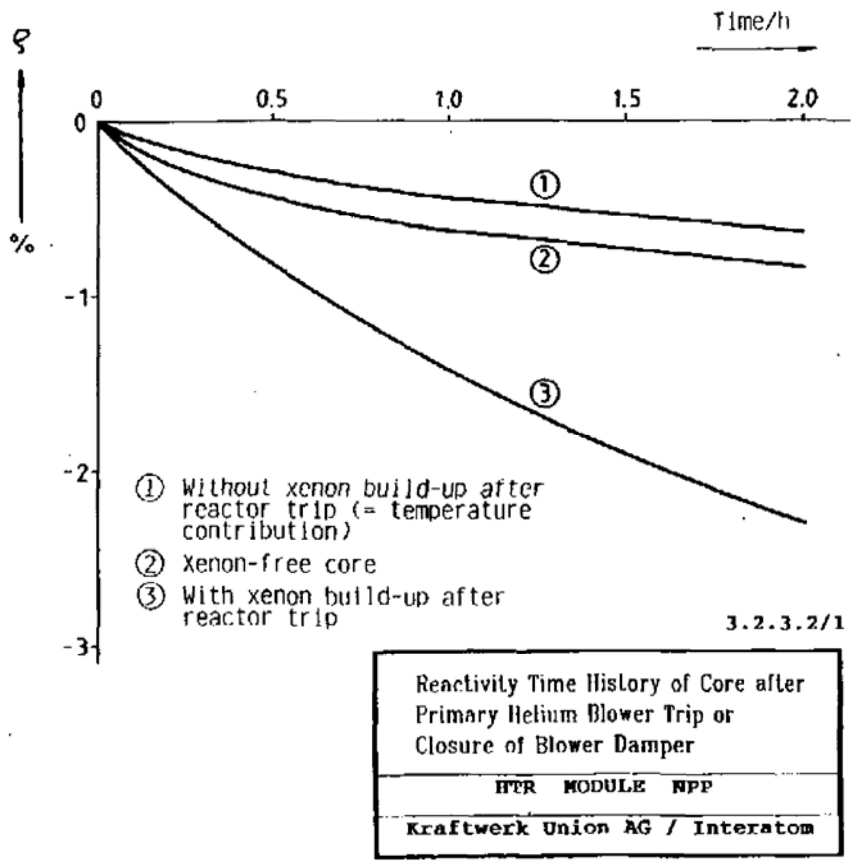
Table 5-10: Reactivity Balance of Cold Shutdown System for Equilibrium Core

Necessary	**Reactivity (%)**	
	18 Units	17 Units + 6 Rods
Control reserve for 100% – 50% – 100% including control margin	1.2	1.2
Reduction of core temperature to 50°C	3.0	3.0
Isotope decay at 50°C	3.6	3.6
Maximum accident-induced reactivity	---	0.5
Subcriticality	0.3	0.3
Total	8.1	8.6
Available	**Reactivity (%)**	
	10.6	11.0

Assuming worst-case calculation uncertainties, shutdown margins of 2.5% and 2.4% are available.

A further possibility for shutting down the reactor is to stop the flow of primary coolant; this results in a slight rise in mean core temperature and hence causes the reactor to go subcritical owing to the negative temperature coefficient of reactivity. Whenever the reactor is scrammed; the primary coolant flow is interrupted both by tripping the primary gas circulator and by closing the circulator damper. On closure of the circulator damper, reactor power would revert to residual heat level within approximately 100 seconds, even taking into consideration failure of the shutdown systems. On primary gas circulator trip alone, the time it takes for the reactor to shut itself down increases to approximately 200 seconds because of circulator coast-down. Figure 5-23 shows the reactivity time history of the equilibrium core for the first case.

Figure 5-23: Reactivity Time History of Core after Primary Helium Circulator Trip or Closure of Circulator Damper



In the first core, which unlike the equilibrium core does not contain fission products or plutonium isotopes, each fuel element contains 7 grams of uranium, as well. However, the uranium 235 enrichment in the first core is about 4.2 % (HTR-Module UO_2 start up fuel enrichment). The first core comprises about 50 % fuel elements and about 50 % moderator elements. Some absorber elements serve to adjust the absolute reactivity level, reactivity binding and fuel loading scheme during the first period of operation in the running-in phase. Approximately -2% reactivity is provided by the absorber elements.

The shutdown margins for the first core are smaller than for the equilibrium core. However, the necessary shutdown margins can be provided by proper adjustment of power level, hot gas outlet temperature or load cycle operation during the running-in phase, if necessary.

5.6.2 Reactivity Coefficients in HTR-Module

The reactivity coefficients describe the dependence of reactivity on changes in the parameters that influence the neutron balance during reactor operation. They are defined by the quotients delta (ρ)/delta (x), where x is the corresponding variable parameter.

Essentially, the following parameters are variable:

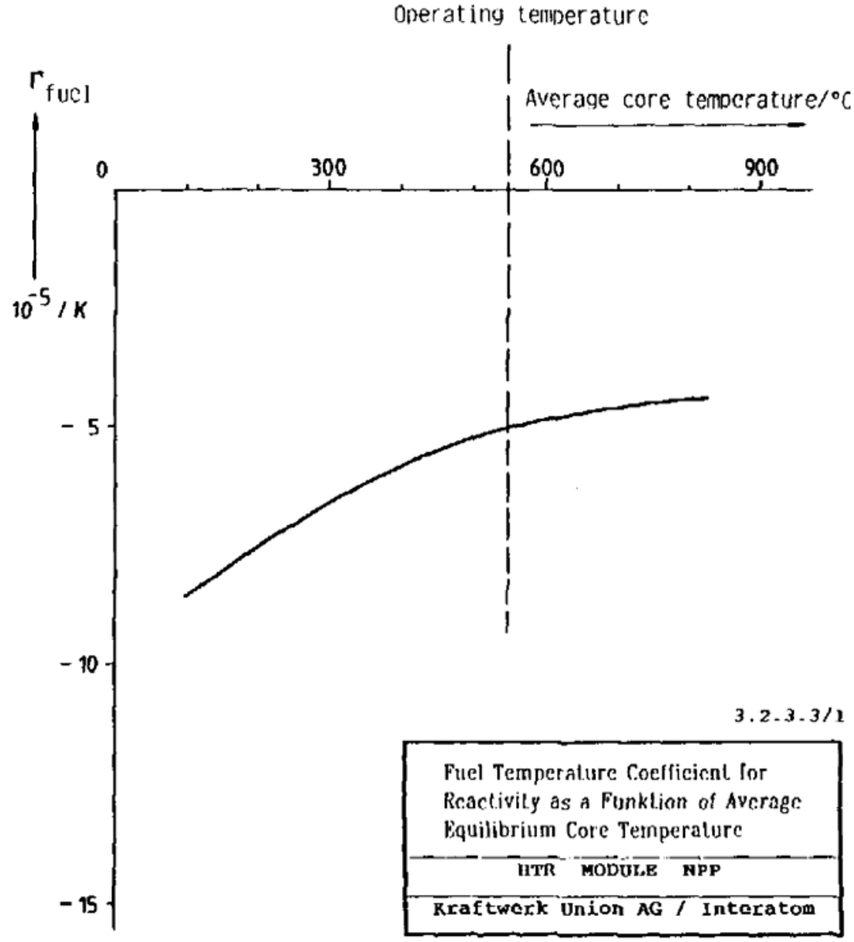
- Fuel temperature
- Moderator temperature
- Reflector temperature

The reactivity coefficients change according to operating condition (first core to equilibrium core, xenon inventory, temperature level in core) and the position of the shutdown elements. The total temperature coefficient for reactivity (total of fuel and moderator coefficients) is always negative in the present system design (total power, operational excess reactivity, heavy metal charge of fuel elements, etc).

Fuel temperature coefficient

Figure 5-24 shows the fuel temperature coefficient has a prompt effect and is always negative. As the temperature rises, however, its absolute value drops. A change in fuel temperature influences the neutron balance of the reactor in the thermal and epithermal energy ranges. In the epithermal energy range, the Doppler Effect causes the greatest change in the neutron balance. It results from the change in the epithermal absorption cross section of uranium 238 and other isotopes such as Pu 240 caused by the increase in the resonance width as fuel temperature rises.

Figure 5-24: Fuel Temperature Coefficient for Reactivity as a Function of Average Equilibrium Core Temperature



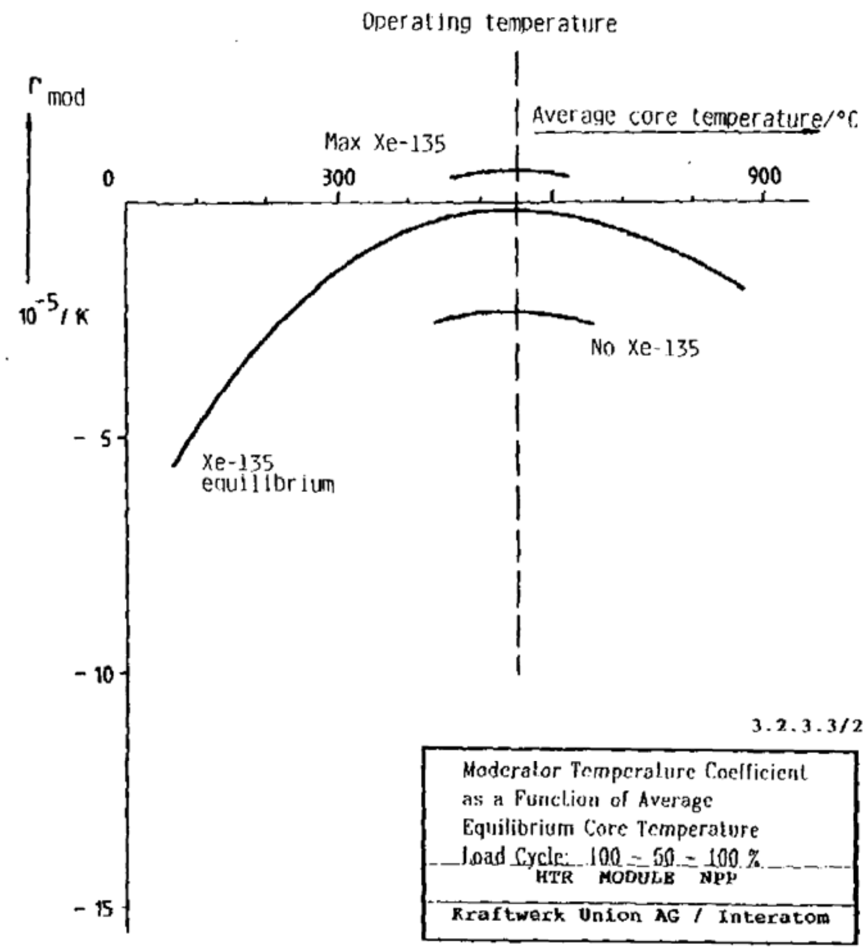
The resulting higher absorption rate plays the dominant role in the negative fuel temperature coefficient for reactivity.

The fuel temperature coefficient is practically independent of the load level (i.e., xenon inventory).

<u>Moderator temperature coefficient</u>

Figure 5-25 shows the moderator temperature coefficient is negative when the reactor is cold; its absolute value falls at first as the temperature rises and then rises again above operating temperature. In the equilibrium core, the moderator coefficient is slightly positive or negative in the region of mean operating temperature. Its absolute value, however, is so small as to be negligible at operating temperature. Neutron leakage plays an essential role in the moderator coefficient. Without it; this coefficient would be more positive at higher temperatures. In addition, the magnitude and polarity of the moderator coefficient are strongly influenced by the xenon 135 level in the core. This is due to the sharp drop in absorption by xenon 135 as temperature rises. The influence of the xenon 135 level is shown in the figure referenced above.

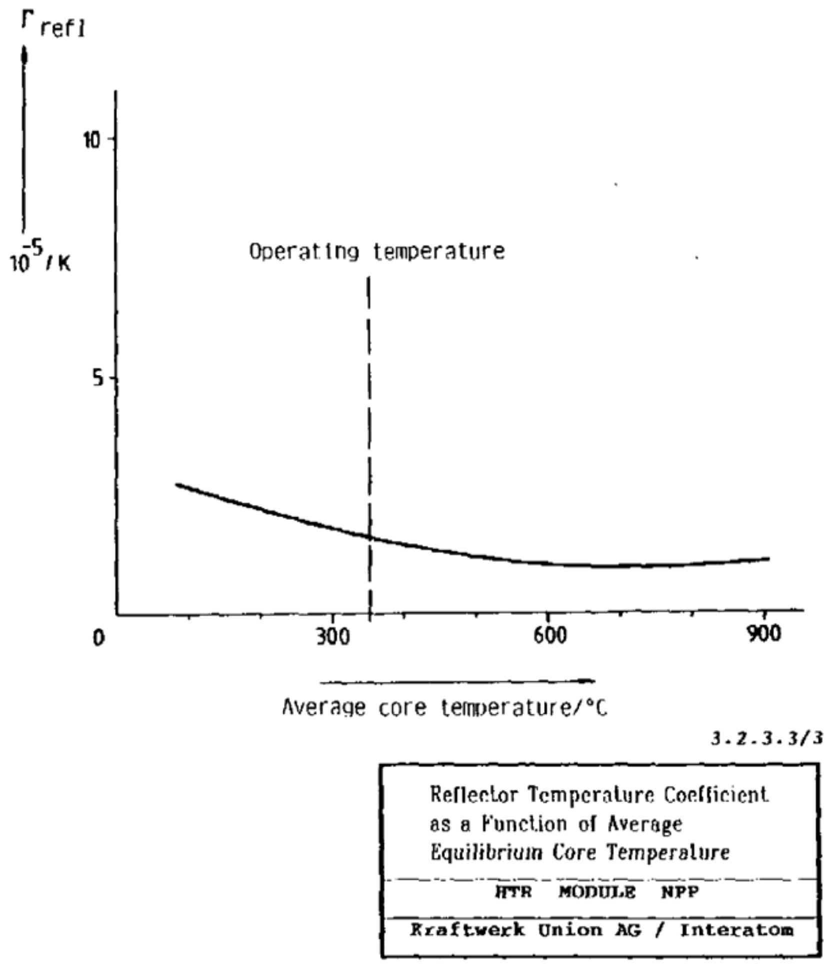
Figure 5-25: Moderator Temperature Coefficient as a Function of Average Equilibrium Core Temperature (Load Cycle 100%-50%-100%)



Reflector temperature coefficient

Figure 5-26 shows the reflector temperature coefficient is slightly positive for all operating states. However, since it does not become effective until after a relatively long delay, it does not influence the dynamic behavior of the reactor.

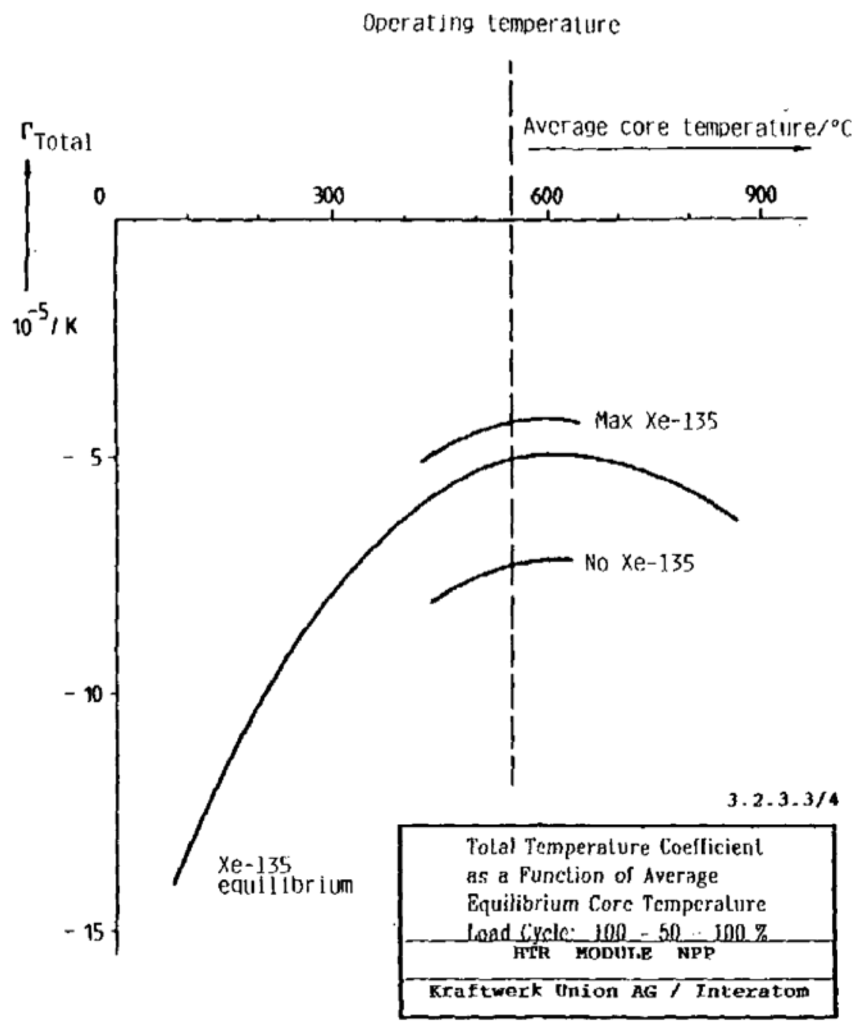
Figure 5-26: Reflector Temperature Coefficient as a Function of Average Equilibrium Core Temperature



Total temperature coefficient

Figure 5-27 shows the total temperature coefficient for reactivity (sum of fuel and moderator coefficients) is always negative because the dominant fuel temperature coefficient is negative over the entire temperature range. Under all operating conditions, its absolute value is sufficient to assure that the safety-related characteristics of the core are retained in all circumstances.

Figure 5-27: Total Temperature Coefficient as a Function of Average Equilibrium Core Temperature (Load Cycle 100%–50%–100%)



The overwhelming contribution of the fuel coefficient also assures that the inherent feedback mechanisms of the system take place quickly enough under accident conditions. Figure 5-27 shows the influence of the xenon-135 level on the total coefficient in the operating temperature range.

Temperature coefficient of first core

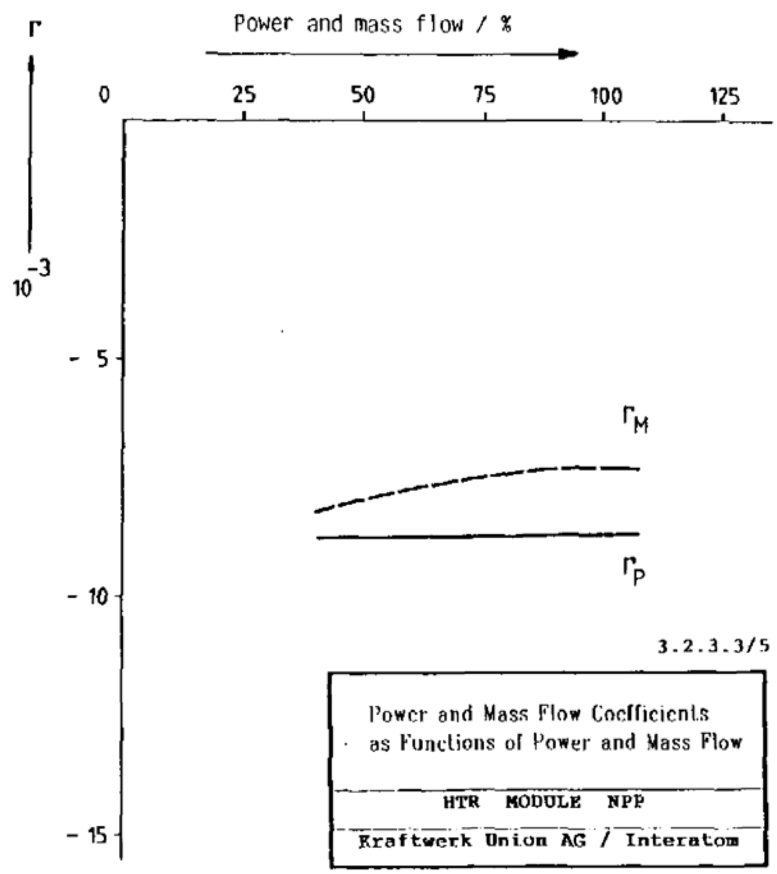
The fuel temperature coefficient of the first core is only half as large as that of the equilibrium core because of the smaller ^{238}U inventory. In contrast to the equilibrium core, the moderator temperature coefficient is dominant in the first core because of lack of fission products and plutonium isotopes, and high neutron leakage from the pebble bed. The first core moderator temperature coefficient is significantly negative and its absolute value far outweighs the reduction in the fuel temperature coefficient.

It is characteristic of the first core that the total temperature coefficient is notably more negative than that for the equilibrium core under comparable core conditions. Since the fuel is distributed almost homogeneously in part of the moderator (i.e., fuel element matrix), it is assured that feedback mechanisms of the system are sufficiently rapid in this instance as well.

Power coefficient

Figure 5-28 shows the power coefficient that describes the dependence of reactivity changes on power changes of the reactor core. On load changes, helium core inlet temperature and helium mass flow are kept constant.

Figure 5-28: Power and Mass Flow Coefficients as a Function of Power and Mass Flow



All of the above coefficients affect the power coefficient; the only significant ones, however, are the following:

- The fuel temperature coefficient
- The moderator temperature coefficient

Helium mass flow coefficient

Figure 5-28 also shows the mass flow coefficient, which describes the dependence of reactivity changes on helium mass flow changes in the reactor core. In determining it, reactor power and helium inlet temperature are kept constant. As with the power coefficient, the fuel and moderator coefficients are the major factors.

5.6.3 Long-Term Xenon Stability

Of the fission products created by uranium fission and decaying because of their instability, the isotope xenon 135 is significant because of its extremely high absorption cross section for thermal neutrons. A state of equilibrium arises at constant reactor power in which the amount of xenon 135 created (by beta decay of Iodine 135 and, to a lesser extent, directly by fission) is equal to the amount lost by beta decay and neutron absorption. Load changes result in transient xenon states.

On load reduction, xenon is destroyed primarily by natural decay. However, it continues to be created by decay of the Iodine 135 atoms that are still present. A maximum (xenon peak) forms due to differences in half life. As time progresses, xenon decay dominate, resulting in a new equilibrium level below the original equilibrium values.

This process is reversed on load increase. Xenon is destroyed primarily by neutron capture. Xenon production as a result of decay of Iodine 135, which starts to be created again by fission, is delayed; consequently xenon poisoning reaches a minimum (xenon burnout). The ensuing equilibrium level is higher than the original steady-state values.

Long-term stability of power density distribution in the reactor core is influenced by movement of the reflector rods and by the (I-135/Xe-135) interaction, which is in turn initiated by transient states of power density distribution. A distortion of steady-state spatial power density distribution takes the form of a xenon peak in regions of low power density and of xenon burnout in regions of high power density. The distortion of power density distribution is magnified by local increases or decreases of xenon poisoning. Neutron diffusion, which tries to compensate for spatial variations in neutron flux, has the effect of coupling spatial regions and thus working against this distortion.

After the xenon peak or the maximum xenon burnout has been passed, the distortion reverses itself and the system relaxes in the opposite direction. This interaction between local xenon poisoning and local power density increases can continue in the form of damped oscillations (xenon oscillation) in which spatially separate regions of decreased or increased power density oscillate in opposite phases.

The degree to which xenon oscillations are damped depends on how effective spatial coupling by neutron diffusion is. In the HTR-Module, with its mean core height of approx. 9.4 m and core radius of only 1.5 m, only damped axial xenon oscillations can take place. After approximately 50 hours, these oscillations decay to the extent that a more or less normal axial power distribution develops.

To evaluate long-term stability with regard to axial xenon oscillations, it is necessary to consider the maximum possible operational excitation that does not give rise to reactor scram and which therefore requires compensation by the reactor control system. This applies to the following distortions and operational changes:

- Inadvertent reflector rod travel.
- Movement of reflector rods on load change from nominal load to the lowest part-load duty point of 50 % nominal load.
- Reflector rod movement on transition from 50 % nominal load to full load. The distortion caused by xenon burnout, which has to be compensated, is greatest in this case if the transition from 50 % nominal load to full load is initiated at the xenon peak.
- Reactor run-up to nominal load after a hot start.

- Reactor start-up at the earliest possible time after scram (approximately 24 hours after shutdown at full load, assuming that hot start was not possible).

Of these distortions, run-up from the 50% part-load xenon peak leads to the largest changes in reactivity and the largest movements of reflector rods; in addition, it starts from a condition of reduced core temperatures at which the xenon 135 absorption cross section is larger than at nominal temperatures.

The following is a quantitative representation of this process:

Figure 5-29 shows the reflector rod position for the critical reactor as a function of time. After the reflector rods are withdrawn almost to their upper limit position to level out the part-load xenon peak, they are reinserted far beyond their normal operational position as a result of sharp xenon burnout after the return to power. This in turn induces (on average) a xenon peak, which is compensated by withdrawing the reflector rods to above normal position. This results in further, strongly damped xenon peaks and troughs. After approximately 50 hours the reflector rods have reached their normal operating position.

Figure 5-29: Critical Reflector Rod Position

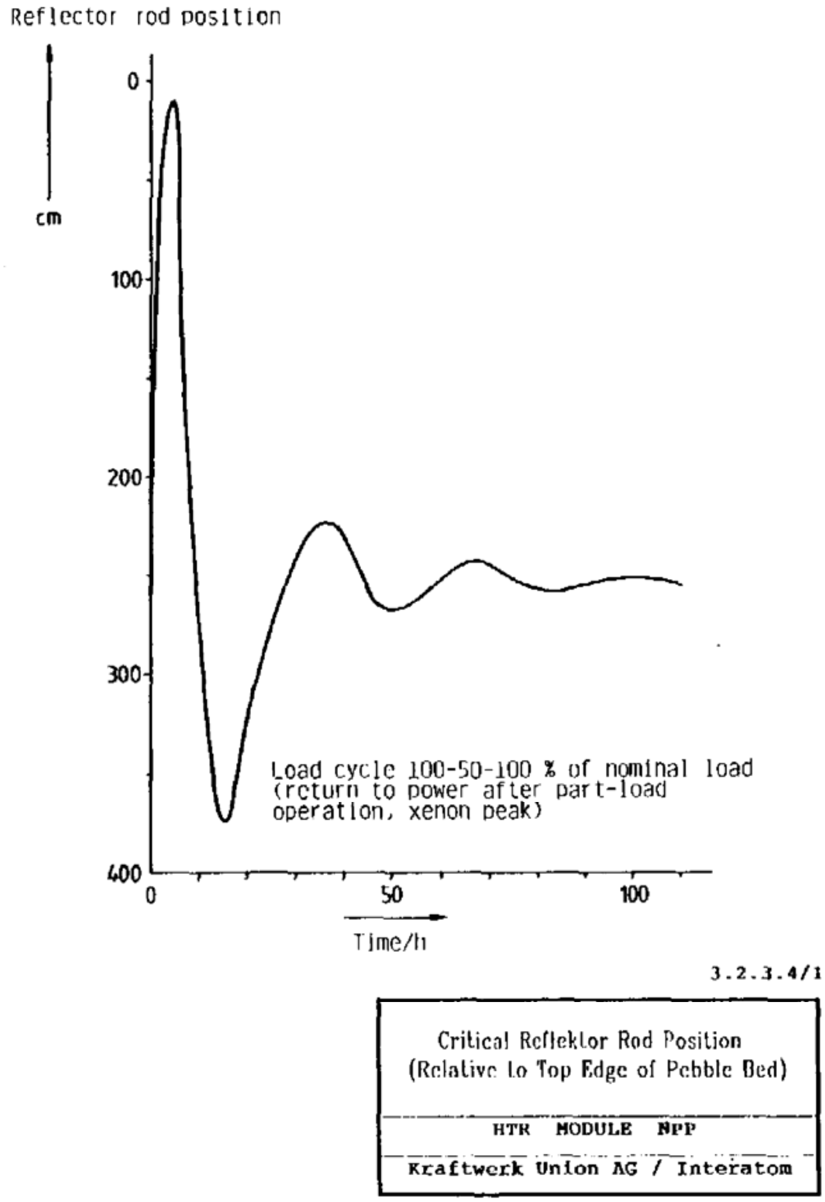


Figure 5-30 shows the absolute axial power density distribution at various points in time during this load cycle. It is apparent that local power density undergoes damped oscillation in time especially in the upper and lower regions of the core; this oscillation essentially follows the xenon-induced motion of the reflector rods.

Figure 5-30: Dependence of Radially Averaged Axial Power Density Distribution

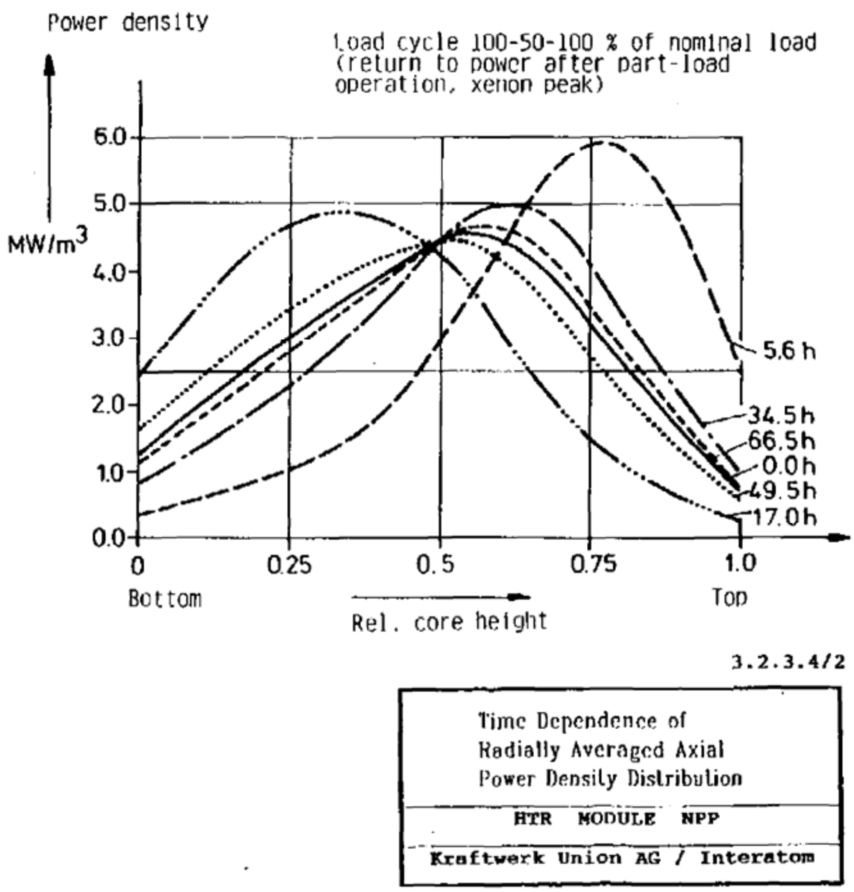


Figure 5-31 shows oscillation in time of xenon concentration for three selected regions of the core plotted against local equilibrium xenon concentration.

Figure 5-31: Time Dependence of Xenon Concentration

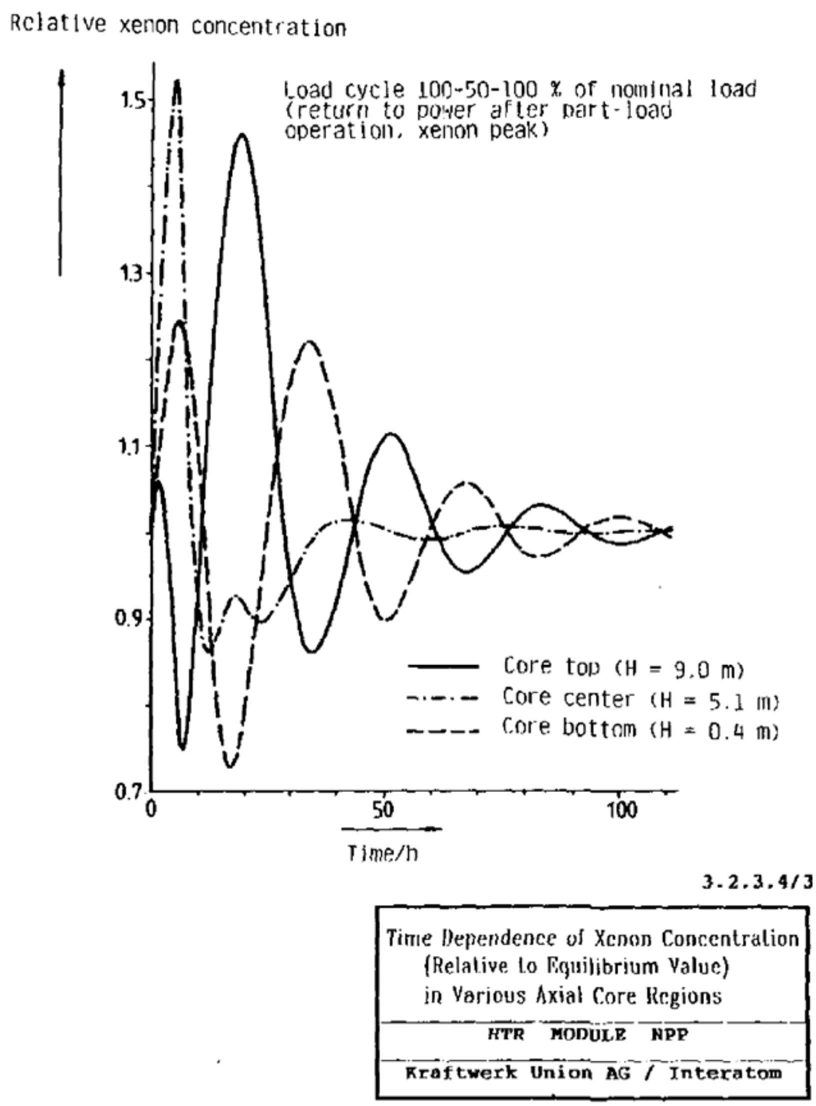


Figure 5-30 illustrates that local power density briefly exceeds maximum power density for normal full-load operation in the core regions between 2.5 m and 8.5 m axial height, exceeding the maximum by up to 30 % at an axial height of 7.5 m. However, particle and fuel element output remain below acceptable limits during these oscillations.

The spatial distribution of decay heat generation during a depressurization accident is significant from the safety standpoint. Because decay heat generation does not adjust to the actual power density distribution until hours or days have passed, no region of the core exceeds maximum decay heat generation rate subsequent to normal full-load operation integrated over 30 hours, which is relevant over this period in the context of a depressurization; therefore, fuel element temperature does not exceed approx. 1600 °C in the event of a depressurization accident concurrent with oscillations in power density distribution.

5.6.4 Shutdown Margin Issue Conclusion

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.

Furthermore, it has been shown that the HTR-Module reactor geometry, with its reflector rod worth and positioning, can address the Xenon stability issue, which otherwise could pose a problem for any long core configuration.

The problem of core recriticality is at low temperature (< 50°C). This prevents the core from being cooled down to ambient temperature for maintenance. However, this is not an 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.

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 far outweigh the low temperature shutdown margin design issue.

5.7 Online Refueling

One of the major characteristics of the PBR concept is the possibility of loading and unloading small portions of the fuel (pebble by pebble) while the reactor is operating based on its particular Fuel Handling System and spherical fuel elements.

This possibility is a key feature of PBR because it potentially allows very high availability of the reactor since refueling outages are not required. This feature is particularly attractive for process heat applications, which for a large majority would require very high availability of the heat source.

However, past experience of PBR has not been able to prove high availabilities:

- AVR averaged time utilization rate: 66.4%, best year (1976) with 92%
- THTR time utilization rate: 61% in 1987, 52% in 1988

In addition, the Fuel Handling System itself was responsible for 3% (among the 33.6%) of unavailability of the AVR due to operating issues.

The question whether the on-line refueling is a real advantage in terms of availability or if it could be, on the contrary, source of malfunction and unavailability is addressed.

The approach is to consider that the on-line refueling concept is of benefit to reactor availability only if:

- The Fuel Handling System is highly reliable/available, and

- The Fuel Handling System has adequate performances in terms of circulation rate. As presented further down, this rate is linked to the BUMS (Burn-Up Measurement System) performance.

Thus, the reliability of the Fuel Handling System is assessed in two parts. First, the performance of the system as a whole is studied. Second, specific questions related to the performance of the BUMS performance are addressed. This assessment mainly relies on a study of the AVR and THTR operating experience. In addition, PBMR and HTR-PM qualification programs are briefly presented to account for the most up-to-date experience in PBR qualification programs.

5.7.1 AVR Fuel Handling System Experience

The AVR has been used as an experimental reactor for mass testing of the fuel spherical elements, testing a large number of components, and testing the fuel handling system from fresh fuel feeding, via fuel circulation to used fuel intermediate storage [63][64]. It shall be taken into account that the AVR was first of its kind.

5.7.1.1 Design of the AVR Fuel Handling System

Figure 5-32 and Figure 5-33 depict the configuration of the AVR FHS.

Figure 5-32: AVR Fuel Handling System Schematic

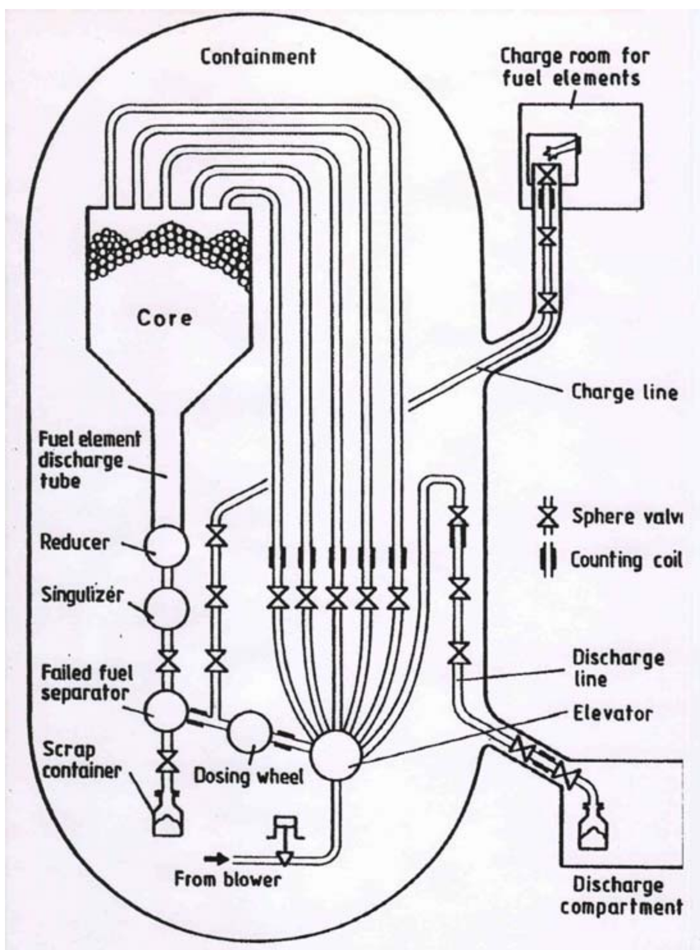
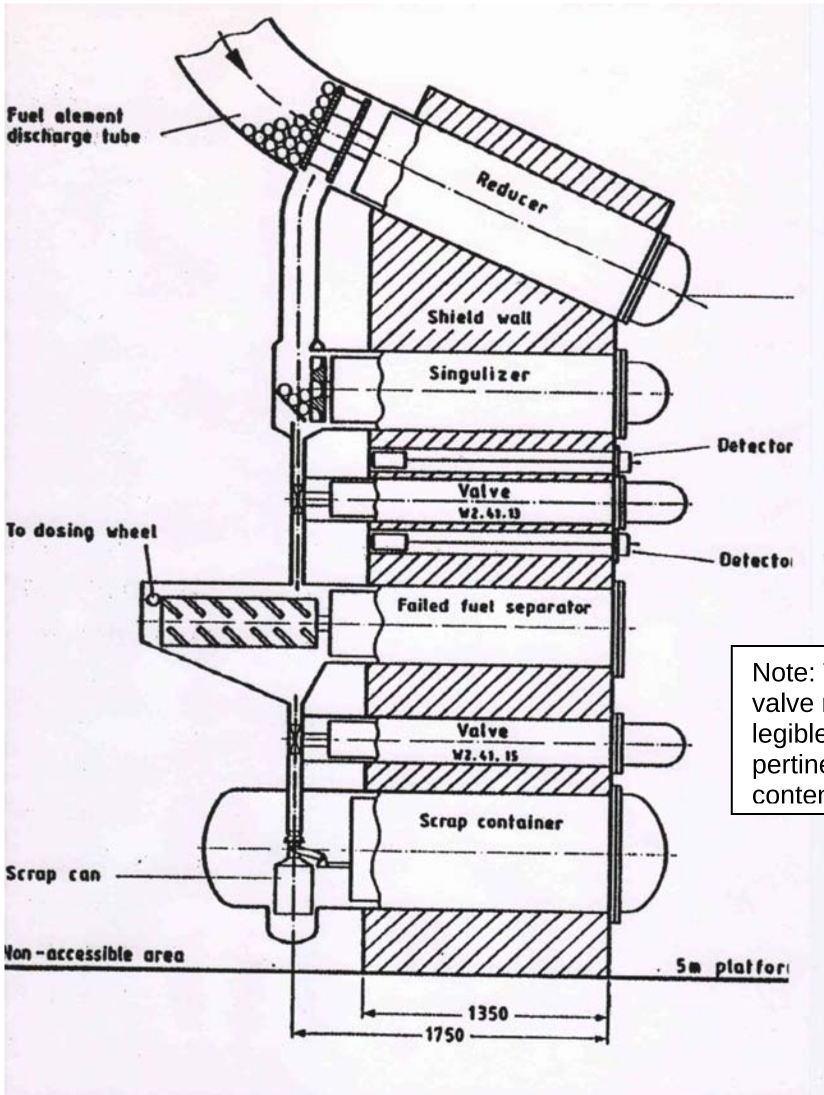


Figure 5-33: AVR Fuel Handling System Discharge Wall



Note: Text indicating valve numbers is not legible and is not pertinent to technical content

The lower part of the core, which has a diameter of 3 m, is shaped like a funnel and opens into the pebble discharge tube with an inside diameter of 500 mm. Its end is formed by the reducer. In principle, the reducer is a rotating disk containing a radial slot that is about 62 mm wide. It rotates slowly, about 1 revolution per minute. Small quantities of pebbles, with a diameter of 60 mm, drop through this slot stochastically. The pebbles fall into a storage queue in front of the singulizer. This scoops the pebbles out of the queue and transfers them individually to the damaged fuel separator. This sorts-out damaged pebbles and scrap fragments. Sorted-out scrap pebbles drop into the scrap container. Undamaged pebbles roll to the dosing wheel, which conveys individual pebbles from the column of pebbles to the elevator.

The burnup and differentiation measurement is performed in the elevator. The different pebbles are sorted out here between pure graphite pebbles, fuel pebbles containing different mass of fuel, and test pebbles (e.g., temperature measurement pebbles). The burnup of fuel pebbles is also measured by the Burnup Measurement System (BUMS).

Depending on the results of the measurement, the elevator automatically rotates to one of the six positions: five to the core and one to the discharge line. The pebble is transported pneumatically by gas pressure. Four of the five lines to the core lead to the outer feeding positions, and one lead to the centre feeding position (AVR had a 2 zone core).

Each of the charge and discharge lines consists of a system of two locks built by four sphere valves through which the pebbles roll in groups of ten. When charging or discharging the pebbles, the individual locks are pumped down with compressors and subsequently flushed with pure gas.

Apart from the positioning of fresh fuel pebbles in the charge room or interventions in case of malfunction, the operation of the Fuel Handling System is fully automatic. During full power operation, about 500 pebbles are circulated each day, 60 fresh fuel pebbles are fed in and 60 spent fuel pebbles are discharged accordingly.

With the exception of the reducer and the singulizer, all other components of the Fuel Handling System can be isolated from the core by sphere valves. When closed, valves offer gastight closure with double seal. Repairs can therefore be performed during power plant operation if necessary.

5.7.1.2 Failures and Improvements

The AVR Fuel Handling System experienced problems caused by several of its components, which is typical of a first of kind system. For instance, insufficient knowledge of the specificities of in dry helium tribology induced a number of blockages that would be prevented with the present knowledge.

A general measure for the quality of the FHS is the number of pebbles that were transported by it. These operational figures of the AVR in the period 01-01-1969 to 12-31-1988 are:

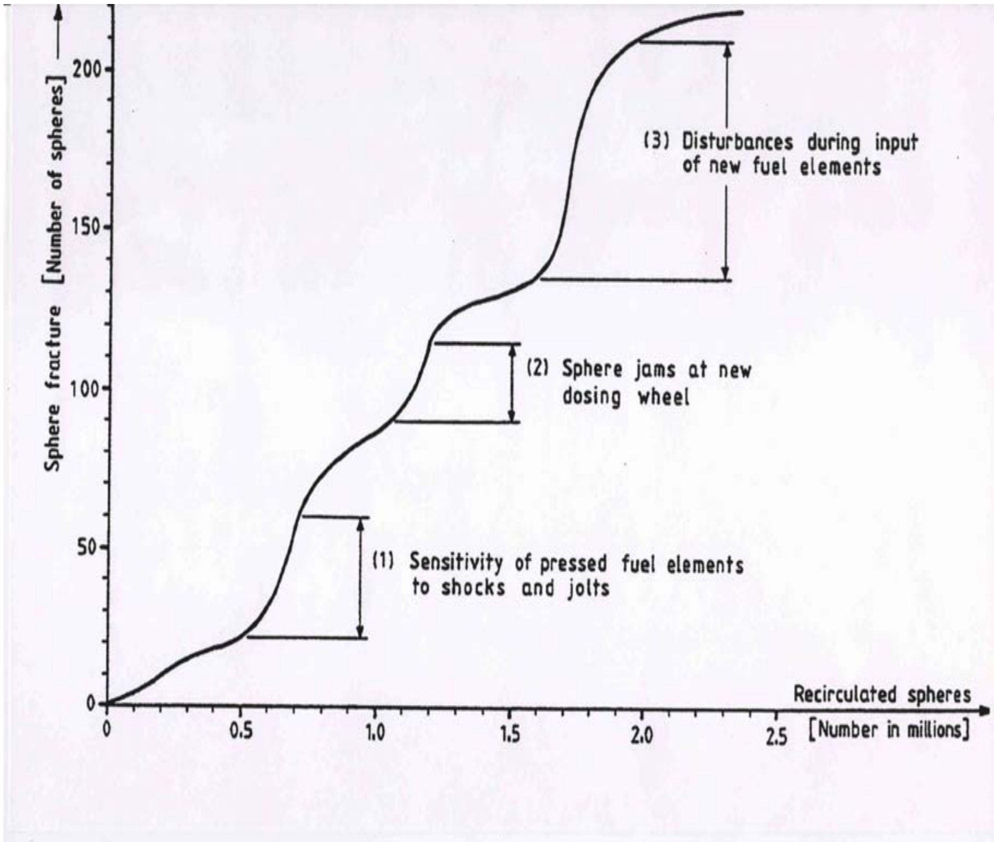
Number of loaded-in fuel pebbles	290,645
Number of loaded-in graphite pebbles	79,950
Number of discharged fuel pebbles	180,404
Number of discharged graphite pebbles	76,651
Number of circulated pebbles	2,408,974*

* This is the number from start of filling pebbles in the core, which was earlier than 01-01-1969.

The loads to which the pebbles are exposed during handling are of great importance to the FHS. An additional measure for the quality of the FHS is the number of damaged pebbles in relation to the number of circulated pebbles. In AVR the respective statistic is given in Figure 5-34. In these statistics, 3 significant increases shall be underlined and explained:

1. The first increase is from a new fuel pebble type: pressed,
2. The second increase resulted from the dosing wheel issue,
3. The third increase is from the fuel pebble feed line.

Figure 5-34: AVR FHS Sphere Fracture Diagram



In summary: The number of damaged pebbles in the Fuel Handling System of the AVR in 18.5 years of consideration was in the range of 1 to 2 damaged pebbles in 10,000 circulated pebbles. In the final three years, of operation, 1986 to 1988, after a new dosing wheel was installed, the number went down to 1 to 2 damaged pebbles per 100,000 circulated pebbles.

An additional important AVR result is that the damages of fuel pebbles did not yield increases in the cooling gas activity. That indicates that coated particles were not damaged in damaged pebbles.

5.7.1.3 Reliability of the AVR FHS

The reliability of the FHS of the AVR can be evaluated considering the following results:

- The Fuel Handling System of the AVR was unavailable during 12.8% of the operating time,
- The forced outages of the reactor caused by the Fuel Handling System unavailability represented 3% of the operating time.

These results show that a large number of components of the FHS can be repaired without stopping the power production.

The numbers of outage hours, leading to these results, are given with a special differentiation on the various components in Figure 5-35, which presents a diagram of the outage times in hours of the components and resulting forced outage time of the generator for the period 01-01-1970 to 06-30-1988 with 2315 million circulated pebbles, and Figure 5-36, which displays non-availability in percentage as a function of time from 01-01-1970 to 06-30-1988, in total and for the components. The components responsible for most disturbances were not as significant after 1984, because of improvements. It needs to be mentioned that the time period for this evaluation is only 01-01-1970 to 06-30-1988, and therefore shorter than the operational period. The reasons are, first, to exclude about one year at the beginning with initial problems; second, to finish the study earlier than the end of operation.

Figure 5-35: AVR FHS Reliability Diagram

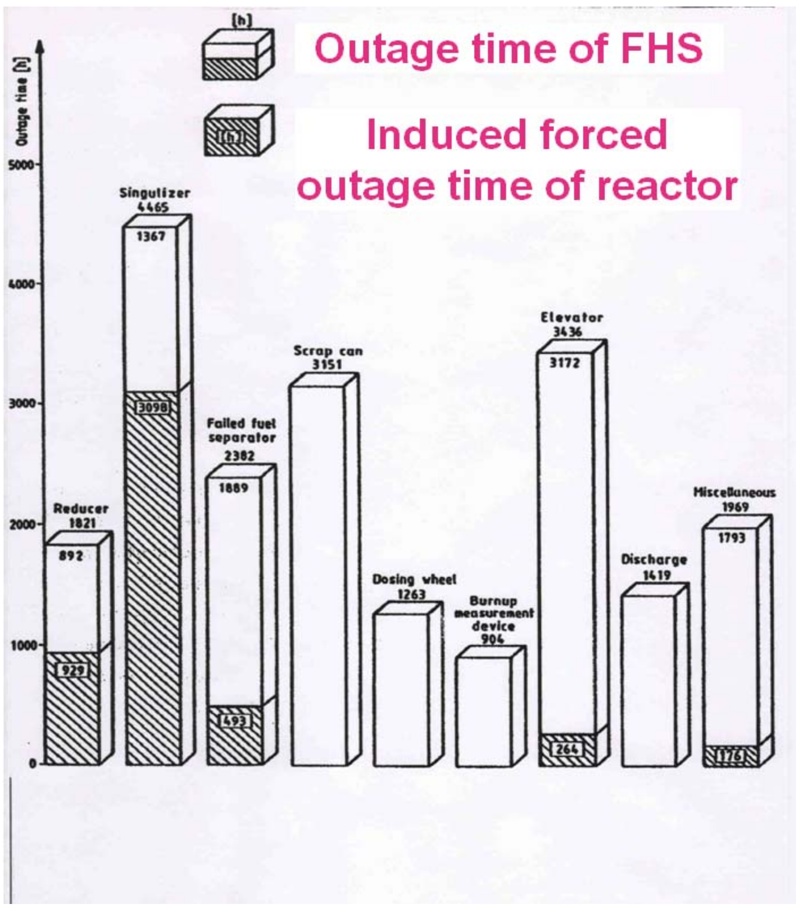
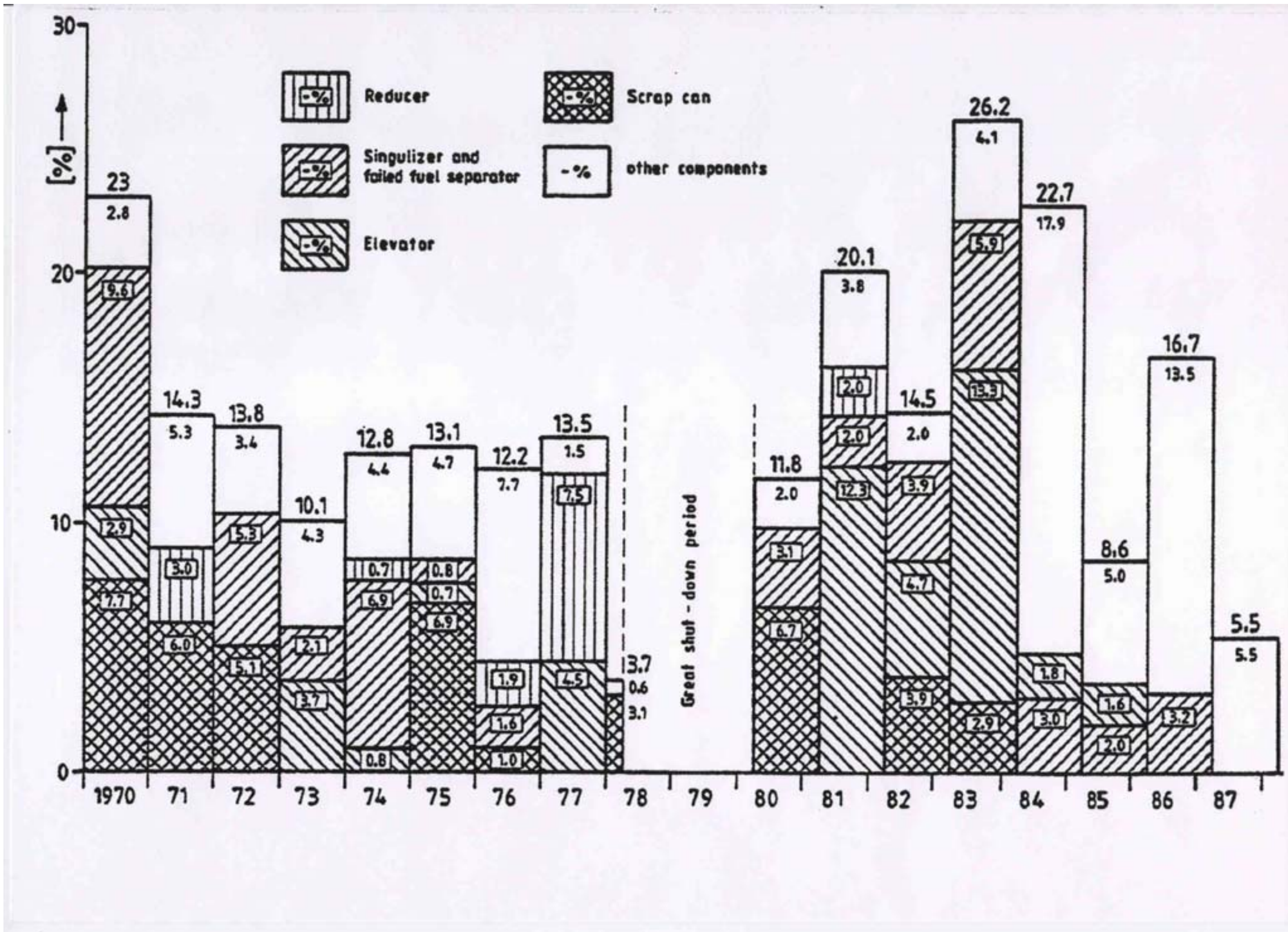


Figure 5-36: AVR FHS Reliability Diagram by Component



5.7.2 THTR Fuel Handling System Experience Feedback

5.7.2.1 Design of the THTR Fuel Handling System

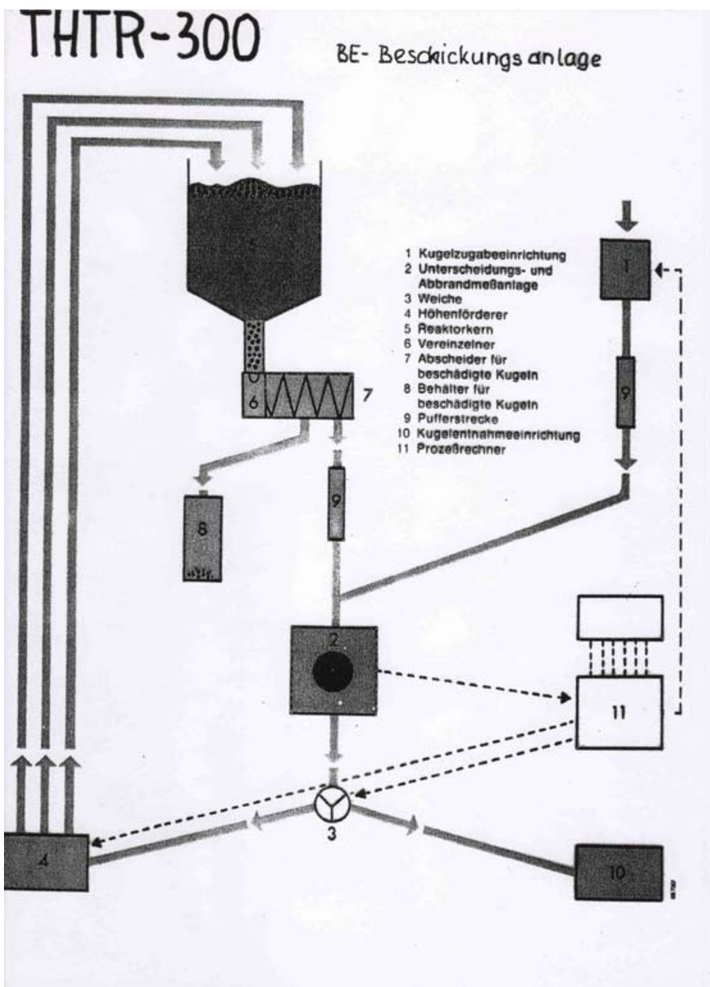
The concept of the FHS of the THTR is the same as that of the AVR [65]: handling and transportation of pebbles by pneumatic and gravity forces. Figure 5-37 gives a general scheme of the system. There were many lessons learned from AVR. The most important were:

- The invention of the double seat isolation valves,
- The combination of respective valves to valve blocks,
- The integration of shielding in the wall of the blocks,
- The combination of reducer, singulizer and separator.

In addition, the improvements included, for instance, the simplification that the axis of the reducer does not need to be declined as in AVR, but could be horizontal, as shown in Figure 5-38. This change had been tested before with the result that an inclined bottom of the pebble box in front of the reducer (not shown in Figure 5-38) would be sufficient.

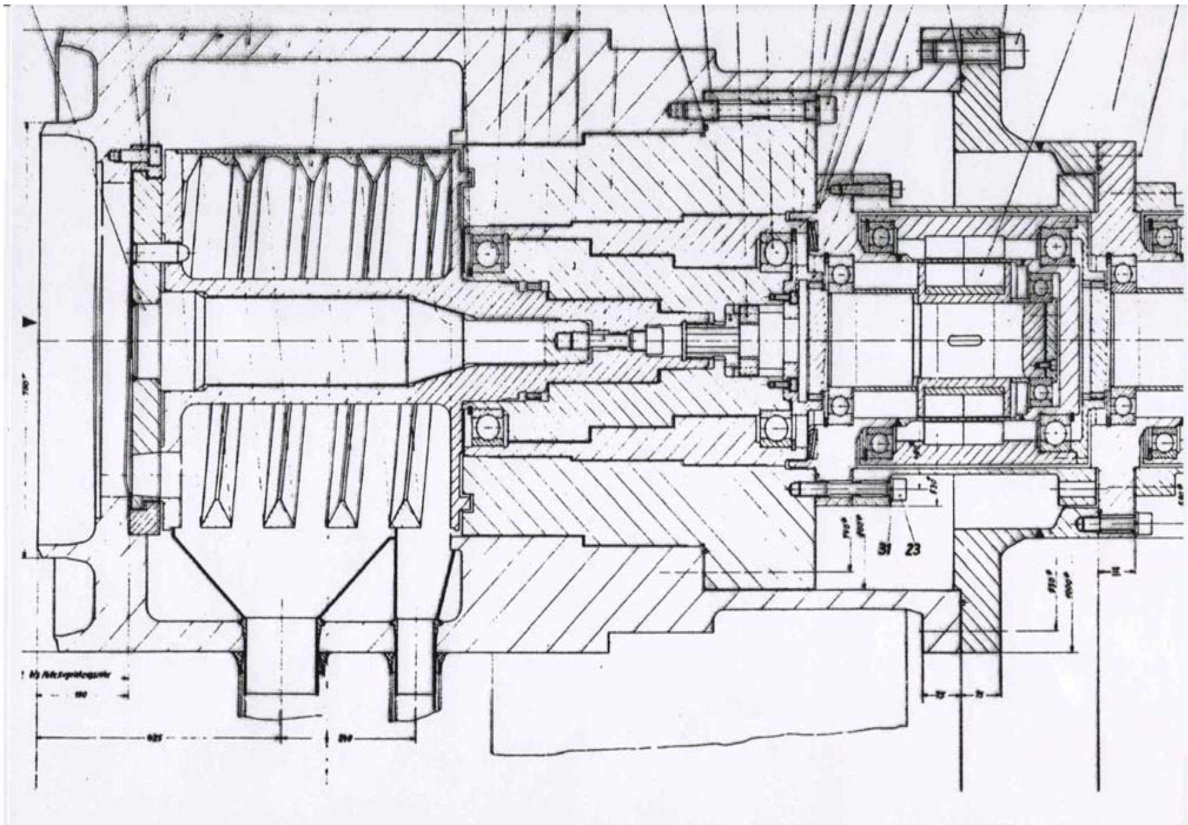
An additional consequence of that change was a triple simplification, see Figure 5-38. That is the combination of three functions in one single unit, called singulizer-separator. The reducer function is realized by a rotating disk, but with a hole of 65 mm diameter (instead of a slot as in AVR), which in average gates-out one pebble per rotation. The singulizer function is realized by a rotating spindle, directly starting behind the disk and rotating with the same speed, because undamaged pebbles roll in that spindle from the disk to the “good” pebble funnel and pebble line. The damaged pebble separator function is realized by the form of the spindle, which is so that the width between the rail of the spindle is about 57 mm. Consequently, damaged pebbles fall down by gravity into the funnel, which collects all the damaged pebbles and leads them to the failed fuel cask. The complete testing of the rolling pebble in the spindle is achieved by the triangular profile of the rail with slightly different angles to the base of the triangle.

Figure 5-37: THTR FHS Schematic



1: Charge station, 2: Burn-up measurement, 3: Diverter, 4: Lift lines, 5: Core, 6: Reducer, 7: Separator for damaged pebbles, 8: Container for damaged pebbles, 9: Buffer, 10: Discharge Station,11: Computer

Figure 5-38: THTR FHS – Vertical Cut through Reducer/Singulizer/Separator



Note: Text indicating dimensions is not legible and is not pertinent to technical content.

5.7.2.2 Lessons Learned from THTR FHS Operation

The Fuel Handling System of the THTR has circulated about 2 million pebbles. This number is about the same as for AVR, but was produced in only its two years of operation.

One initial problem was the back-stream (against the intended pebble flow) through the singulizer gate hole in the rotating disk of the singulizer-separator. It was produced by an overpressure in the spindle compartment at higher load of the operating plant (above 70%). It was solved by setting up a direct opening to the pebble box, located in front of the rotating disk.

Some sphere blockages have also occurred, due to pebble fragments generated upstream in the core (see hereunder the damaged pebble issue). Strong counter-pressure has been used to solve this kind of problem but consideration must be made of the risk of pressure relief by the protective devices of the circuit and subsequent activated dust rejection to the environment (which happened in THTR in May 1986).

In addition, the reliability of I&C was not satisfying, and notably that of the sphere counters. Progress made since the 1980s in I&C performance should, however, largely improve the reliability of a future system.

Overall, the reliability of the Fuel Handling System of the THTR was improved after a number of initial problems had been solved.

However, a relatively high portion of the pebbles were damaged in THTR, which could have been attributed to the FHS without further explanation. Starting from the end of 1985, an unexpectedly high number of damaged pebbles occurred, which were sorted out by the scrap separator. Seventeen thousand (17,000) damaged spherical elements were discarded. The damaged pebble ratio was 1.5% at the beginning of the refueling operation but continuously decreased to reach 0.6% at the end.

It has been confirmed, by experimental data, that this damage was caused by frequent and deep insertion of the in-core control rods during the THTR commissioning phase. This type of control rods are not envisioned in modular HTGRs like the HTR-Module.

5.7.3 The PBMR FHS Qualification Program

The concept of the FHS of PBMR is the same as that of the AVR and THTR. The below discussion is based on the information contained in References [66], [67], and [68].

PBMR has started and operated the Helium Test Facility (HTF), in the Research Centre, Pelindaba, South Africa. It was operating from April 2007 to March 2010 for all important systems of the PBMR, but operations have been interrupted since April. These tests included the testing of components of the Fuel Handling System at scale 1:1 (see Figure 5-39) with representative pressure, temperature and flow.

The following are the main topics that tests intended to represent and qualify:

- Dynamic process behavior
- Process control evaluation
- Operation in dust environment
- Operational reliability

The tests identified for the Fuel Handling System on the HTF mainly comprise sphere conveying, transportation and gas control. The following tests on sub-systems of the Fuel Handling System can also be performed:

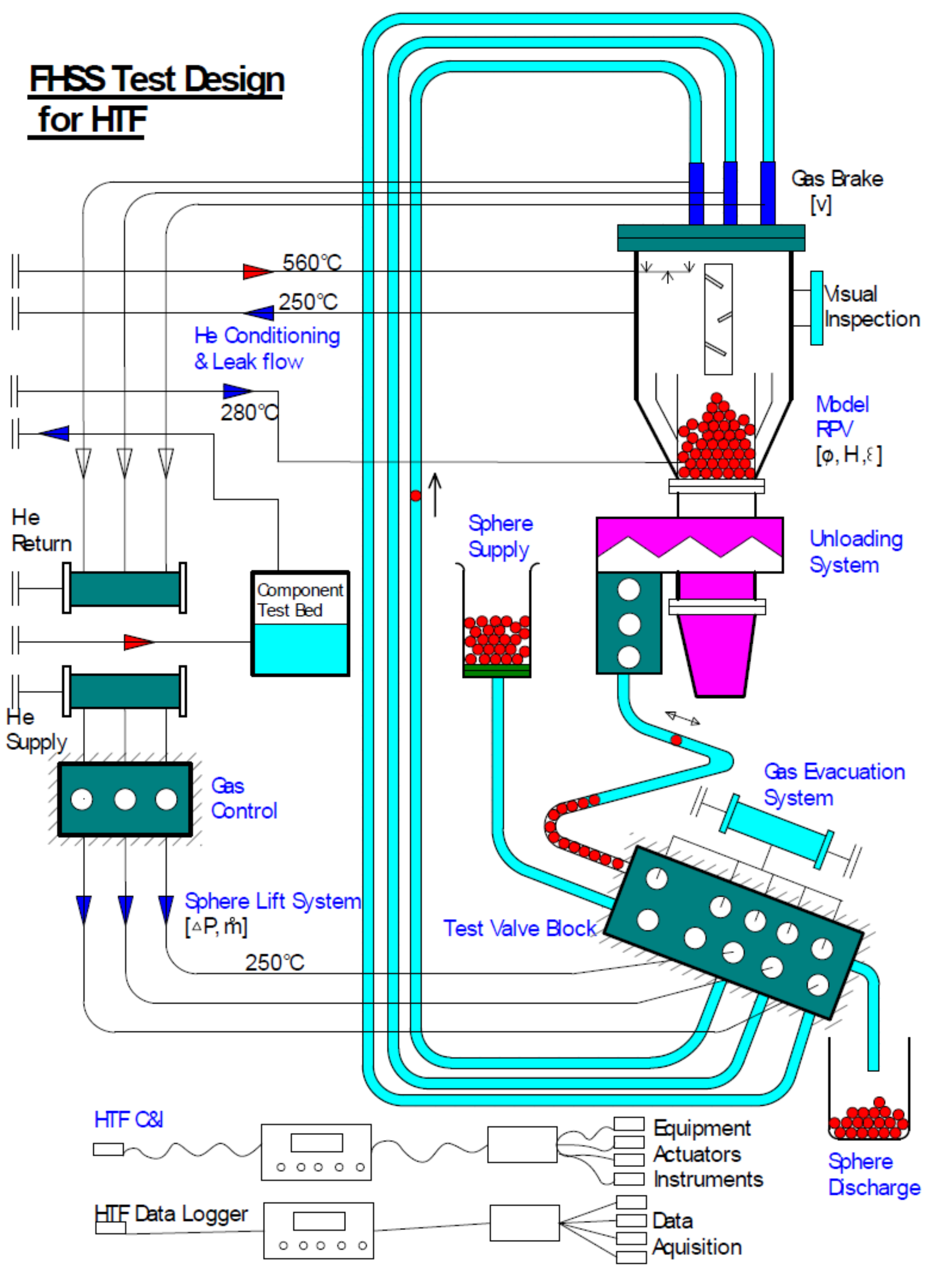
- All types of valves,
- The core unloading device,
- The tank unloading device,
- Sphere counters,
- The mechanical sphere break,
- The pneumatic sphere break.

The PBMR Fuel Handling System test achievements are:

- More than 19,000 pebble passes were completed,
- Function of pebble counter was tested,
- Leak tests were performed to determine the helium leak flow rate.

The results of these tests have, however, not been much published yet.

Figure 5-39: Fuel Handling System Test Design for HTF



5.7.4 The HTR-PM Fuel Handling System Qualification Program

The HTR-PM FHS adopted the international experience at design and operation of similar systems, especially based on that of HTR-10. However, some key components and technologies were improved so that the Fuel Handling System becomes simpler and more reliable. All of the improved components and technologies will be tested in a full-scale hot testing facility, and some of them were validated and verified with the help of cold testing facilities.

Up to present, a demonstration device of critical equipment has basically completed a stand-alone type test and functional test. On this basis, systematic studies and comprehensive tests are carried out. Those studies and tests include on-line test benches of discharging devices, transport and transfer devices, unitized fresh fuel feeding devices, high activity gamma spectrum BUMS technology and fuel handling control system.

5.7.5 FHS Experience Conclusion

In both the AVR and THTR, numerous problems have been encountered with the FHS, but most of them would find answers today:

- Dry helium tribology has now been studied more extensively and it is know that testing of wear mechanism in helium are generally necessary,
- Blockages by the helium counter-flow at the THTR singulizer were due to an hydraulic design error that has no reason to be reproduced,
- Blockages due to damaged pebbles fragments was a specific THTR issue due to the in-core control rods insertions (again, eventually explained by representative tests).

Overall, the significant time of unavailability of the AVR and THTR Fuel Handling Systems shall be considered as inherent to FOAK system operation, but not as intrinsic to the PBR Fuel Handling System concept.

Moreover, the ability of running the reactor for a certain time (to be carefully assessed) while the Fuel Handling System is unavailable is very beneficial to limit the dependence of the reactor on the FHS availability and should largely benefit to the reactor availability in case of residual problems.

Lastly, the necessity of important test qualification programs is now well understood (see PBMR and HTR-PM scale 1:1 hot test facilities) and their benefits should be very valuable to support successful operation of a future PBR Fuel Handling System.

5.7.6 PBR Burnup Measurement System Performance Assessment

5.7.6.1 Impact of BUMS Performance on On-Line Refueling Capability

In a PBR, each pebble loses its identity after insertion into the core. The procedure, therefore, is to look at all fuel elements every time they leave the core through the exhaust chutes and decide whether to re-insert the pebble or to remove it, based on the comparison of its burnup level to the design limit.

There are several techniques to check the burnup status of a fuel element:

- Determination of remaining fissile content:

- By attenuation of a neutron beam (utilized in AVR before 1981),
- By measuring the power variations of a small critical device (utilized in THTR with a 100 W reactor).

- Measuring a fission product content (^{137}Cs) that can be uniquely correlated with the total accumulated burnup (utilized in AVR after 1981 and in HTR-10). This method is also retained for HTR-Module, HTR-PM and PBMR.

Thus, modern consensus is to concentrate on the BUMS based on the measurement of fission product ^{137}Cs in a fuel sphere with high precision spectroscopy. This method allows adequate precision provided that the measurement and decay times are sufficient.

HTR-Module design considers an average measuring time of 10 seconds (with a ±5% statistical error) to support the design circulation rate. The acceptable error window is set by two requirements:

- No pebble should stay in the reactor too much time above the average time because the pebble are not qualified to reach too high burnups,
- No pebble should be taken out of the reactor too much before it has reached the burnup limit for obvious economical reasons.

In the HTR-Module, the recirculation time of each pebble is given by the pneumatic circulation time between the forwarding bank (which channels the fuel pebbles to their appropriate destinations) and the reactor core, which amounts to 16 seconds and determines the average circulation rate of 5360 pebbles discharged from the core each day (most of them, 5000, are recirculated and actually need the 16 s).

However, as shown hereafter, past BUMS needed much more measurement time per pebble (ex: 30 s in AVR), so that the BUMS could become the limiting system for the FHS circulation rate if the expected performance were not reached. In addition, this system is the main high technology device of the Fuel Handling System and, as such, its potential to reach the design performance shall be carefully evaluated.

5.7.6.2 The Cesium-137 Spectroscopy Principle

Spectroscopy allows measuring the content of a specific atom (and its isotopes) in a material by interpretation of its radiation spectrum.

The chosen nuclide, ^{137}Cs, has unique properties that make it an ideal nuclide to measure burnup:

- It has a long half-life (30.1 years) compared to the time a fuel sphere spends in the reactor core resulting in an activity that increases linearly with the number of fissions and thus the burnup; any deviations from linearity can be numerically accounted for.
- The fission yields of ^{137}Cs for ^{235}U, ^{239}Pu, and ^{241}Pu fission are very similar, so that the ^{137}Cs content is almost independent of the mix of fissile species present in the fuel that changes throughout the lifetime of a fuel sphere in the reactor core.
- Although ^{137}Cs itself is not a gamma emitter, it decays to a short-lived daughter, ^{137m}Ba (half-life 2.552 minutes), with a probability of 0.946 per decay. ^{137m}Ba emits a gamma ray of energy 661.6 keV with a

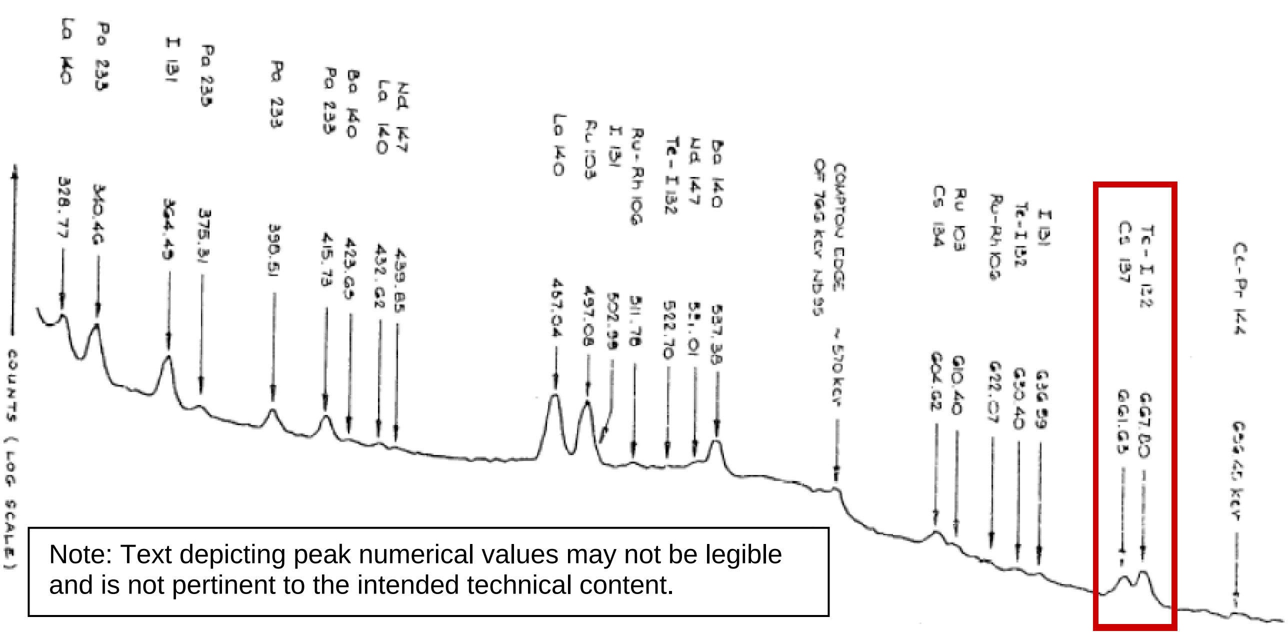
probability of 0.898 per decay. Thus for every ^{137}Cs decay, the probability of the emission of a gamma ray is 0.850. In nuclear spectroscopy terminology, the 661.6 keV gamma is called the ^{137}Cs line.

- While cesium is considered as a volatile fission product, modern HTGR fuels guarantee retention ratio inside the fuel of better than 10-4, therefore placing no restriction on ^{137}Cs as a reliable burnup indicator.

Gamma spectrometry is a powerful tool to determine fission product content of irradiated reactor fuel elements and also the releases from the fuel, when components outside the fuel element are assayed. Germanium detectors are used for this application. Typically, these measurements are performed weeks, months or even years after the irradiation and measurement times are hours. The statistical error in activity determination can be reduced to a relative level of a few percent when the appropriate conditions are fulfilled.

Figure 5-40 shows a typical gamma spectrum of irradiated fuel after three weeks cooling in the 300-700 keV range, including the ^{137}Cs line at 661.6 keV that overlaps with the neighboring ^{132}I peak.

Figure 5-40: Typical Gamma Ray Spectrum for Fuel Element with 3 Weeks of Cooling



5.7.6.3 Cesium-137 Spectroscopy Performance Limitations

The performance of the measure shall be evaluated by the pair: measurement time – statistical error (an increase of the measurement time extends the measured data and hence lowers the statistical error).

Two main difficulties should be tackled when measuring the burnup via ^{137}Cs gamma line spectroscopy:

- Soon after leaving the core, the ^{137}Cs count rate (number of gamma detected by the detector around 661.6 keV each second) will be dominated by short-lived fission products with intense high energy of 1 to 2.5 MeV gammas. Compton scattering in the Germanium detector produces a massive broad Compton background. The long-lived ^{137}Cs produces only relatively low count rates and its contribution to counts in the 650 to 680 keV energy range is not significant. In the extreme case, where there are 10,000 background counts and only 100 ^{137}Cs counts, the statistical uncertainty of the background is ±100 counts and the ^{137}Cs contribution becomes insignificant.

- There are three short-lived competing neighbors that may additionally swamp the ^{137}Cs peak (Table 5-11 below).

Table 5-11: Peaks in Vicinity of Cesium-137

Peaks in the vicinity of ^{137}Cs				
	^{97}Nb	^{137}Cs	^{143}Ce	^{132}I
Photo Peak Energy (keV)	657.9	661.6	664.6	667.7
$T_{1/2}$	16.8 hours	30.07 years	1.377 days	3.2 days
λ (s^{-1})	1.15×10^{-5}	7.30×10^{-10}	5.83×10^{-6}	2.507×10^{-6}

As a result, the precision of short time measurements is significantly improved for:

- Higher cooling times because:
  - Short-lived, high energy fission products progressively disappear, reducing the background noise
  - Neighboring fission products disappear faster than 137Cs, reducing the statistical error due to overlapping of the peaks
- Higher burnups because:
  - Fissile content decreases, reducing the background noise
  - As a fission product, 137Cs quantity increases, increasing the associated counts, which reduces the statistical error

Both effects are illustrated by Figure 5-41, which presents AVR statistical errors in burnup determination of highly burnt spherical fuel elements after 30 days of cooling as a function of measurement time, and Figure 5-42, which presents statistical uncertainties in burnup determination from the inventory of 137Cs of the irradiated spherical fuel element of AVR. The “PBR” curve is a simulation result in different conditions.

Figure 5-41: AVR Statistical Errors in Burnup Determination

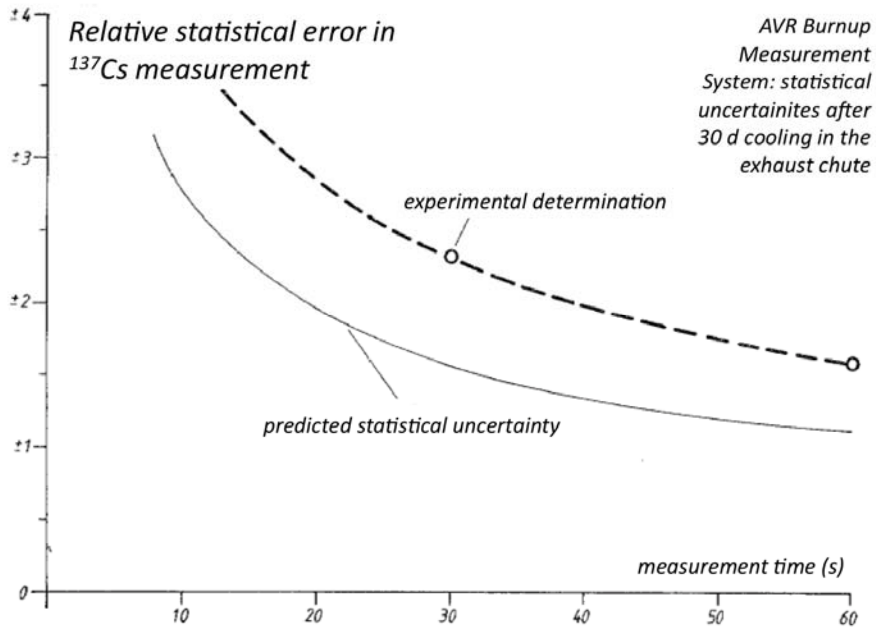
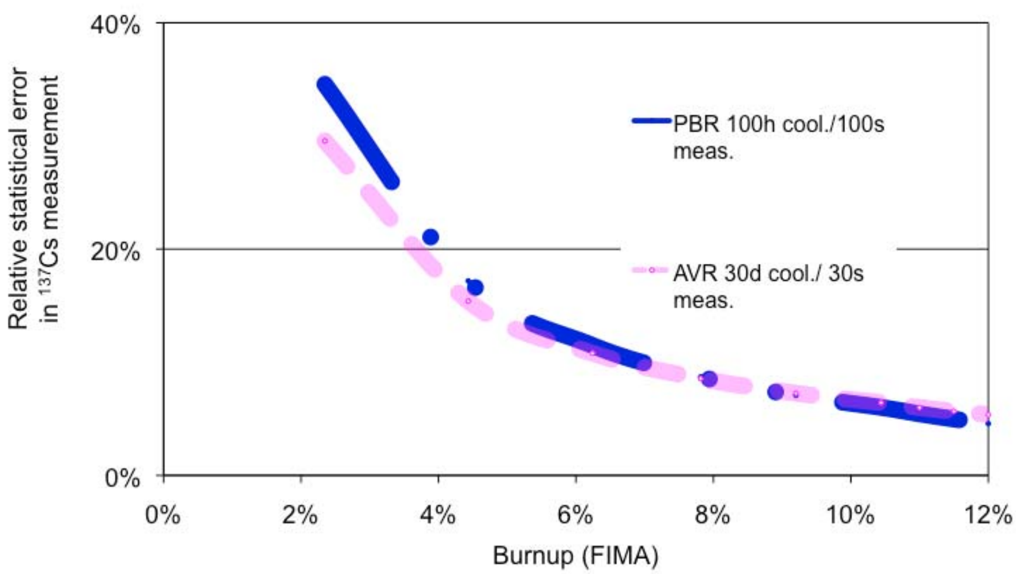


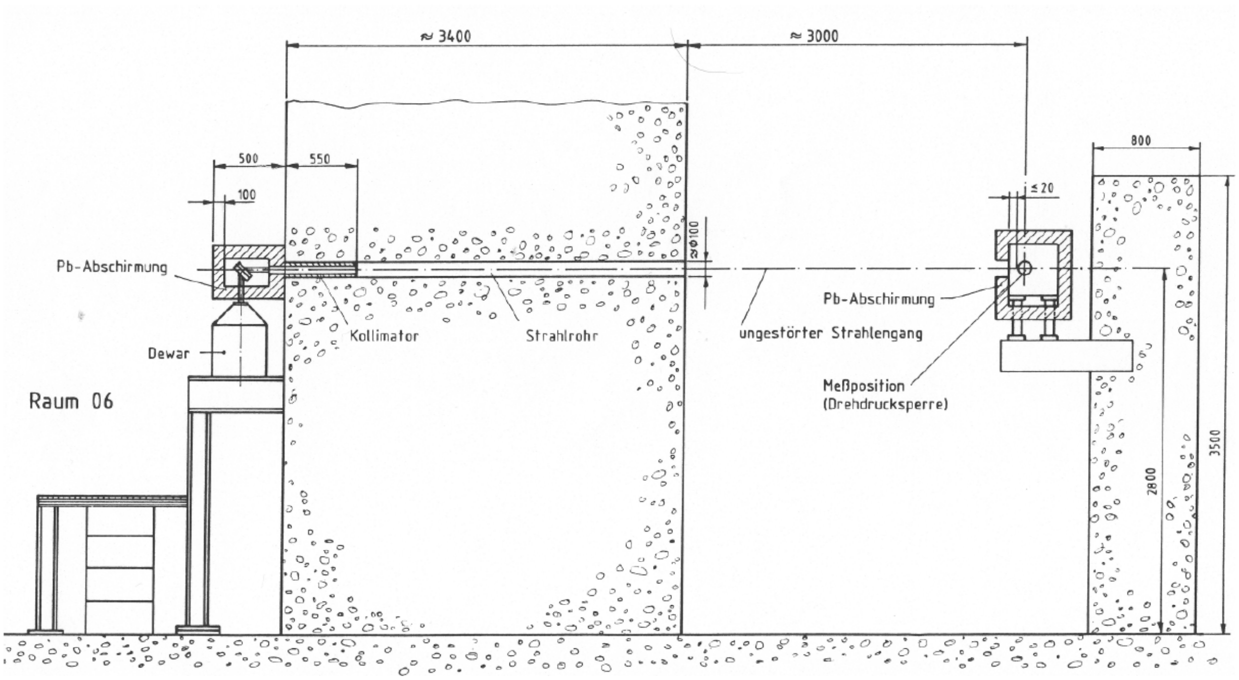
Figure 5-42: Statistical Uncertainties in Burnup Determination



Moreover, performance of the Germanium detectors, computation speed and software optimization, and other parameters linked to the design of the whole system (focal distance, shielding, etc) may have an impact on the results of the measure. Figure 5-43 gives an overview of the Germanium gamma ray detector (left) and spherical fuel element (right) as planned for the HTR-Module. Precision design of source collimator and detector collimator is essential.

All of these parameters should be taken into account in the assessment of the adequacy between required performances and technology capabilities.

Figure 5-43: Proposed HTR-Module BUMS Detector Layout



5.7.6.4 Comparison of Existing Systems

As pointed out above, a multitude of parameters may influence the measure performance, some of which are possibly linked to the reactor design. Therefore, comparison between the systems performance shall be made with tremendous care.

However, Table 5-12 allows comparison between measuring time, statistical error, burnup and decay time, hence giving interesting trends.

Table 5-12: Comparison of PBR BUMS Performance

Item	*AVR*	*PBMR DPP 400*	*HTR-Module*	*HTR-PM*
Decay time	30 days	100 hrs	55 hrs	~30 hrs (very small impact on the accuracy)
Measurement time	30 sec	29 sec	10 sec	10 sec
Relative error	2.5%	5%	5%	5%
Burn-up limit	"highly burnt" pebbles	90,000 MWd/tU	80,000 MWd/tU	80,000 MWd/tU

Taking into account the sensitivity of the measure precision to the burnup and the decay time, this table shows that HTR-Module and HTR-PM designs rely on particularly high performances of their BUMS compared to previous PBR ones.

Some high level considerations about the qualification level of each system are given hereafter:

- AVR system has of course been fully demonstrated during the reactor 21 years of operation.
- PBMR DPP 400 system has been qualified by tests in the Institute of Nuclear Material in Zarechny (Russia). Today, a supplier capable of furnishing a BUMS complying with the above mentioned specifications for the PBMR DPP 400 design exists (Thermo Fisher Scientific Inc.).
- HTR-Module system has been tested in Jülich in the DIDO reactor. The results need to be further analyzed.
- HTR-PM system performances are deduced from numerical simulation. The high performance obtained by the calculation is notably due to the 50.000 -100,000 counts per second performance assumed for the Germanium detector while the state-of-the-art products are closer to 10,000 counts per second [69] .

Except AVR, where BUMS performances were poor, the only BUMS that has reached the commercial stage of development (PBMR) needs about three times mores measurement time and two times more decay time than that of the HTR-Module to reach the precision of 95% required by every design.

Therefore, specific investigation on this subject would be required to explain these differences and to confirm the feasibility of the HTR-Module BUMS performance.

In case the measuring time needed to be longer, several solutions can be envisioned, for instance based on:

- Use of multiple BUMS in parallel. The impact of this kind of solution would have to be measured in terms of complexity, reliability and cost of the fuel discharge equipment.

- Increase of the decay time. The decay occurs during the passage of the pebbles through the exhaust chute (where the fuel is no longer critical). The impact of the exhaust chute design and other impacted systems, together with the quantitative improvement of the measurement should be evaluated in more detail.

- Adjustment of the core recirculation rate. Each pebble passes 15 times through the core before attaining the burnup limit in the HTR-Module design. Lowering this number would relax the specification of the BUMS measuring time but would probably have large implications on the reactor core design (notably in terms of core homogeneity) that would need to be assessed.

For short enough outages of the Fuel Handling System, the reactor can be kept running until the Fuel Handling System resumes operation (see AVR experience). At this moment, the number of pebbles inside the core that have reached the burnup limit is higher than usual. Therefore, operation of the Fuel Handling System with the nominal circulation rate would naturally force the reactor system back to its nominal equilibrium in terms of average burnup and reactivity reserve without increasing the circulation rate, but by natural increase of the proportion of spent pebbles discarded by the BUMS and the associated higher number of new fuel pebbles charged into the reactor. The outcome is finally that some pebbles will have:

- Either spent more time than expected inside the reactor and hence reached a higher burnup (the over limit burnup should be limited for safety reasons, however),

- Or passed through the core a lower number of times than expected before reaching the burnup limit (which is equivalent to a short term decrease of the circulation rate).

5.7.6.5 BUMS Conclusion

The BUMS of the HTR-Module could be the bottleneck for the pebbles circulation rate if the performance of the ^{137}Cs Spectroscopy system is not significantly improved compared to the past experience. Available tests results and information from the manufacturer Thermo Fischer Scientific Inc. suggest that the final qualification of the HTR-Module BUMS probably needs some more confirmatory tests.

In case the dual requirements of 10 s measurement time and 5% statistical error can not be reached, reasonable solutions exist to adjust the design so as to support a sufficient circulation rate. Multiple parallel BUMS seems to be the most practical solution and the less impacting one for the design.

5.7.7 Overall Online Refueling Conclusions

Past PBR Fuel Handling Systems availability experience feedback and performance of the BUMS technology by ^{137}Cs spectroscopy has been examined and assessed regarding the current PBR technology based on the HTR-Module.

It has been concluded that past experience of frequent Fuel Handling System unavailability should not be considered as an intrinsic feature of PBR Fuel Handling Systems and that careful consideration of past experience as well as appropriate qualification test programs should largely support the successful operation of a newly design of pebble Fuel Handling System.

As for the BUMS, particular care should be granted to the qualification tests of the highly demanding design specification of the HTR-Module. However, it has been shown that if the expected performances could not be reached, practical solutions would exist to tackle the difficulty, like the use of multiple BUMS in parallel.

Overall, this analysis has shown that the performance and reliability limits of PBR Fuel Handling Systems would not constitute a potential show stopper for efficient on-line refueling of future designs of PBR, including for process heat applications and associated high availability requirements.

5.8 Tritium

Tritium is not unique to the PBR and is an issue for all HTGRs and LWRs, since it is a product of nuclear fission and neutron capture reactions.

5.8.1 Tritium Issues

Tritium (T or ^{3}H, half-life of 12.5 years) is considered as an important source term because it can permeate into graphite and through metals. The following are specific concerns for Tritium in PBR:

- Tritium permeation through the heat exchanger tube wall causes contamination of steam production and process heat application cycles
- Accumulation of Tritium in Helium purification system for waste removal
- Tritium inventory in spent and scrapped fuel pebbles for disposal

Discharge of Tritium into the environment poses a contamination problem because it can accumulate over a relatively long period of time due to its long half-life. However, because Tritium decay emits only a low-energy beta particle, concern with Tritium emissions arises only if it is ingested.

For example, in a hydrogen production process using a steam reforming process; the product (i.e., Hydrogen gas), to be considered as a normal commodity, must have a Tritium concentration below the allowable limits specified by the regulatory agencies. Therefore, one safety requirement is to reduce the amount of Tritium released into the processing products. Hence, the required safety items are not directly linked to the reactor safety and thus can be classified into the lowest safety level [70] (in Germany these were considered non-safety systems).

5.8.2 Tritium Production in PBR

Tritium is primarily produced by fission in the fuel and neutron capture in the Lithium impurities in the fuel element graphite matrix material, graphite reflectors, and ^{3}He fraction in the circulating primary coolant.

There are at least five Tritium production mechanisms in the PBR governed by the following neutron capture reactions:

1. Ternary fission product [U-235 (n,f) T], with a fission yield of 1.2×10^{-4} (UO_2 fuel)
2. ^{3}He (n, p) T in the helium coolant passing through the reactor core
3. ^{6}Li (n, α) T in the fuel spheres and the graphite reflectors
4. ^{10}B (n, 2α) T in the control rods
5. ^{10}B (n, α) ^{7}Li (n, n α) T in the control rods

Among the five production mechanisms, ternary fission accounts for about 50% of total Tritium production in HTGRs. Activation reactions from the traces of Lithium in graphite and Boron in control rods account for about

35% of the total Tritium production. Activation from the ^3He fraction in the Helium coolant accounts for the remaining 15% of the total Tritium production [71].

Because Tritium behaves chemically as hydrogen, it can undergo isotope exchange reactions with hydrogen-containing chemicals (e.g., H_2O, H_2, or CH_4). Tritium is produced in HTGRs by neutron absorption in boron and lithium in the coolant. Additionally, Tritium generation in the coolant is caused by graphite and hydrogen from coolant impurities of CH_4, H_2O, and H_2 [72].

Even after careful decontamination, the lithium content in nuclear graphite decreases at best to $5x10^{-5}$ %. The activity of Tritium formed in graphite due to the lithium impurity, at a lithium content of 0.1 ppm (by weight) in the nuclear graphite and a thermal neutron fluence of 3.6×10^{9} cm^{-2}, approximately $2x10^4$ Bq Tritium activity forms per gram of graphite. Thus, even when the lithium content of graphite is low, a significant amount of Tritium is accumulated at high neutron fluences [73].

Impurity levels in Tritium-forming materials expected in HTGRs are given in Table 5-13 [70].

Table 5-13: Expected Tritium Forming Impurity Levels in HTGR Reactor

Source	Expected Tritium Impurity Level
Lithium	
Fuel graphite matrix	0.05 ppm[(a)]
Reflector graphite	0.05 ppm[(a)]
Boron	
Control Rod	30% B
Reflector graphite	10.3% B
Helium-3	0.1 - 0.2 ppm[(b)]

(a) ppm by weight; (b) ppm by volume.

5.8.3 Tritium Transport inside Fuel Pebbles

Experimental results indicate that most Tritium produced in a fission process will be retained within the intact TRISO coated fuel particles; only a small fraction originating from fuel particles with a broken coating or from uranium contamination of the matrix graphite is expected to escape into the coolant [70].

The Tritium diffusion coefficient in pyrolytic carbon coating displays an unusually high temperature dependency, which explains why Tritium is retained almost completely at 973°K to 1073°K but is quickly released in post-irradiation annealing between 1373°K and 1573°K [70].

Experimental studies on the TRISO coated particles showed that [70]:

- Tritium release from different kernel types followed the trend $UO_2 > UC_2 > (Th,U)C_2$.
- The degree of Tritium release depended on burnup and temperature.
- TRISO coatings were more effective in retaining Tritium than BISO.

Experimental studies on the BISO coated particles showed that [70]:

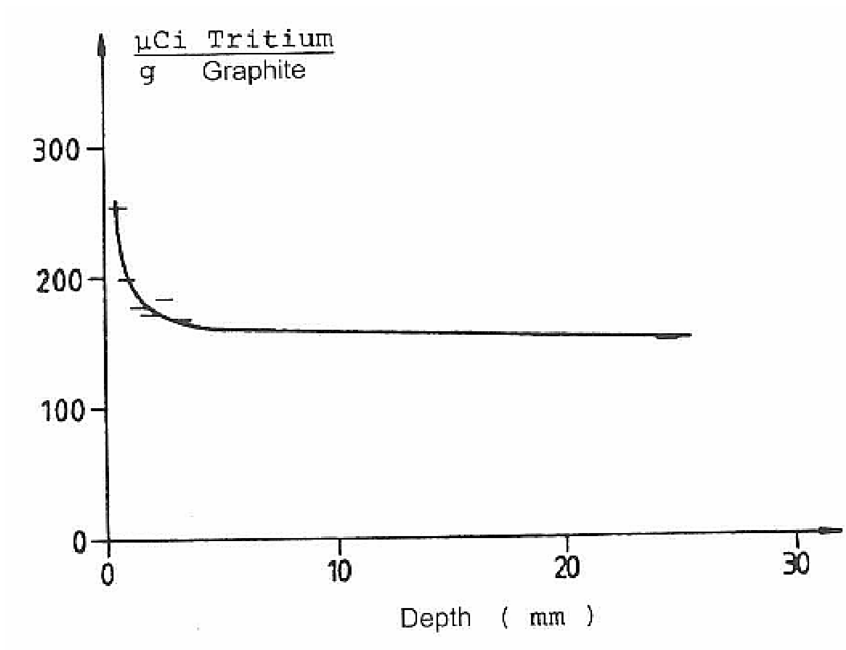
- Tritium retention was $UO_2 > UC_2 > (Th,U)C_2$.
- Tritium retention depended upon temperature and burnup.
- At low burnup (<25% FIMA), the Tritium retention characteristics of $(Th,U)C_2$ particles were very temperature dependent (i.e., 70% retained below 1248°K but only 15% at 1438°K).
- Although the coating was known to be broken on one of the UC_2 particles at 75% FIMA burnup, the Tritium retention was equal to that of a particle having an intact coating.
- At 1023°K and 60% burnup, UO_2 and UC_2 retained >70% of the Tritium; at 1548°K and 75% burnup the UO_2 particles retained 41% Tritium but the UC_2 particles only retained 23%.

Note that the $(Th,U)C_2$ particles were only investigated at low burnup (<25% FIMA); whereas UC_2 and UO_2 particles were tested only at high burnup (>60% FIMA).

In an experiment conducted at a burnup of 17% FIMA and temperatures of up to 1523°K, at least 90% of the Tritium produced was retained [74].

It is known that hydrogen/Tritium can be adsorbed onto graphite at high temperatures. Out-of-pile measurements made on A-3 graphite (fuel matrix graphite) and measurements on the AVR fuel pebble showed that Tritium adsorption increases with both temperature and time. The Tritium activity profile in the AVR fuel pebble is shown in Figure 5-44 [75]. As shown, the Tritium concentration observed in the fuel pebble from the AVR indicates that a much larger Tritium adsorption occurred at the surface of the pebble, originated from to initial failed fuel particles. The Tritium activity profile also shows a strong retention of ternary Tritium fission product by the intact TRISO particles.

Figure 5-44: Tritium Activity Profile of an AVR Fuel Pebble after Six Years of Operation



5.8.4 Tritium Permeability in Heat Exchanger

There are two approaches to reduce the Tritium concentration in the products; one is removing Tritium from the coolant and the other one is protecting against permeation through the heat exchanger tube walls.

The hydrogen/Tritium permeability is high for clean tube surfaces; however, it decreases if an oxide layer covers the surface. Under steam reforming conditions, an oxide layer will rapidly develop on the tube surface. As part of the German program on process heat applications of very high temperature gas-cooled nuclear reactors (VHTR), hydrogen/Tritium permeation rate of Incoloy 800 over the temperature range 873° to 1223°K and between hydrogen partial pressures of 50 and 5 x 10^8 Pa (0.5 to 500 mbar) were measured as shown in Figure 5-45 [70][76]. As shown, hydrogen permeation rate increases with partial pressure and temperature. The same experiment also determined that an oxide layer decreases the permeation rate by more than two orders of magnitude as shown in Figure 5-46 [70][76]. It should be noted that these values are representative of steady-state conditions. During plant transients, it is expected that spalling of the oxide layers will result in greater tritium permeation for some time period before the layers can be re-established.

Figure 5-45: Hydrogen Permeation Rate vs. Partial Pressure and Temperature for Incoloy 800

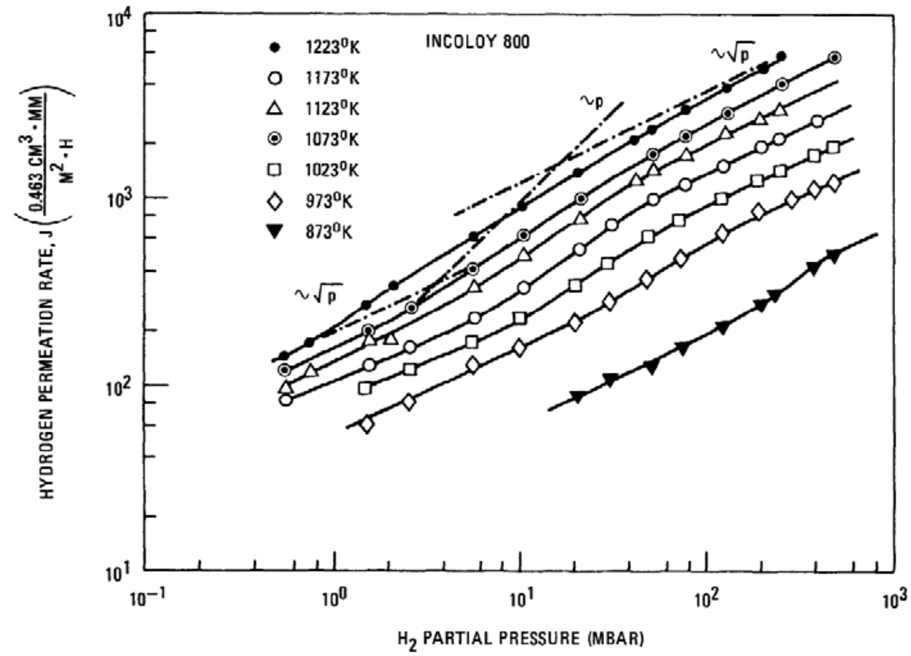
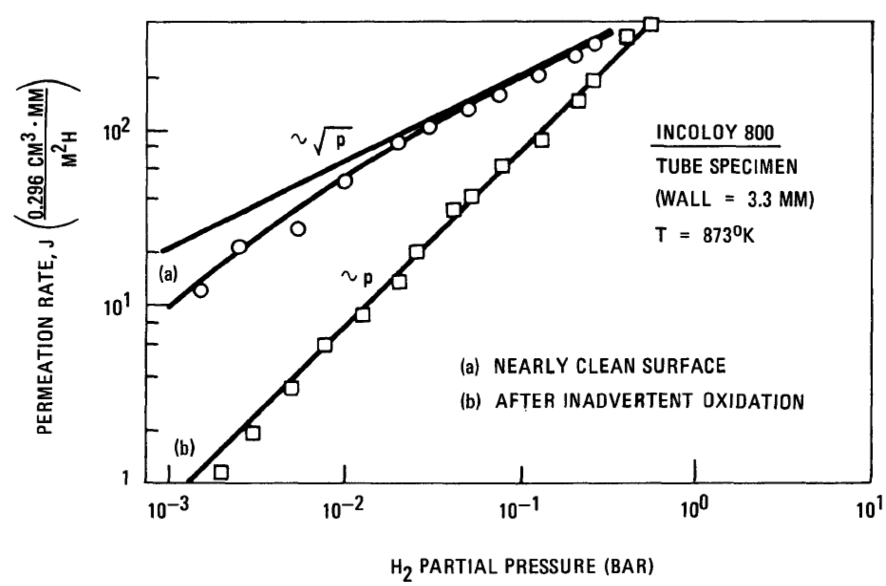


Figure 5-46: Influence of Oxide Film on Hydrogen Permeation Rate into Incoloy 800



5.8.5 Tritium Activity in HTR-Module

In HTR-Module, on account of burnup effects, the power dependent integral Tritium activity generation rate drops from an initial 1.4 x 10^8 Bq/hr-MW to a quasi-equilibrium 5.5 x 10^7 Bq/hr-MW within a few years [3]. For the Tritium produced in the fuel pebbles and graphite reflectors, only a fraction is released to the primary Helium coolant as results of diffusion and adsorption mechanisms. Therefore, the power-dependent Tritium activity release rates are: 2.8 x 10^7 Bq/hr-MW initially, and 1.8 x 10^7 Bq/hr-MW at equilibrium [3].

For licensing reasons, effluent with Tritium via turbine hall was defined in [3] as were tritium values for the process steam. For both paths, the regulator and the Safety commission agreed with the assumed values.

5.8.6 Tritium Release Control

There are at least three Tritium control mechanisms can be taken into design consideration: (1) permeation through steam generator or IHX wall; (2) gas purification system; and (3) adsorption on reflector graphite and graphite matrix in fuel pebbles.

Under nominal operating conditions, an oxide scale with a thickness of a few to tens of micrometers is expected to form on the steam generator tube alloys due to impurities (H_2O, CO, CH_4) in the helium coolant. The experimental results show that oxide layers formed on steam generator tube metal can decrease the hydrogen /Tritium permeation rate by more than two orders of magnitude [76] during steady-state operation. In addition, an indirect steam cycle employing reboilers (as shown in Figure 3-1) introduces a second barrier (i.e., reboiler) into the Tritium release path, providing an effective way of reducing the permeation of Tritium into the process steam.

Moreover, gas purification system can be used to control the impurities in the helium coolant, thus further reducing primary system Tritium concentration.

Finally, experiments and Tritium measurements in AVR fuel pebbles show strong adsorption of Tritium on graphite surfaces and most of Tritium fission products are retained inside the intact TRISO fuel particles. Therefore, loss of Tritium through permeation process will be minimal.

5.8.7 Tritium Assessment Conclusions

The mechanisms by which Tritium is produced in the HTGR are well understood, namely, due to ternary fission and neutron activation of He-3, Li-6, and B-10. Uncertainties in the Tritium production rate in the HTR-Module are mainly associated with an imprecise knowledge of fission yield, and accurately assessing Tritium-forming impurity levels.

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 employing the steam reboilers before subsequent process heat applications such as steam reformers 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.

6.0 READINESS OF SUPPORTING TECHNOLOGY DATABASE

6.1 Design Data Needs Assessment Introduction

This assessment of the design data needs is mainly based on an analysis of the DDNs issued by the NGNP project for the development of the NGNP 750°C, steam cycle version of the PBR [77], excluding DDNs devoted to the hydrogen production process. But these DDNs are based on the needs for a PBMR-based NGNP. If the reference design is the HTR-Module, the needs are not exactly the same due to the different level of maturity of the technology for the two designs and also due to the different options selected for the design. This 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. This is the case more particularly for the steam generator, the design of which, in the case of the HTR-Module, was not only finalized, but justified through many tests, including large scale qualification tests in the KVK helium loop. On the contrary some DDNs that could not be found in the current NGNP list have been added, which correspond to the views of AREVA experts. For adding these DDNs, the analysis was mainly (but not exclusively) based on the work synthesized in an AREVA report [78], as many needs are common to prismatic and pebble bed designs, the differences affecting essentially the core and other internal structures. Moreover the DDNs of the current NGNP PBR concept were written assuming that the results of research programs for the PBMR DPP would be available for NGNP. Now the situation is rather different and the analysis performed here had to take it into account. This was the case, for instance, for graphite DDNs (NHSS-02-01 to NHSS-02-04) which took into account the expected results of the PBMR-Specific Materials Test Reactor Program (PSMP).

The current list classified the DDNs into two categories, enabling and enhancing. The enabling DDNs were the ones “supporting the present PBMR-CG” (“in the present PBMR-CG reference design the steam generator is placed in the primary loop” and this design “provides the reference basis for the NGNP”). The enhancing DDNs are defined in the following way: “In order to allow DDNs to drive the direction of technology development, and recognizing that the longer-term goals of direct heat, higher temperature applications still remain, the DDNs related to configurations and temperatures focused on such applications have been kept in the list, categorized as “Enhancing”” (all quotes in this paragraph are from [77]).

Here the definition of enabling is very similar to the current NGNP one, though referring to a different status of acquired technology: the enabling DDNs are the ones still necessary for constructing the NGNP on the basis of the HTR-Module design, as well as the technologies and knowledge incorporated in this design.

Now, in this assessment, the “enhancing DDNs” of the current NGNP list have been renamed “long term (LT) DDNs,” because an intermediate category was introduced between “enabling” and “LT” in which are grouped possible (but not necessary) improvements of the 20 years old HTR-Module design and technology. In the terminology used here, this category was classified as “enhancing.”

In Table 6-1 all current NGNP DDNs are listed, including the DDNs that AREVA proposes to cancel (with a grey background), as well as new DDNs identified by AREVA. The origin of each DDN is given (either PBMR or AREVA), its reference in the documents referenced above (for some of the needs identified by AREVA, it is mentioned that they are only partly covered by an existing DDN), its title, its nature (enabling, enhancing or long term (LT)), and the assessment that AREVA makes on this need (agreement, reasons for modifying the need as expressed, reason for suppressing a need or reason for adding a need and description of the added need). If the explanations are too long, they are reported into comment sections following the table.

6.2 General Assessment of DDNs of the PBR Steam Cycle NGNP Design

There are 82 DDNs identified (17, in addition, have been suppressed because they are specific to PBMR and either are not relevant for the HTR-Module or have already been addressed). Only 23 of them are enabling, 16 enhancing and 43 for the long term. This distribution shows that the HTR-Module relies on rather mature technologies that cannot expect too much from short term enhancements, but that there is still an important R&D effort to be made before considering industrial applications at higher temperatures.

6.2.1 Enabling DDNs

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 amplitude 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 ASME in the discussions.

A similar question exists for the qualification of the critical components: if they are built following strictly 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 transfers, 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 his 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.

6.2.2 Enhancing DDNs

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 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 of qualifying the 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 allow decreasing significantly 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 for requiring a significant program of tests, including integral tests of the whole component.

6.2.3 Long-Term Technology Development Needs

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. But the main challenge with 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 envisaged 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 exchange and for process will have to be developed as well as technologies for heat transport, which is not a common industrial practice for the time being at temperatures above 550°C. But this is a domain 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.

Table 6-1: PBR DDN Assessment Summary Table

Origin	Reference	Rev.	Title	Status	Assessment
PBMR	DDN NHSS-01-01	B	Fuel Irradiation Test for Normal Operational Conditions	Enabling	See comments in Section 6.3.
PBMR	DDN NHSS-01-02	B	Fuel Heating Tests for Accident Conditions	Enabling	See comments in Section 6.4.
PBMR	DDN NHSS-01-03	C	Fuel Graphite Irradiation Tests	Enabling	See comments in Section 6.5.
New	AREVA DDN 1.1.1.1		Kernel Materials – Advanced Carbon Source Development	Enhancing	For UCO fuel
New	Partly covered by AREVA DDN 1.1.1.2a&b, 1.1.2.2 & 1.1.4.1a		Improving the mastering and the optimization of the fuel fabrication process	Enhancing	Industrialization requires optimization of the production tools, better identification and sensitivity of all parameters that influence the quality of the fuel. Not necessary for the first NGNP core.
New	Covered by AREVA DDN 1.2.1.0 for particle QC		Quality control methods - Fuel QC Inspection Techniques	Enhancing	Modern methods for pebble QC should be developed: - Economic incentive is high - More robust demonstration of first barrier leak tightness
New	Modified from AREVA DDN 1.3.1.0a&b		Fuel Air and Water/ Steam Oxidation	Enhancing	For UCO fuel: comprehensive data exist for UO_2 pebbles that will be applicable at least for the pebble matrix corrosion data and for coating layer data. But as the kernel and the design of the pebble (packing fraction) are different, tests will be needed with actual UCO pebbles.
New	AREVA DDN 1.4.1.0a&b		Data required for storage of spent fuel and irradiated graphite	Enabling	These data are needed for designing the short term storage of spent fuel and irradiated graphite. Additional data will be needed once the US strategy for fuel cycle and final disposal of high-level-long-lived wastes will be defined.
PBMR	DDN NHSS-02-01	B	Extended Properties of Irradiated Graphite at Low Temperatures	Enabling	See comments in Section 6.6.
PBMR	DDN NHSS-02-02	B	Extended Properties of Irradiated Graphite at High Temperatures	LT	See comments in Section 6.7.

<table>
<tr><th>Origin</th><th>Reference</th><th>Rev.</th><th>Title</th><th>Status</th><th colspan="2">Assessment</th></tr>
<tr><td>PBMR</td><td>DDN NHSS-02-03</td><td>D</td><td>Influence of Irradiation Creep on the Properties of Graphite – NGNP Demonstration Plant</td><td>Enabling</td><td colspan="2">Agreed</td></tr>
<tr><td>PBMR</td><td>DDN NHSS-02-04</td><td>B</td><td>Influence of Irradiation Creep on the Properties of Graphite – Advanced NGNP Applications</td><td>LT</td><td colspan="2">Agreed</td></tr>
<tr><td>PBMR</td><td>DDN COMP-01-01</td><td>B</td><td>Characterize Race Track Strap and Tie Rod Materials</td><td>LT</td><td colspan="2">Enabling for PBMR design but not relevant for HTR-Module design: maintaining a stable core geometry is performed by graphite anchors and by metallic internals. At the moderate temperature of the reference design, the use of Alloy 800H is possible for these internals. Advanced materials could be needed for higher temperature operation.</td></tr>
<tr><td>PBMR</td><td>DDN COMP-01-02</td><td>C</td><td>RCS Materials Characterization</td><td>Enabling / LT</td><td colspan="2">See comments in Section 6.8.</td></tr>
<tr><td>PBMR</td><td>DDN COMP-01-03</td><td>C</td><td>Core Outlet Connection or Hot Gas Duct Liner</td><td>LT</td><td colspan="2">Solutions that would not require R&D have been developed in Germany.</td></tr>
<tr><td>PBMR</td><td>DDN COMP-01-04</td><td>B</td><td>Insulation Materials (HGD, IHX)</td><td>LT</td><td colspan="2">Aerogels are commonly used in industry up to 650°C. There are in laboratories aerogels that can withstand temperatures up to 1200°C, but they are not industrialized and their compatibility with very high temperature reactor conditions should be verified. But solutions that would not require R&D have been developed in Germany.</td></tr>
<tr><td>PBMR</td><td>DDN COMP-01-05</td><td>A</td><td>Insulation Materials (Lower Reflector)</td><td>LT</td><td colspan="2">Solutions that would not require R&D have been developed in Germany.</td></tr>
<tr><td>New</td><td>Based on AREVA DDN 3.1.1.0a</td><td>B</td><td>Primary Gas Circulator Impeller Tests</td><td>Enhancing</td><td rowspan="2">For new design with magnetic bearings, tests are needed in full range of operating conditions</td><td>Might be necessary if the impeller design has to be changed. Tests at ambient temperature and pressure in air, at scale at least 0.2 to 0.4</td></tr>
<tr><td>New</td><td>Based on AREVA DDN 3.1.1.0b</td><td></td><td>Primary Gas Circulator Rotating Assembly with Magnetic and Catcher Bearings</td><td>Enhancing</td><td>Complete rotating equipment to be tested in high temperature pressurized He for representative friction and wear conditions.</td></tr>
</table>

Origin	Reference	Rev.	Title	Status	Assessment
New			Electrical Conductors, Insulation and Penetration in Helium	Enhancing	To eliminate the risk of electric discharge and detect leakages from the circulator housing, tests of representative motor and bearing circuits in high temperature pressurized He.
New	Based on AREVA DDN 3.1.1.0c		Primary Gas circulator Shutoff Valve Tests	Enhancing	To check the operation and reliability of the shut-off valve including possible interference with the circulator, tests in air could be acceptable.
New	Based on AREVA DDN 3.1.1.0d		Integrated Full Size Tests	Enhancing	At least in air at a manufacturer facility. Final qualification in helium could be postponed until NGNP commissioning tests.
New	AREVA DDN 3.3.1.0		Helium Purification system - Charcoal Qualification	Enabling	The charcoal for NGNP must be selected and qualified
New	AREVA DDN 3.3.3.0		Fuel Handling System - Material/ Subcomponent Testing	Enabling	Even if the design of the system is kept, the components and their parts have to be qualified as procured for NGNP.
New	AREVA DDN 3.3.4.0a		RCCS - Characterization of the Heat Transfer Characteristics of the Surface Treatments for the Reactor Vessel and the Panel Heat Exchanger	Enabling	Separate effect tests to determine the heat transfer characteristics of the surface treatments for the reactor vessel and the panel heat exchanger. Emissivity data are needed as well
New	AREVA DDN 3.3.4.0b		RCCS - Large Scale Test	Enhancing	Tests for the improved RCCS design

Origin	Reference	Rev.	Title	Status	Assessment
New	Based on AREVA DDN 2.2.4.1a to d		Complementary Data on SA-508 and SA-533 for HTGR Applications	Enabling	There are extensive data on SA-508 and SA-533 properties for nuclear applications. Additional data might be required for HTGR specific needs: ◆ Emissivity, ◆ Corrosion effects in impure He atmosphere, ◆ Complementary data on mechanical properties in accident high temperature conditions The German grades have slight differences with SA-508 and SA-533; thus German databases cannot be used.
New	Partly covered by AREVA DDN 2.2.3.1b, d, f & h.		Qualification of 9Cr1Mo Steel	Enhancing/LT	9Cr1Mo steel can be considered for replacing Alloy 800H for the lower temperature internals (enhancing) or for the vessel for higher operating temperatures (LT). Data needs concern material properties, emissivity, irradiation behavior, weldability (including in large thickness) and corrosion in impure helium environment
PBMR	DDN HTS-01-01	D	Establish Reference Specifications for Alloy 617	LT	Agreed
PBMR	DDN HTS-01-02	D	Thermal/Physical and Mechanical Properties of Alloy 617	LT	Agreed
PBMR	DDN HTS-01-03	D	Welding and As-Welded Properties of Materials of Alloy 617 for Compact Heat Exchangers	LT	See comments in Section 6.9.
PBMR	DDN HTS-01-04	D	Ageing Effects of Alloy 617	LT	Agreed
PBMR	DDN HTS-01-05	D	Environmental Effects of Impure Helium on Alloy 617	LT	Agreed
PBMR	DDN HTS-01-06	D	Influence of Grain Size on Materials Properties on Alloy 617	LT	Agreed - for the very thin plates used in most of the plate designs, a study of the minimum acceptable number of grains in the plate thickness would be desirable.
PBMR	DDN HTS-01-07	C	Establish Reference Specifications for Alloy 230	LT	Agreed
PBMR	DDN HTS-01-08	C	Thermal/Physical and Mechanical Properties of Alloy 230	LT	Agreed

<table>
<tr><th>Origin</th><th>Reference</th><th>Rev.</th><th>Title</th><th>Status</th><th colspan="2">Assessment</th></tr>
<tr><td>PBMR</td><td>DDN HTS-01-09</td><td>C</td><td>Welding and As-Welded Properties of Materials of Alloy 230 for Compact Heat Exchangers</td><td>LT</td><td colspan="2">See comments in Section 6.10.</td></tr>
<tr><td>PBMR</td><td>DDN HTS-01-10</td><td>C</td><td>Aging Effects of Alloy 230</td><td>LT</td><td colspan="2">Agreed</td></tr>
<tr><td>PBMR</td><td>DDN HTS-01-11</td><td>C</td><td>Environmental Effects of Impure Helium on Alloy 230</td><td>LT</td><td colspan="2">Agreed</td></tr>
<tr><td>PBMR</td><td>DDN HTS-01-12</td><td>C</td><td>Influence of Grain Size on Materials Properties on Alloy 230</td><td>LT</td><td colspan="2">Similar comment as for DDN HTS-01-06</td></tr>
<tr><td>PBMR</td><td>DDN HTS-01-13</td><td>D</td><td>Methods for Thermal/Fluid Modeling of Plate-Type Compact Heat Exchangers</td><td>LT</td><td colspan="2" rowspan="4">Agreed but urgency 5 and not 1/2 as stated in PBMR DDN: there is a need only for LT application with a gas-gas IHX for temperatures > 800°C.</td></tr>
<tr><td>PBMR</td><td>DDN HTS-01-14</td><td>D</td><td>Methods for Stress/Strain Modeling of Plate-Type Compact Heat Exchangers</td><td>LT</td></tr>
<tr><td>PBMR</td><td>DDN HTS-01-15</td><td>E</td><td>Criteria for Structural Adequacy of Plate-Type Compact Heat Exchangers at Very High Temperatures</td><td>LT</td></tr>
<tr><td>PBMR</td><td>DDN HTS-01-16</td><td>E</td><td>Methods for Performance Modeling of Plate-Type Compact Heat Exchangers</td><td>LT</td></tr>
<tr><td>PBMR</td><td>DDN HTS-01-17</td><td>D</td><td>IHX Performance Verification</td><td>LT</td><td colspan="2">See comments in Section 6.11.</td></tr>
<tr><td>PBMR</td><td>DDN HTS-01-18</td><td>E</td><td>Data Supporting Materials Code Case</td><td>LT</td><td colspan="2">Similar comment as for DDN HTS-01-13</td></tr>
<tr><td>PBMR</td><td>DDN HTS-01-19</td><td>C</td><td>Data Supporting Design Code Case</td><td>LT</td><td colspan="2">Agreed</td></tr>
<tr><td>PBMR</td><td>DDN HTS-01-20</td><td>C</td><td>Influence of Section Thickness on Materials Properties of Alloy 617</td><td>LT</td><td colspan="2">Agreed</td></tr>
<tr><td>PBMR</td><td>DDN HTS-01-21</td><td>C</td><td>Corrosion Allowances for Alloy 617</td><td>LT</td><td colspan="2">Agreed - no clear difference with needs identified in DDN HTS-01-05</td></tr>
<tr><td>PBMR</td><td>DDN HTS-01-22</td><td>D</td><td>Establish Reference Specifications for Alloy 800H and Hastelloy X</td><td>LT</td><td rowspan="2">See comments in Section 6.12.</td><td>Agreed</td></tr>
<tr><td>PBMR</td><td>DDN HTS-01-23</td><td>D</td><td>Supplemental High Temperature Mechanical Properties of Alloy 800H and Hastelloy X</td><td>LT</td><td>Should also include physical properties (including emissivity) and impact of irradiation on properties of the base material and of welded joints (Alloy 800H not only considered for the IHX, but also for metallic internals)</td></tr>
</table>

Origin	Reference	Rev.	Title	Status	Assessment	
PBMR	DDN HTS-01-24	D	Effects of Joining Techniques on the Properties of Alloy 800H and Hastelloy X	LT		Similar comment as for DDN HTS-01-03
PBMR	DDN HTS-01-25	D	Effects of Aging on the Properties of Alloy 800H and Hastelloy X	LT		Agreed
PBMR	DDN HTS-01-26	D	Effects of Exposure in Impure He on Alloy 800H and Hastelloy X properties	LT		Agreed
PBMR	DDN HTS-01-27	D	Influence of Grain Size on Material Properties of Alloy 800H and Hastelloy X	LT		Similar comment as for DDN HTS-01-06
PBMR	DDN HTS-01-28	D	Influence of Section Thickness on Material Properties of Alloy 800H and Hastelloy X	LT		Agreed
PBMR	DDN HTS-01-29	C	Corrosion Allowances for Alloy 800H and Hastelloy X	LT		Similar comment as for DDN HTS-01-21
PBMR	DDN HTS-01-30	C	Brazing and Diffusion Bonding Processes for Alloy 800H and Hastelloy X	LT		Similar comments as for DDN HTS-01-03
PBMR	DDN HTS-02-01	D	Ceramic/Composite HX: Review Existing Technology	LT	See comments in Section 6.13.	
PBMR	DDN HTS-02-02	C	Ceramic/Composite HX: Materials Property Database	LT		
PBMR	DDN HTS-02-03	C	Ceramic/Composite HX: Design Methods	LT		
PBMR	DDN HTS-02-04	C	Ceramic/Composite HX: Performance Verification	LT		
PBMR	DDN HTS-02-05	C	Ceramic/Composite HX: Manufacturing Technology	LT		
PBMR	DDN HTS-02-06	C	Ceramic/Composite HX: Codes and Standards	LT		
PBMR	DDN HTS-03-01	C	Mixing Chamber Performance Test	LT	Specific to PBMR design.	
PBMR	DDN HTS-04-01	C	High Temperature Ducts and Insulation - Active Cooling	LT	Hot gas duct designs have been qualified in Germany up to 1000°C	

Origin	Reference	Rev.	Title	Status	Assessment
PBMR	DDN HTS-04-02	B	High Temperature Piping And Insulation: Passive Insulation		Not relevant for AREVA reference design
PBMR	DDN PCS-01-01	D	Secondary Side Corrosion Characteristics Alloy 800H & 2¼Cr-1Mo and Weldments	Enabling	The use of 2¼Cr-1Mo is specific to PBMR design: AREVA reference design includes full 800H tubes. The corrosion behavior of this material and of its weldings both in impure primary He atmosphere and secondary side environment in AREVA reference design conditions are mastered.
PBMR	DDN PCS-01-02	D	Helium Environment Effects on 2¼Cr-1Mo	Enabling	
PBMR	DDN PCS-01-03	D	Helium Environment Effects on Alloy 800H	Enabling	
PBMR	DDN PCS-01-04	C	Acoustic Response of Helical Bundle	Enabling	Specific to PBMR design: AREVA reference design has been qualified by representative tests.
PBMR	DDN PCS-01-05	C	Large Helical Coil Fabrication Methods	Enabling	Fabrication methods have been developed and tested by the HTR-Module steam generator manufacturer. Moreover other large helical coil steam generator have been manufactured in Germany (THTR) and in France (Superphenix). Nevertheless a new manufacturer should develop and qualify his own manufacturing methods through representative tests.
PBMR	DDN PCS-01-06	C	Inlet Flow Distribution	Enabling	Specific to PBMR design: the issues addressed in these DDNs have been addressed through calculation and tests in AREVA reference design, including full scale qualification tests in the KVK loop.
PBMR	DDN PCS-01-07	C	Insulation Verification Test	Enabling	
PBMR	DDN PCS-01-08	C	Fretting & Sliding Wear Protection Tests	Enabling	
PBMR	DDN PCS-01-09	D	Tube Wear Protection Device Testing	Enabling	
PBMR	DDN PCS-01-10	C	Shroud Seal Test	Enabling	
PBMR	DDN PCS-01-11	C	Lead-in/Lead-out/Transition/Expansion Loop Mockups	Enabling	Fabrication methods have been developed and tested by the HTR-Module steam generator manufacturer. But a new manufacturer should develop and qualify their own manufacturing methods through representative tests.
PBMR	DDN PCS-01-12	C	Flow Induced Vibration Testing of Helical Bundle	Enabling	Similar comment as for DDN PCS-01-06 to 10
PBMR	DDN PCS-01-13	C	Orifice Qualification Test	Enabling	
PBMR	DDN PCS-01-14	C	Instrumentation Attachment Test	Enabling	

Origin	Reference	Rev.	Title	Status	Assessment
PBMR	DDN PCS-01-15	D	Bi-Metallic Weld Structural Integrity	Enabling	The use of 2¼Cr-1Mo is specific to PBMR design: AREVA reference design includes full 800H tubes. The structural integrity of these welds has been verified.
PBMR	DDN PCS-01-16	C	Helical Bundle and Transition Region Heat Transfer Test	Enabling	Specific to PBMR design: for AREVA reference design, the heat transfer performance of the helical bundle has been addressed through calculation and tests, including full scale qualification tests in the KVK loop.
PBMR	DDN PCS-01-17	C	Tubing Inspection Methods and Equipment	Enabling	Agreed. In addition to the existing experience mentioned by PBMR, the results obtained by AREVA for the inspection of Superphenix helical steam generator should be mentioned as should the experience in Germany with the KVK-SG and IHX inspections.
PBMR	DDN PCS-01-18	C	Review and Re-assemble Existing SG Development Data.	Enabling	Agreed
New	Partly covered by AREVA DDN 3.3.5.0a		Development of Improved Instrumentation	Enhancing	Future reactors should benefit from the progress of sensor technology, in particular, but not only for temperature burnup, and radiation measurement
New	AREVA DDN 3.3.5.0b		Instrumentation - qualification Testing in Helium	Enabling	The instrumentation selected for NGNP - whether industrial standard or advanced - should be qualified.
New	Partly covered by AREVA DDN 3.3.4.0c		Dust issues	Enabling	See comments in Section 6.14.
New	AREVA DDN 4.1.2.1a		Thermal-hydraulics - Modeling of RELAP 5-3D	Enabling	An alternative would be the AREVA code MANTA with the same modeling development needs as for RELAP5
New	AREVA DDN 4.1.2.1b		Thermal-hydraulics - Coupling of CFD Model s to RELAP5-3D	Enabling	An alternative to RELAP5 would be the AREVA code MANTA. A relevant alternative to the CFD codes FLUENT or STAR-CD mentioned in the DDN form would be the THERMIX code which was used for HTR-Module 200 design and which allows much faster convergence than STAR-CD

Origin	Reference	Rev.	Title	Status	Assessment
AREVA	DDN 4.1.2.2		Thermal-Hydraulics – STAR-CD Graphite Oxidation Model Development for Water and Air Ingress	Enabling	An alternative would be to use the German code THERMIX-REACT
AREVA	DDN4.1.3.1a		Fuel – Improvement of the Diffusion and the Coating Corrosion Modeling in ATLAS	Enabling	ATLAS models are state-of-the-art. Nevertheless they might have to be tuned for the presently manufactured fuel. In particular, the fission product diffusion coefficients of coating layers should be measured.
New	AREVA DDN 4.1.3.1b		Fuel - Development of Fuel Hydrolysis Modeling in ATLAS	Enhancing	Specific need for water ingress. A conservative alternate is to assume that once the pebble matrix is oxidized, the fission product content of the fuel is released.
New	AREVA DDN 4.1.4.1a to g		Radio-Contaminant Transport	Enhancing	Codes have been developed in particular in Germany (e.g., FRESCO). But the existing models have large uncertainties, but with the performances aimed at for NGNP based on the HTR-Module, these uncertainties should be acceptable for licensing. If improved performances are required, significant developments will be required for improving the modeling and reducing uncertainties.
New	AREVA DDN 4.1.4.2a		Structural Analysis - Completion of Experimental Databases for Structural Mechanical Codes	Enabling	At least necessary for present graphite grade modeling
New			Complements to Core Physics code qualification	Enhancing	See comments in Section 6.15.
New	AREVA DDN 4.2.3.1a&b		ATLAS Fuel Code Qualification for NGNP Fuel	Enabling	Benchmarking ATLAS with AGR irradiation and heat-up tests.
New	AREVA DDN 4.2.4.1		Radio-contaminant Transport Model Qualification	Enhancing	

Notes:

1. Rows highlighted in gray are not necessary for the HTR-Module reactor design.
2. This assessment is based on the status of existing technology, not necessarily mastered by AREVA.
3. Enhancing = can improve the steam cycle HTGR, but not necessary.
4. Long term = addressing a larger market scope with higher temperature range. In that case, only assessment of key enabling technologies, not related to a specific design.

6.3 DDN NHSS-01-01: Fuel Irradiation Tests for Normal Operational Conditions

PBMR claims equivalence regarding design and manufacturing processes between former German fuel and PBMR-based NGNP fuel.

Based on this equivalence, PBMR considers it legitimate to integrate the German database, which covers a wide range of operating parameters, in the PBMR-based NGNP fuel qualification database. PBMR therefore recommends a limited irradiation testing program to verify this equivalence.

AREVA does not consider this equivalence between former German fuel and PBMR NGNP fuel or any other present fuel to be acceptable:

- Regarding the design, this equivalence is not demonstrable as the characteristics at the origin of the good behavior of the fuel are not identified and clearly listed in the former German program, not documented, and therefore not reproducible.
- Regarding the manufacturing process, as good design characteristics are not identified, manufacturing duplication is not able to assure exact equivalence between German fuel and PBMR-based NGNP.
- The similarity of "direct materials" procured from other sources than originally in the German fuel is questionable, as none of the detailed characteristics of the materials composition and microstructure has been identified to be critical or non critical for the good behavior of the fuel.

Moreover sticking to this equivalence approach prevents from improving any step of the fabrication processes or quality control methods, as the resulting fuel can no more be considered as equivalent to the former German fuel, because none of these elements have been identified as critical for the good behavior of the fuel.

It should be added that "the [German] data were not produced under a quality assurance program that explicitly met the requirements of 10CFR50 Appendix B" [79].

Therefore the German fuel database should not be considered as part or combined with the qualification of present fuel. AREVA recommends a full qualification program of the fuel manufactured for NGNP.

Moreover AREVA recommends the fuel to be UCO, with particles manufactured from the industrialization of the process presently developed by B&W. In such conditions, the AGR program meant at qualifying this fuel is still relevant, with an additional demonstration, to show that the pebble manufacturing does not damage the particles more than the pressing of compacts. This should be done through an irradiation of pebbles with a statistically significant number of particles.

6.4 DDN NHSS-01-02: Fuel Heating Tests for Accident Conditions

The analysis of the approach based on the claim of equivalence regarding design and manufacturing processes between former German fuel and PBMR-based NGNP fuel developed in DDN NHSS-01-01 is also relevant in this DDN.

Nevertheless, if the fuel is, following the recommendation of AREVA, UCO, with particles manufactured from the industrialization of the process presently developed by B&W, the AGR program should satisfy all data needs. No additional heating test should be necessary with pebbles, as long as the irradiation proposed in DDN NHSS-

01-01 shows that the pebble fabrication process does not damage the particles more than the pressing of compacts.

6.5 DDN NHSS-01-03: Fuel Graphite Irradiation Tests

The graphite matrix of the fuel pebbles is a typical example of material that will likely be different from the one used in former German fuel. Therefore the approach based on the claim of equivalence regarding design and manufacturing processes between former German fuel and PBMR-based NGNP should not be applicable in this case. An appropriate grade will have to be selected and qualified, without fully benefiting from the former German fuel qualification program. The DDN for fuel matrix graphite should therefore include:

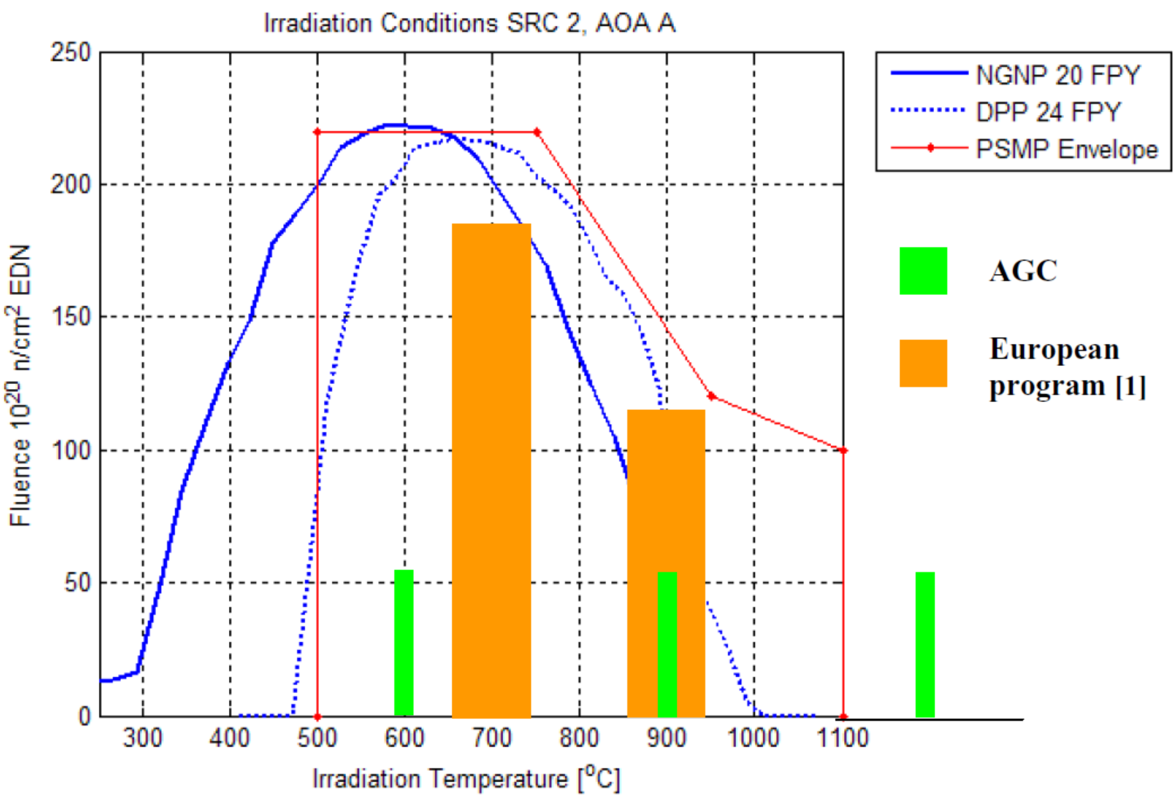
- The identification of an appropriate graphite grade (the most appropriate for fabrication and for irradiation behavior),
- The characterization of the selected grade,
- The qualification of the selected grade (through fabrication of pebbles and their irradiation).

6.6 DDN NHSS-02-01: Extended Properties of Irradiated Graphite at Low Temperatures

The main implicit assumption of this DDN is that the irradiation data from the PBMR-Specific Materials Test Reactor Program (PSMP) will be available and identifies the complementary data required for addressing the PBMR NGNP design data needs.

If it is assumed that the PSMP data will not be available, the only irradiation programs that will have available data with present commercial grades considered for NGNP application are the AGC program and the European program (data accessible through GIF). The domains explored by both programs are shown in Figure 1 (Figure 6-1 here) of the DDN:

Figure 6-1: Irradiation Requirements for NGNP Reflector Graphite



With the two programs, the domain of high temperatures will be reasonably well covered, but additional data will be needed for temperatures between 300 and 600°C.

A full characterization of graphite must be made before irradiation. The data required concern:

- Thermal properties
  - Thermal expansion*
  - Thermal conductivity*
  - Specific heat
  - Emissivity
- Mechanical properties
  - Static and dynamic elastic modulus*
  - Shear modulus
  - Poisson's ratio
  - Strength*
  - Fracture toughness*
  - Multi-axial failure criteria
- Physical characteristics
  - Grain size and distribution*
  - Morphology/anisotropy*
  - Pore size and distribution*
  - Density*

- Fracture properties: assessment of localized surface effects, particularly identification of potential local failure modes and frequencies.
- Fatigue strength

For irradiated graphite, at least the properties identified with * should be measured.

The DDN focuses on the effect of irradiation on graphite properties, but should also address the effect of oxidation (AREVA DDN 2.4.1.0f, g & h), possible cross effect of irradiation and oxidation, interaction between graphite and relevant radionucclides (AREVA DDN 2.4.2.0a & c) and selected grade machineability (AREVA DDN 2.4.3.0).

Reference [80] supports this DDN development.

6.7 DDN NHSS-02-02: Extended Properties of Irradiated Graphite at High Temperatures

The main implicit assumption of the DDN is that the irradiation data from the PBMR-Specific Materials Test Reactor Program (PSMP) will be available and identifies the complementary data required for addressing the PBMR NGNP design data needs.

If it is assumed that the PSMP data will not be available the only irradiation programs that will have available data with present commercial grade that are considered for NGNP application are the AGC program and the European program (the European data being accessible through GIF). The domains explored by both programs are shown of Figure 1 (Figure 6-1 here) of the DDN:

Additional data will be needed for temperatures between 350 and 600°C. It also appears that even above 600°C, there are data needs for fluences higher than those covered by the present experimental irradiation programs.

The data required after irradiation concern:

- Thermal properties
    - Thermal expansion
    - Thermal conductivity
- Mechanical properties
    - Static and dynamic elastic modulus
    - Strength
    - Fracture toughness
- Physical characteristics
    - Grain size and distribution
    - Morphology/anisotropy
    - Pore size and distribution
    - Density

The DDN focuses on the effect of irradiation on graphite properties, but should also address the effect of oxidation (AREVA DDN 2.4.1.0f, g & h), possible cross effect of irradiation and oxidation and interaction between graphite and relevant radionucclides (AREVA DDN 2.4.2.0a & c).

Moreover preliminary examinations performed on the samples of the last European irradiation at 950°C suggest that turn round may have already occurred after 4-7dpa and that the samples have then moved to a phase of fast expansion, indicating that the grades considered until now could be unsuitable for pebble bed application at such high temperature [80]. If these indications are confirmed, new grades should be needed for designing internal structures for high temperature levels.

6.8 DDN COMP-01-02: RCS Materials Characterization

The material for the reference design is Alloy 800H.

ASME code case N-201-5 and Section III subsection NH allow 800H operation up to 760°C. Now during conduction cooldown events, the temperature of control rods reaches about 850°C. The code domain should be extended to allow operation at such temperature. Existing data, as well as the German draft standard 3221 give a sound basis for such an extension [81].

Moreover as mentioned in the DDN, additional data on irradiated high temperature mechanical properties and on corrosion in impure helium atmosphere might be required for Alloy 800H.

Therefore the part of this DDN addressing Alloy 800H is enabling.

The situation for Hastelloy X, which can be a substitute to Alloy 800H, is similar.

On the other hand composites should be necessary for long term applications.

6.9 DDN HTS-01-03: Welding and As-Welded Properties of Materials of Alloy 617 for Compact Heat Exchangers

In the Summary of Data Needed, the need to address long-term evolution of welded specimens (thermal aging and creep) is an important part of the characterization needs, which should be added.

On the other hand, R&D issues depend on the welding process used, which will vary with the selected IHX design. The R&D needs are unilaterally focused on diffusion bonding, which is the solution considered for some designs. Brazing, used in particular in PFHE, a variant of which is considered as the most promising by PBMR [82], should be also studied very carefully: the microstructure of brazing appears to be very complicated, with many phases co-existing, and therefore its behavior in hot conditions should be studied carefully.

6.10 DDN HTS-01-09: Weldability of Alloy 230

In the Summary of Data Needed, the need to address long-term evolution of welded specimens (thermal aging and creep) is an important part of the characterization needs, which should be added.

On the other hand, R&D issues depend on the welding process used, which will vary with the selected IHX design. The R&D needs are unilaterally focused on diffusion bonding, which is the solution considered for some designs. It should be noted that the welding of Alloy 230 is more difficult than the welding of Alloy 617, even for conventional processes; therefore attention should be paid to the development of the processes required for the selected IHX concept, whatever these are.

6.11 DDN HTS-01-17: IHX Performance Verification

In the frame of the present NGNP specifications (steam supply for cogeneration), the same remark as for other IHX related DDNs applies to this DDN: the urgency 2 is no more relevant and therefore it should be downgraded to 5.

Moreover in this DDN the final experimental results for full validation of the design in terms of performance, life prediction, durability and acceptability of fabricated materials should be obtained in the second half of FY2011.

Such results cannot be obtained, but from a thermo-hydraulic and endurance test of a full scale IHX module in a representative impure helium atmosphere (representative temperature, pressure, flow conditions and impurity content). The conditions for the test would require operating a large scale helium loop (at least 10 MW). The DDN form, dated from 10/15/2009, implicitly suggests that designing and building the helium loop, manufacturing the IHX module mock-up and performing the tests could be achieved in about 2 years. Even if the high funding required for the construction of the loop is flowing smoothly when needed, this seems quite unrealistic.

In order to be able to address the needs in a more realistic way, they should be described and scheduled with more details. The data required during the preliminary design phase should be not be the same as those that will be needed during the final design phase: answers to the key feasibility issues should be obtained as soon as possible, likely at the beginning of the preliminary design, while the final qualification of the IHX might wait until the procurement of the IHX, when the final design is sufficiently mature, with maybe some intermediate data supply required for advancing the design.

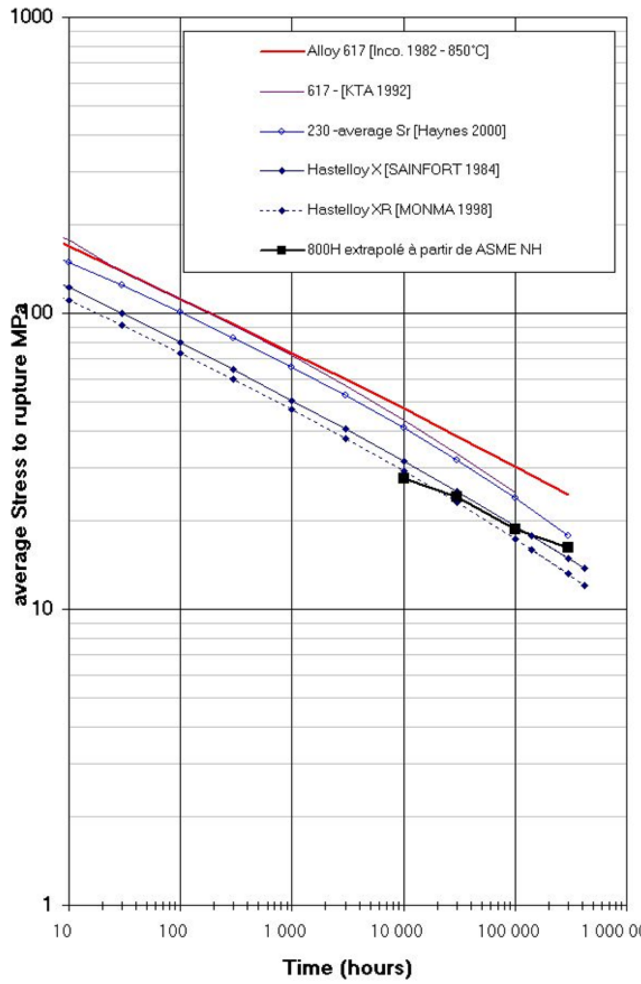
Such progressive data requirements gradually reducing design risks could match with a step by step experimental approach: the first results could be obtained relatively early from separate effect tests performed on simple and inexpensive facilities, most of them already existing, but the final qualification would be enabled only by the development of a dedicated large high temperature helium loop, with, as a possible intermediate step, a first integral validation of the design obtained from the test of an IHX module mock-up with a reduced number of full size plates in a medium scale helium loop (~ 1 MW). Such a loop could be erected in a much shorter delay than the large one, at a much smaller cost.

6.12 DDN HTS-01-22 to HTS-01-30: Alloy 800H/Hastelloy X IHX

For metallic materials, though R&D needs are generally correctly identified, their potential for IHX applications is generally overestimated, most particularly for plate IHX. It is difficult to make a precise statement without having selected a specific IHX design and defined its duty cycles, for which the stresses can be evaluated, but, by assessing different IHX designs, AREVA acquired the experience of the possible stress values that can be expected in the IHX due to high thermal gradients expected in this component in steady state and transient conditions.

Now, in the range of temperature of interest for HTGR applications, from 700°C to 950°C, there is a dramatic decrease of the allowable stress for all the materials considered for IHX applications, about a factor 7, but Alloy 800H remains in all this domain about 2 times lower than Alloy 617 and Haynes 230. Concerning the long term behavior, it can be seen in Figure 6-2 that, at 850°C, the lifetime expected with these two Nickel base alloys is 10 times longer than with alloy 800H. At a given stress level, the IHX lifetime with Hastelloy X will be of the same order of magnitude than with Alloy 800H. The studies made in AREVA have shown that with an appropriate plate IHX concept, the maximum stress in normal operating conditions could be kept in the range 20 to 40 MPa in the hottest parts, if the design of the IHX is carefully optimized. And even with such an optimized configuration, it cannot be expected that the demonstrated lifetime of an IHX in Alloy 617 operated at 850°C could exceed 10 years. If the plate IHX is in Alloy 800H, it is very unlikely to be able to operate it in HTGR conditions at 850°C for a reasonable period of time.

Figure 6-2: Average Stress to Rupture of Different High Temperature Alloys



- Alloy 617, Haynes 230: manufacturer data
- SAINFORT 1984/ "Mechanical characterization of metallic materials for high temperature gas cooled reactors in air and in helium environment," Nuclear Technology Vol. 66, July 1984.
- MONMA 1998: "Creep and stress to rupture – long term," Superalloys, supercomposites, superceramics, Academic Press, Inc.

For corrosion in the impure helium atmosphere of an HTGR, the situation is similar for Alloy 800H: its internal corrosion rate is two times higher than for Alloy 617 and Haynes 230, which will affect its mechanical strength above 800°C even faster than for non-oxidized material. On the contrary the corrosion rate of Hastelloy X is very low, with practically no internal corrosion.

Therefore, taking into consideration their corrosion behavior as well as their mechanical properties, the upper limit of temperature that AREVA would recommend for plate IHX application would significantly differ from PBMR views. Table 6-2 provides an order of magnitude of this upper limit a well as the positioning of the different alloys in terms of mechanical behavior and corrosion, in comparison to Alloy 617.

Table 6-2: Alloy Limiting Factors

	PBMR	AREVA	Limiting factor	
			Mechanical behavior	Corrosion
Alloy 800H	850	750-800	-	-
Hastelloy X	850	800	-	+
Alloy 617	950	850	0	0

Therefore, as the range of temperature identified by PBMR for possible IHX application of Alloy 800H and Hastelloy X should be narrower than presented in the DDNs, even if the R&D needs are in principle well identified, their scope should be more limited than assessed in these DDNs, and most of the data required should be already available, though ASME Section NH do not authorize Alloy 800H above 760°C.

6.13 DDN HTS-02-01 to HTS-02-06: Ceramic/Composite IHX

The technology of ceramic/composite heat exchangers is already used in non-nuclear industry for high temperature and corrosive atmosphere conditions. A quick review has not shown any tube product with a power about 8 MW and much less for plate heat exchangers. For the application proposed by PBMR in its NGNP preconceptual design study [77], the power of the high temperature IHX should be about 160 MW. Therefore there is a large scale difference between the existing industrial products and the needs for NGNP. The factors presently limiting the size of this kind of heat exchangers, which could be technological issues or the absence of market, should be understood.

Moreover, as there is no nuclear experience of ceramic/composite HX, the issues related to this application have not been identified yet.

Therefore the TRL for the kind of component needed for very high temperature nuclear applications should be quite low, 1 or 2. The development of a high temperature ceramic/composite IHX is likely not to be limited to the DDNs listed in the PBMR document. As there are many different concepts implemented for this kind of heat exchangers, at least one scalable concept should be selected and the design of the IHX should be developed. An appropriate material should be selected for the design and operating conditions considered and tested in representative forms. It is in relation with this design approach that relevant DDN should be defined.

The technological risks related to this development are rather high and ceramic/composite heat exchangers for very high temperature nuclear applications cannot be considered, but for very long term applications. This kind of heat exchanger cannot therefore be considered as an alternative to a nickel base alloy one, but a further enhancement of the technology.

6.14 New DDN Need: Dust Issues

A significant quantity of dust is produced in a pebble bed. Several issues could be raised by this presence of dust in the reactor. The answers to be given to these issues rely mainly on design features of the HTR-Module 200, selected for minimizing possible dust releases in the reactor building, maybe on additional design analyses that will have to be performed and on the use of experimental data coming from the German R&D program, that were obtained mostly from normal operation and tests of AVR. The available data will have to be carefully analyzed. As a result of this analysis, possible additional data will be required on the following topics:

- Dust generation: need of experiments with representative materials and representative operating conditions (including chemical environment) to be able to assess the rate of dust production and identify possible influence of operating conditions on this rate.
- Dust plateout tests in representative operating conditions (including chemical environment), varying these conditions in order to see how they can impact the physical, mechanical and chemical properties of the dust layers and most particularly the strength of its binding on the plateout surface.
- Influence of dust dispersed in the reactor building atmosphere in concentrations representative of accident conditions on radiative heat transfer between the reactor vessel and the RCCS (see AREVA DDN 3.3.4.0c).
- Influence of dust deposited on radiative heat transfer surfaces on the emissivity of these surfaces (see AREVA DDN 3.3.4.0c).

In addition, the existence of dead zones in the primary circuit, where loose dust piles can deposit and be easily remobilized in case of sudden change in the flow pattern will have to be avoided as much as possible. This effort of minimizing dead zone will be part of the R&D process: the possible dead zones will be identified through CFD analyses. But an experimental validation of their results will possibly be required with hydraulic tests to be performed in regions with complex geometries, depending on the design.

6.15 New DDN Need: Complements to Core Physics Code Qualification

The methods used for HTR-Module core design are sufficient for NGNP with similar core performance targets. If higher margins are required for enhanced performances, several tracks can be considered for reducing uncertainties:

- Reactivity calculations with codes able to calculate pebble bed cores (for instance VSOP) are qualified at least up to a fuel enrichment of 17% (used for the first criticality of HTR-10, which was the object of an international benchmark. Therefore there is no need of additional critical experiment in a zero power reactor.
- For going to higher burn-up than the target burn-up of HTR-Module (80 GWd/tHM), data on isotopic composition of fuel irradiated to higher burn-up are required. Some data have been acquired of a pebble irradiated to 110 GWd/tHM in HFR (irradiation HFR EU1bis). Additional data would be required at higher burn-up (for instance isotopic analysis of compacts from AGR irradiations).
- The calculation of the temperature reactivity coefficient of graphite at high temperature, which is critical for safety, can be validated from the data that should be obtained in the ASTRA zero-power reactor of the Kurchatov Institute.
- For improving the qualification of the coupled reactor physics-thermal calculations, a melt-wire experiment in HTR-10 would be desirable.
- Experimental data on decay heat should be obtained for improving the margins in the core conduction cool-down.

7.0 PBR FUEL SUPPLY READINESS

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.

7.1 Fuel Selection and Qualification

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.

<u>Option 1- US AGR Program:</u> 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 an accident conditions heating tests. Based on initial indications, there is no reason to believe that anything is wrong with US-made UCO particles or that performance requirements will not be met. However, it must be remembered that important performance aspects have not yet been demonstrated. These objectives will not be fully demonstrated until later phases of the AGR fuel development program are completed. It is also 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.

<u>Option 2 – German Fuel:</u> 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 South African and Chinese fuel programs.

<u>Option 3 – French Program:</u> CEA Cadarache has made UO_2 TRISO fuel and AREVA-CERCA in Romans has made compacts that are presently being irradiated in AGR-2. At present, work on the Cadarache setup has been discontinued, so this is perhaps not a viable option, but could be explored in greater detail.

<u>Option 4 – South African/PBMR Program:</u> As part of a well funded development program, the Fuel Development Laboratory in Pelindaba, South Africa, has developed high quality UO_2 TRISO fuel and made spherical fuel elements fulfilling or exceeding German manufacturing specifications. Particles were put into compacts by ORNL and these are being irradiated now in AGR-2. Sixteen spherical fuel elements containing enriched UO_2 TRISO particles have been shipped to Zarechny in Russia for an extensive set of irradiation tests. These, however, were never started due to PBMR breakdown. It is assumed that, in Pelindaba, all the equipment is still there, though the fuel manufacturing experts have moved on.

<u>Option 5 – Chinese Program:</u> China makes good high quality UO_2 TRISO particles and spherical fuel elements completely to German standards and mostly with German equipment. The recent Petten irradiation test of Chinese spherical fuel elements in HFR-EU1 showed a factor of 3 lower gas release rates than equivalent German spheres, similar to the R/B values measured for the fuel in AGR-1. No PIE has been performed yet on these, nor has any accident condition heating test been conducted so far. However, the Chinese have placed large PIE contracts with Petten and ITU Karlsruhe to do all this as part of their HTR-PM fuel qualification program.

Though each of these options represent 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.

7.1.1 AGR Fuel 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 (~15%) UCO kernel, 425 µm in diameter, 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 based on the anticipated needs of the prismatic HTGR design. This fuel configuration is assumed to be more limiting than the PBR design, as such, qualification results are anticipated to be bounding and applicable for either of the HTGR designs.

For the pebble bed version of a NGNP, it is assumed that the coated particles will be over-coated with a graphitic powder and binders mixed with additional graphitic powder and binders, and then molded into a 50-mm-diameter sphere. An additional 5-mm fuel free zone layer is added to the sphere before isostatic pressing, machining, carbonization, and heat-treating. Under the current AGR program test plan, there is no intent to fabricate or irradiate a complete PBR fuel pebble. All testing is carried-out utilizing prismatic reactor-style fuel compacts. This is primarily the result of limitations in test rig size imposed by the ATR facility, which hosts the AGR program irradiations.

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 the size limitations discussed above, 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. Four pebbles can be irradiated at the same time in this test rig, in 2 independent capsules, each of them with on line monitoring of fission gas releases. To get sufficient statistics in a given period of time, the possibility of getting a second rig in parallel should be discussed with the operator of the reactor, NRG, who has to make the arbitration between experimental needs and production of medical radio-isotopes. PIE can be done in NRG hot cells and heat-up tests in ITU (Karlsruhe). Nevertheless the reactor is old, subject to technical and public opinion hazards. The only alternative that appears to be applicable would be in Russia, where some Chinese test pebbles have been irradiated.

7.1.2 Selection of PBR Fuel Type and Design

Given the above discussion, it is clear that a fuel development strategy based on US regulatory and market expectations must be pursued. For the NGNP Project, it makes the most sense to do this in the context of the AGR program. However, the question of the specific fuel form to be qualified still needs to be settled.

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.

As part of the PBR fuel selection process, an exercise was conducted to determine if the AGR UCO particle design was still appropriate for use as the basis for qualification or if a different basic particle should be selected and implemented in ongoing AGR tests. This exercise applied a weighted scoring process to identify and analyze issues surrounding various conceptual fuel options, and to achieve an unbiased decision on the best fuel for the concept PBR. The exercise leveraged the knowledge and judgment of the AREVA fuel assessment team to identify various fuel selection considerations, weighed each consideration for the conceptual fuel options, and chose the fuel with the highest weighted score as the “best” concept fuel for the PBR.

During the exercise to determine the fuel form for the PBR reactor, 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. The specific particle options considered are listed in Table 7-1.

A number of factors were considered in this evaluation process. The fuel selection criteria used in this analysis, and the associated weighting factors, are presented in Table 7-2. The factors (fuel selection criteria) considered were given a score from 1 to 10, based on the relative importance of each selection criteria as perceived by the fuel assessment team. The assessment team analyzed and rated each fuel form on a scale of -3 to +3 based on how each fuel was perceived to support the selection criteria. A weighted score for each fuel form was obtained by taking the sum of the multiple of the average rating and score for each selection criteria (Table 7-3).

Table 7-1: Fuel Options Considered

Option	Fuel Kernel	Particle	Description
A	UO_2	"German-Like" HTR-Module particle	Attempt to exactly replicate the German experience in the US, relies on historical processes and data.
B	UO_2	PBMR particle	Build on German experience, but take full advantage of more recent PBMR fuel development work to license the product for US..
C	UCO	German particle – low enrichment (~8 w/o)	Change kernel composition, but keep low enrichment so that historical particle geometry can be maintained.
D	UCO	AGR particle – low enrichment (~8 w/o)	Adopt current AGR particle geometry and kernel composition, but use historical HTR-Module enrichment and burnup target.
E	UCO	AGR particle – high enrichment (~14 w/o)	Adopt current AGR particle geometry and kernel composition. Take advantage of higher enrichment and burnup target consistent with current AGR qualification effort.
F	UO_2	Optimized particles	Taking into account global UO_2 particle experience, develop a new, optimized particle design based specifically on the performance envelope of the NGNP PBR concept.

Table 7-2: Fuel Selection Criteria and Relative Importance

Selection Criteria	Relative Importance
Safety Performance	10
Normal Operating Performance	8
Applicability of the AGR Development Program	5
Required Development Impact	5
Cost and Economics	7
Eliminate Dual Development Path	6
Potential for Future Enhancement	6
Use of Historical Data	4

Table 7-3: Fuel Option Scoring

Option	Fuel Kernel	Particle	Final Score
A	UO_2	"German-Like" HTR-Module particle	-39.5
B	UO_2	PBMR particle	-32
C	UCO	German particle – low enrichment (~8 w/o)	9
D	UCO	AGR particle – low enrichment (~8 w/o)	69.25
E	UCO	AGR particle – high enrichment (~14 w/o)	76.25
F	UO_2	Optimized particles	14.6

Using the methodology described in above, the UCO high enrichment (~14% ^{235}U) AGR particle (Option E) was determined to have the highest weighted score of 76.25 and thus selected as the best fuel for the concept PBR. The reasons for selecting the high enriched uranium UCO fuel are summarized as follows:

- UCO minimizes particle internal pressure build-up that results from the accumulation of gaseous fission products. This provides less particle failure and less fission product release.
- UCO provides favorable operating performance and significantly 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.

7.1.3 Fuel and Cycle Design Selection

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.

Fortunately, the pebble bed core is very simple, and the basic parameters of the fuel element design are already established. For this study, the basic dimensions of the fuel element are maintained from the reference HTR-Module. The pebble has an inner fueled region 5 cm in diameter with a 5 mm unfueled outer shell for a total diameter of 6 cm. 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. It is important to take into account the special requirements for initial core pebbles, since a large number of pebbles are required for the first core of each reactor and this will place a large short-term demand on the fuel production facility for each new reactor to be started up. The pebbles used for the first core will typically have a significantly lower loading than new pebbles for the equilibrium core. This is expected to reduce the challenge of producing the first core, since reduced fuel loading will result in fewer particles per pebble, which will significantly reduce per-pebble costs.

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 is not possible in the limited scope of the current PBR assessment. Instead, an interim fuel loading design has been 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.

To estimate the required loading for an equilibrium pebble, the primary factors considered were total mass of heavy metal per pebble and total mass of ^{235}U per pebble (both at beginning-of-life). The reference HTR-Module UO_2 core design has 7 g of heavy metal per pebble and approximately 0.5 g ^{235}U per pebble (for equilibrium core new pebble). This requires about 11,600 kernels (500 μm) 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 7 g 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 7 g 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 7 g heavy metal per pebble, the annual replacement pebble requirement is 150,000 new pebbles per year for a two reactor HTR-Module plant.

This is clearly a manageable number of replacement pebbles to process, both for the fuel handling system and for the fuel production facility. The number of new pebbles processed in the reference UO_2 HTR-Module concept is considerably higher due to the lower burnup.

It is important to note that the fuel acquisition strategy is not particularly sensitive to the uncertainty in the assumed heavy metal loading per pebble. The largest costs for production of pebble bed fuel are fuel particle production and uranium feedstock. The cost of the pebble production step is a minor fraction of the cost. The amount of uranium and hence the number of particles required on an annual basis is strictly a function of the target burnup. If the final design had less than 7 g of heavy metal per pebble, the number of particles would not change significantly. There would just be more pebbles, each with fewer particles. Thus, as stated previously, the 7 g per pebble case is bounding for the fuel qualification strategy, and it has only a minor effect on the fuel acquisition strategy.

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 0.27 g ^{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 3.4 g. 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.

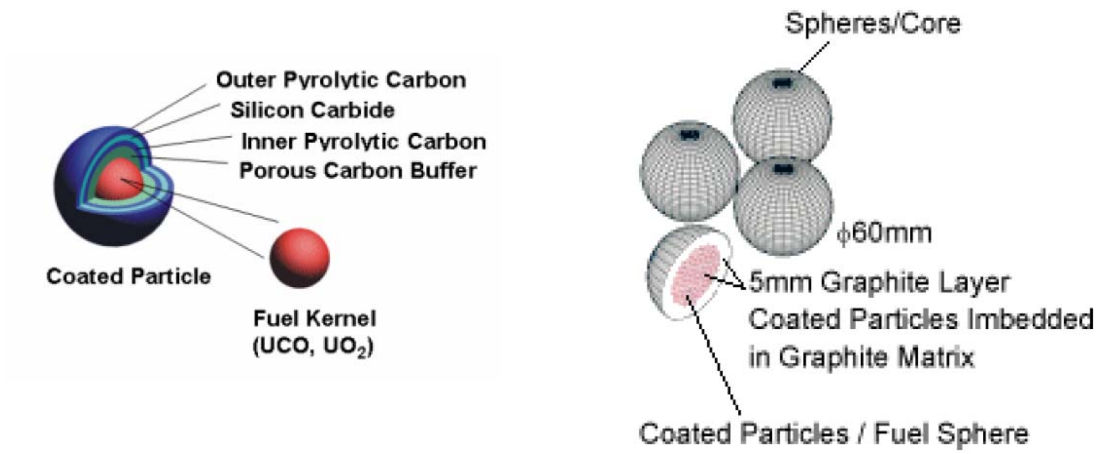
7.2 Fuel Acquisition Strategy

The intent of this section is to document a plan for establishing and operating coated-particle fuel manufacturing facilities in support of the Pebble Bed Reactor and for a follow-on fleet of commercial HTGR's utilizing the Pebble Bed technology. Included in this plan is the estimated cost required in establishing and operating facilities to fabricate pebbles for an initial PBR (Plant 1) as well as pebbles for up to 10 commercial PBRs. Scrap re-cycle, estimated throughputs, projected Uranium losses and staffing plans are described. Finally, a deployment schedule and necessary steps for licensing approval are outlined.

7.2.1 Introduction

The focus of this task is to document a plan 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. The PBR utilizes spherical kernels consisting of a combination of UO_2, UC, and/or UC_2 encapsulated with layers of carbon and silicon carbide (hence referred to as TRISO coated fuel). The TRISO coated fuel kernels are surrounded in a carbonaceous matrix forming a pebble. The pebble is encased with the same carbonaceous mix thus providing a fuel free zone. The pebbles, once finished, are loaded into the PBR. For a single HTR-Module, approximately 720,000 pebbles are required. Uranium loading for each Initial Core (IC) is estimated at 2.45 MT low-enriched Uranium (3.4 grams U/pebble). Each re-load is estimated at 150,000 pebbles containing 1.05 MT low-enriched Uranium (7 grams U/pebble). Specific enrichments for the IC and re-loads are set at 8% and 14% respectively. Enrichment values, unless they are expected to exceed 19.75% U-235, will not have an impact on the fuel acquisition plan. Figure 7-1 identifies the TRISO coated particles and the configuration of a typical pebble.

Figure 7-1: TRISO Coated Fuel and HTGR Pebble



For the initial HTGR utilizing PBR technology, one IC and 11 re-loads are scheduled over a 13 year period. This equates to approximately 14 MT Uranium (2.4 x 10^6 pebbles). Commercial deployment up to 10 HTGRs (approximately 79 MT U, 14.6 x 10^6 pebbles) is possible over the same 13 year period. The proposed fuel delivery schedule is identified in Table 7-4. While this schedule outlines only 13 years of re-fueling (well short of the 60 year life design of a reactor), all cost estimations utilize a 20 year capital depreciation and assumes that fuel fabrication efforts occur past the scheduled 13 year period.

Table 7-4: Fuel Delivery Schedule

Fuel Delivery Schedule to Plant Sites- AREVA Pebble Bed HTR
1 HTR-MODULE Plant = 2 x 200 MWt reactor units

Total MWt			400						800		1600	
Years	-1		0		1		2		3		4	
# of New Plants Starting			1						1		2	
Fuel delivery date	2020		2021		2022		2023		2024		2025	
Plant	IC(k)	R (k)	IC(k)	R (k)	IC(k)	R (k)	IC(k)	R (k)	IC(k)	R (k)	IC(k)	R (k)
1	720					150		150		150		150
2							720					150
3									720			
4									720			
5											720	
6											720	
7												
8												
9												
10												
Total HTR-M Plants	0		1		1		1		2		4	
# of IC pebbles (k)	720		0		0		720		1,440		1,440	
# of R pebbles (k)		0		0		150		150		150		300
Uranium (metric tons)												
IC (~8%)	2.45		0.00		0.00		2.45		4.90		4.90	
R (14%)		0.00		0.00		1.05		1.05		1.05		2.10
Total Uranium (metric tons)	2.45		0.00		1.05		3.50		5.95		7.00	

Total MWt	2400		3200		4000									
Years	5		6		7		8		9		10		11	
# of New Plants Starting	2		2		2									
Fuel delivery date	2026		2027		2028		2029		2030		2031		2032	
Plant	IC(k)	R (k)	IC(k)	R (k)	IC(k)	R (k)	IC(k)	R (k)	IC(k)	R (k)	IC(k)	R (k)	IC(k)	R (k)
1		150		150		150		150		150		150		150
2		150		150		150		150		150		150		150
3		150		150		150		150		150		150		150
4		150		150		150		150		150		150		150
5				150		150		150		150		150		150
6				150		150		150		150		150		150
7	720					150		150		150		150		150
8	720					150		150		150		150		150
9			720					150		150		150		150
10			720					150		150		150		150
Total HTR-M Plants	6		8		10		10		10		10		10	
# of IC pebbles (k)	1,440		1,440		0		0		0		0		0	
# of R pebbles (k)		600		900		1,200		1,500		1,500		1,500		1,500
Uranium (metric tons)														
IC (~8%)	4.90		4.90		0.00		0.00		0.00		0.00		0.00	
R (14%)		4.20		6.30		8.40		10.50		10.50		10.50		10.50
Total Uranium (metric tons)	9.10		11.20		8.40		10.50		10.50		10.50		10.50	

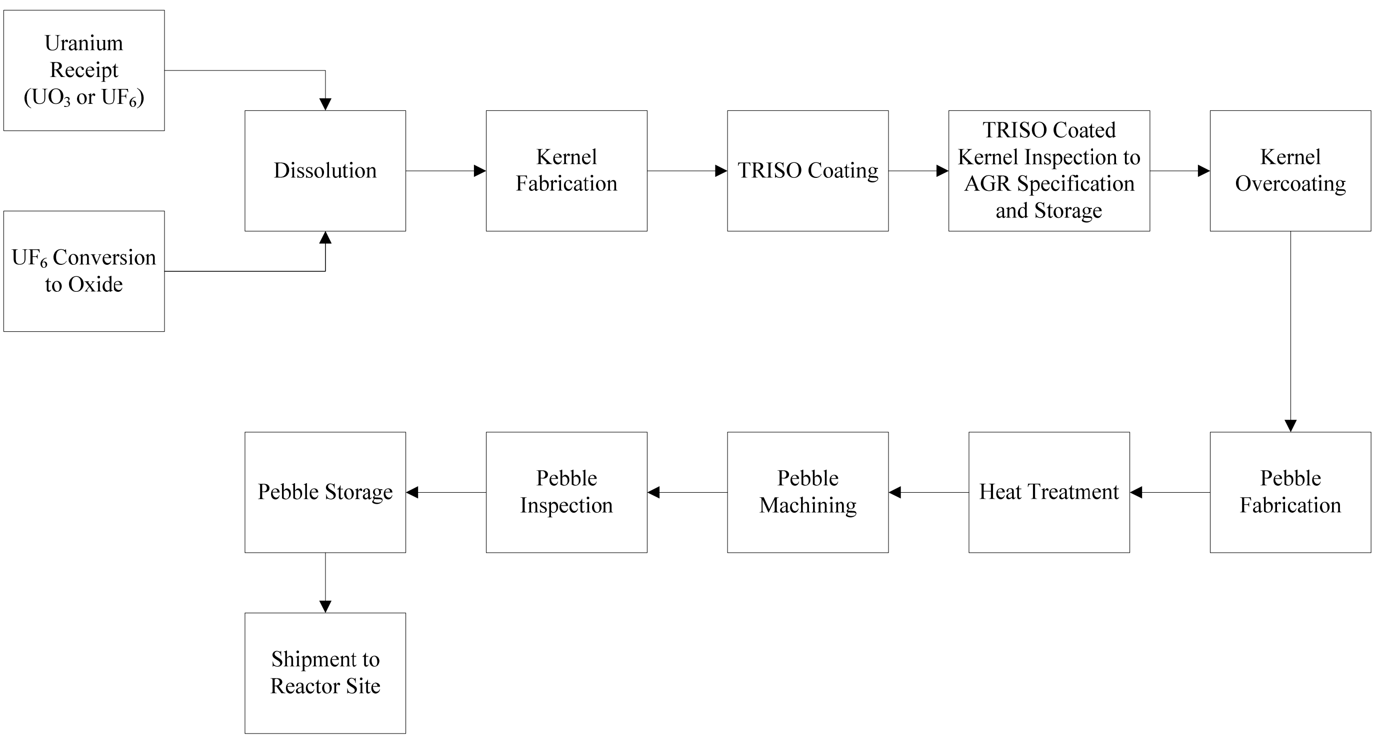
IC = Initial Core — Initial core fuel elements for two units (~8%)
R = Refuel — Refueling elements - 75,000 elements per year/unit (14%)

# of R pebbles /yr/Rx (k) — 75 - Conservative yearly refuel pebbles after year 1 of operation
# of IC pebbles/Rx (k) — 360 - First core

7.2.2 Process Overview

The baseline compact fabrication process, as outlined in Figure 7-2, uses elements from the combined AGR development of B&W NOG-L and Idaho National Labs over the past 10 years. The process consists of the following main units: Kernel Fuel Fabrication, TRISO-Coating of kernel fuel, over-coating of TRISO particles with the resin/graphite matrix, pebble fabrication, heat treatment to stabilize the pebble, machining of the pebble and QC inspection. Scrap from each process is re-cycled through a Uranium extraction process and fed back into kernel fabrication. Shipping containers, following loading with pebbles, are stored until shipment to the reactor site.

Figure 7-2: Process Flow for Initial HTGR and Commercial HTGR Pebble Fuel Fabrication



7.2.3 Design Bases and Assumptions for HTGR 1 Plant Fuel Fabrication

The following is assumed as part of the cost preparation for HTGR 1 Pebble Bed Fuel Fabrication.

- The HTGR Initial Core (Plant 1) and 11 re-loads are deliverable over a 13 year period in accordance with Table 7-4.
- The Plant 1 Initial Core and 11 re-loads equate to approximately 14 MT Uranium delivered in the form of 6 cm diameter pebbles. The interior of the pebble is a 5 cm diameter core consisting of TRISO coated particles encapsulated in a carbon matrix. The outer 0.5 cm skin of the pebble is a fuel free zone consisting only of the carbon matrix.
- Uranium loading pebbles is dependent on their intended use. Pebbles for the initial core (IC) are loaded at 3.4 grams U/pebble. Pebbles for re-load segments are loaded to 7 grams U/pebble.

- Actual enrichment values of the Uranium are not a factor in the fuel estimation providing that it does not exceed 19.75% U-235. The facility design will allow for processing not to exceed 19.75 % U-235.
- The initial core is comprised of approximately 2.45 MT Uranium in approximately 720,000 pebbles. Each re-load is approximately 1.05 MT U in 150,000 pebbles.
- All estimates are in 2010 dollars and include a 25% contingency.
- Building and Facility estimates are based upon recent commercial fuel feasibility studies performed at NOG-L.
- As part of the estimate for the Plant 1 fuel fabrication, there are no provisions for subsequent commercial HTGRs.
- Uranium feedstock, in the form of UO_3 is supplied by the fuel buyer. The feedstock will be of the enrichment meeting the prescribed fuel specification. Enrichment blending capabilities are not part of the estimation for pebble fuel fabrication.
- Uranium deliveries are estimated to begin 12 months prior to fuel fabrication with subsequent deliveries occurring quarterly to meet the needs of the fuel fabrication facility.
- All U-235 is fully fungible. U-235 is not tracked by contract, customer or fuel module.
- Existing B&W licensed facilities are used whenever possible.
- Kernel fabrication is performed using the existing NFS facilities. To meet required throughput, the addition of some equipment will be required. For example, fluidized bed sintering capabilities are required. No additional building construction at the NFS site is required.
- TRISO coating of kernels, pebble fabrication and inspection is performed within the NOG-L protected area. To meet required TRISO coating throughput, the addition of four TRISO coating furnaces is required. Three of the TRISO coaters will be 6” diameter as currently used in AGR development efforts. The fourth will be a 10” design and will be used to develop commercial scale up of coating activities.
- Pebble fabrication, heat treatment, inspection and machining equipment to meet throughputs is required as well. The completion of bay 15A on the NOG-L site is required to contain the TRISO coating furnaces and pebble fabrication equipment.
- Fabrication efforts will utilize existing support staff whenever possible with some “G&A” augmentation.
- Key personnel will be in place three months following award of contract.
- Facility upgrades occur concurrently with Safety and Environmental reviews.
- An approved shipping container is required for finished pebbles. An assumption that one thousand (1,000) machined pebbles will fit in one shipping container. Cost for licensing and procurement of the shipping containers is included in the estimate.
- An estimated 36 months is required to fabricate the 720,000 pebbles for Plant 1. This is following the estimated 48 months required to procure and install equipment in the bay 15A facility.
- An estimated 9 months is required for the fabrication of each HTGR re-load (150,000 pebbles).
- All non-Uranium components not provided by the fuel buyer will be purchased. Costs are included as part of the estimate.
- Shipping containers are returned to the fuel manufacturer within one year of delivery to the reactor site.
- Shipping containers are re-used for subsequent fuel shipments.

- All scrap is re-processed at the NFS site.

7.2.4 Schedule for HTGR Plant 1

A schedule for facility construction and delivery of the PBR Fuel supporting HTGR Plant 1 Initial Core is identified in Figure 7-3 and is intended to meet the fuel delivery schedule identified in Table 7-4.

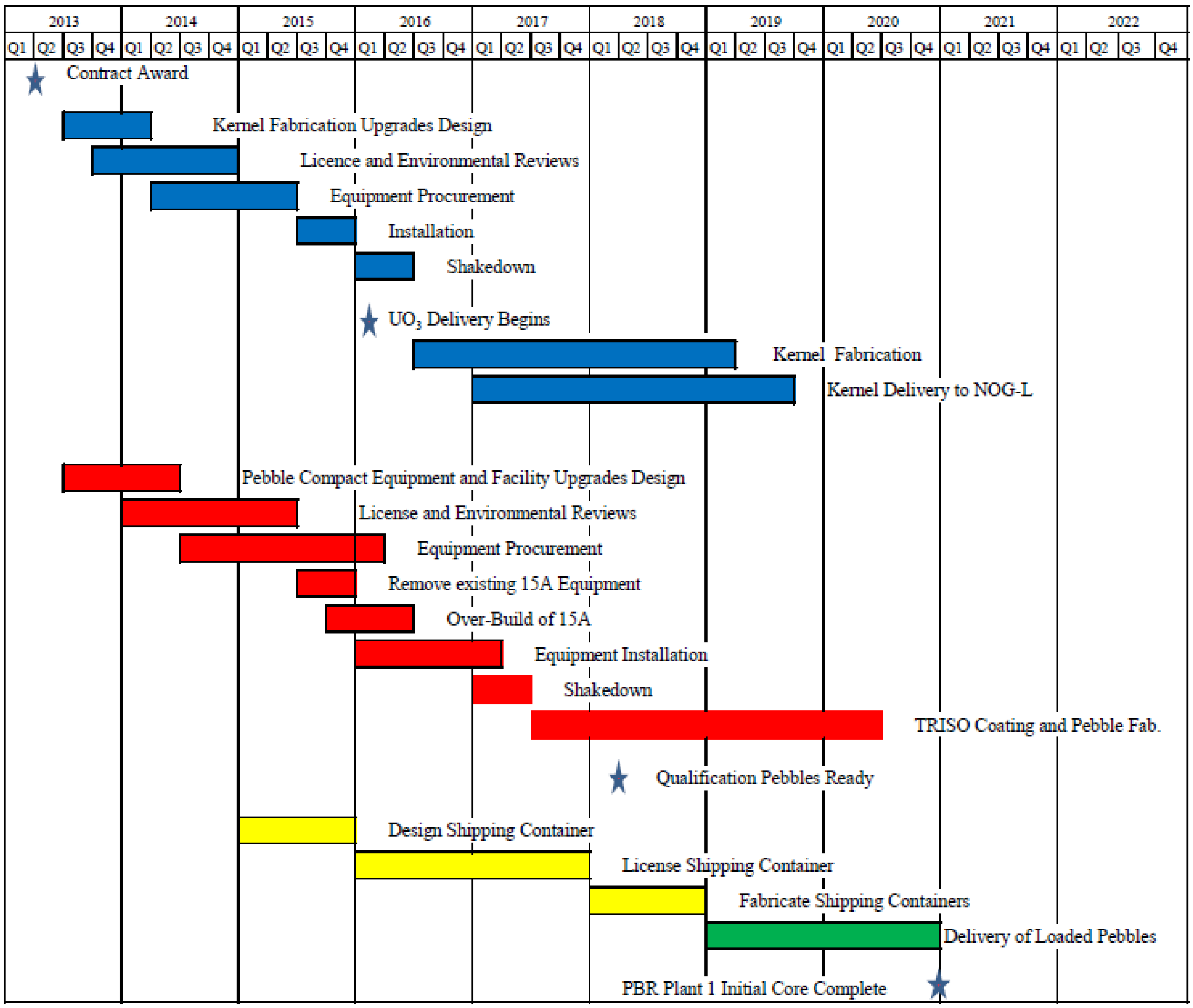
Modifications to the NFS facility include the addition of sintering capabilities beyond those which currently exist. While procurement of this equipment can occur concurrently with the safety and licensing reviews, the installation cannot occur until the licensing approvals are complete.

In order to meet the delivery of the HTGR Plant 1 initial core, 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 (20 kg's U daily). An estimated 220 kg's U is delivered to the NFS site per quarter of fuel production.

At the NOG-L site, equipment procurement would occur concurrently with the safety and licensing reviews. Changes to bay 15 A as well as installation of equipment would occur following licensing approval.

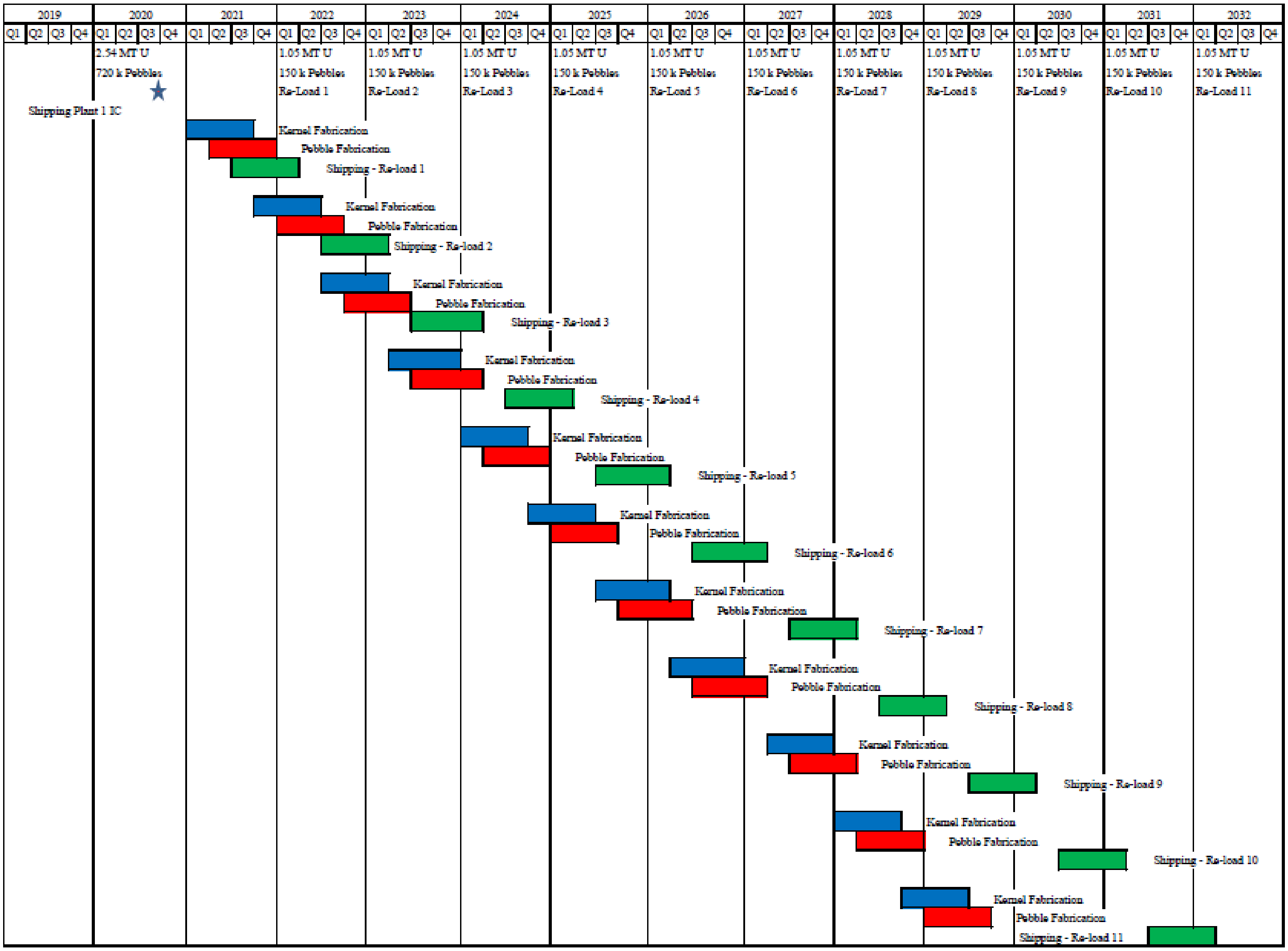
Shipping container design and approval would need to begin in 2015, approximately four years prior to shipment of pebbles. This allows for three years of design and approval followed by one year for fabrication.

Figure 7-3: HTGR Plant 1 Initial Core Facility Upgrades and Fabrication



Delivery of PBR fuel meeting the schedule in Table 7-4 for HTGR Plant 1 re-load segments is graphically depicted in Figure 7-4. Approximately 9 months is required to fabricate the fuel and pebbles for each re-load segment of HTGR Plant 1. To maintain continuous production, storage of completed pebbles is required if shipment to the reactor site, at the time of pebble completion, is required. Completion of pebble shipment is assumed complete during the first quarter in which the pebbles are needed. The cost estimate includes provisions for storage of the completed pebbles until the ship date and assumes that storage containers are returned within one year of arrival at the reactor site.

Figure 7-4: Fuel Fabrication Supporting HTGR Plant 1 for 2019 through 2032



7.2.5 Design Bases and Assumptions for Commercial HTGR Fuel Fabrication

The following is assumed as part of the cost preparation for Commercial HTGR Pebble Bed Fuel Fabrication.

- The estimate is for fuel fabrication and fuel block loading for up to 9 individual HTGR commercial plants (beyond plant 1) utilizing PBR technology. Each plant has an Initial Core and subsequent re-loads. Deliveries are estimated over a 13 year period in accordance with Table 1.
- The estimate assumes that activities to support Plant 1 were funded and successfully initiated.
- Initial cores and re-loads for the commercial HTGR modules equals approximately 79 MT Uranium delivered in the form of 6 cm diameter pebbles. The interior of the pebble is a 5 cm diameter core

consisting of TRISO coated particles encapsulated in a carbon matrix. The outer 0.5 cm skin of the pebble is a fuel free zone consisting only of the carbon matrix.

- Uranium loading pebbles is dependent on their intended use. Pebbles for the initial core (IC) are loaded at 3.4 grams U/pebble. Pebbles for re-load segments are loaded to 7 grams U/pebble.
- Enrichment of Uranium is not a factor in the estimation providing that it does not exceed 19.75% U-235.
- The initial core is comprised of approximately 2.45 MT Uranium in approximately 720,000 pebbles. Each re-load is approximately 1.05 MT U in 150,000 pebbles.
- All estimates are in 2010 dollars and include a 25% contingency.
- Building and facility estimates are based upon recent commercial fuel feasibility studies performed at NOG-L.
- The estimate for commercial HTGR fuel fabrication is independent of the estimate for fuel fabrication in support of the fuel for HTGR Plant 1.
- Upon successful start up and demonstration of the commercial HTGR fuel line (UF_6 Conversion Line, TRISO Particle Line #1 and Compact Line #1), the HTGR Plant 1 fuel fabrication line would cease and all fabrication would occur in the HTGR commercial fuel fabrication line(s).
- Uranium feedstock, in the form of UF_6 is supplied by the fuel buyer. The feedstock will be of the enrichment meeting the prescribed fuel specification. Enrichment blending capabilities are not part of the pebble fabrication process.
- Shipping container suitable for the UF_6 is required and is not part of this estimate.
- Uranium deliveries are estimated to begin 6 months prior to fuel fabrication with subsequent deliveries occurring monthly to meet the needs of the fuel fabrication facility.
- All U-235 is fully fungible. U-235 is not tracked by contract, customer or fuel module.
- Existing B&W licensed facilities are used whenever possible. Existing structures on B&W licensed facilities will not be utilized (this provides the most conservative estimate).
- Fabrication lines are modular in design. For maximum throughput, one UF_6 Dry conversion feeds the entire fuel fabrication facility.
- Following initial construction of the building and support facilities (i.e., lab, vaults, shipping areas etc,) the UF6 conversion line, an 8 MT particle fabrication line and the necessary pebble fabrication equipment would be installed. Subsequent capital outlay for a 4 MT particle fabrication line and installation of the processing lines would occur to support the fuel delivery schedule identified in Table 1. Additional pebble fabrication equipment beyond the initial install is not expected as the transition from 3.4 g U/pebble initial core pebbles is replaced with 7 g U/pebble re-load pebbles.
- Fabrication of TRISO coating of kernels, compact fabrication and fuel block loading is to be performed outside the NOG-L protected area. To meet required TRISO coating throughput, 10" TRISO coating furnaces will be utilized.
- Fabrication efforts will utilize existing support staff whenever possible with some "G&A" augmentation.
- Key personnel will be in place three months following award of contract.
- The facilities and processes will require an Environmental Impact Statement (EIS) as well as new NRC license. Facility construction will not occur until approval from the NRC and local environmental agencies.

- An approved shipping container is required for finished pebbles. An assumption that one thousand (1,000) machined pebbles will fit in one shipping container. Cost for licensing and procurement of the shipping containers is included in the estimate.
- Shipping containers are returned to the fuel manufacturer within one year of delivery to the reactor site.
- Shipping containers are re-used for subsequent fuel shipments.
- Scrap is processed through a LEU recovery facility located adjacent to the TRISO coated fabrication lines. Construction, start up and operation of this facility is included in the estimate.
- HF recovered from the UF_6 conversion process is estimated to be sold to remain cost neutral for the recovery and collection cost.
- Operation of the TRISO coated particle lines and pebble fabrication lines will remain continuous to meet the fuel delivery schedule in Table 7-4.
- All non-Uranium components not provided by the fuel buyer will be purchased. Costs are included as part of the estimate.
- Uranium Losses are estimated at 1% of the total deliverable quantity. This equates to approximately 800 kg's Uranium and is primarily in the form of dry low level waste, air discharges, wet effluent and non-recoverable solid and liquid waste.

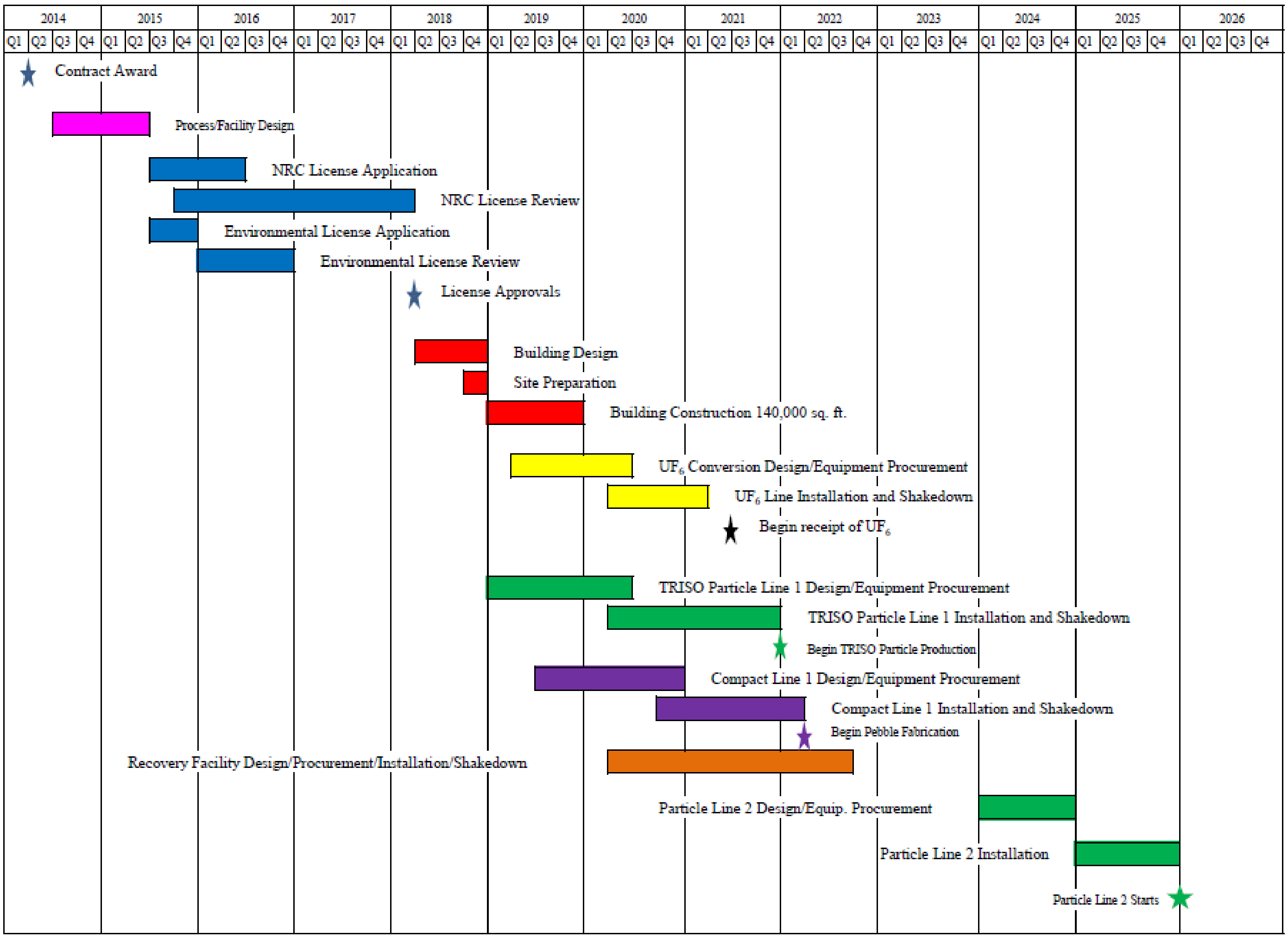
7.2.6 Schedule for Commercial HTGR Fuel

Deployment of the commercial HTGR facility and associated fuel fabrication/compacting modules is outlined in Figure 7-5.

As outlined, 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 (3.4 g U/pebble) to pebbles supporting re-loads (7 g U/pebble).

In order to support HTGR fuel fabrication, monthly Uranium shipments to NOG-L are required for the period of 2023 through 2032 (and beyond based upon life of the plant). At the sustained throughput of 10.5 MT/U annually, an estimated 885 kg's Uranium/month is required to maintain fuel production.

Figure 7-5: Commercial HTGR Facility Development Schedule



7.2.7 Staffing

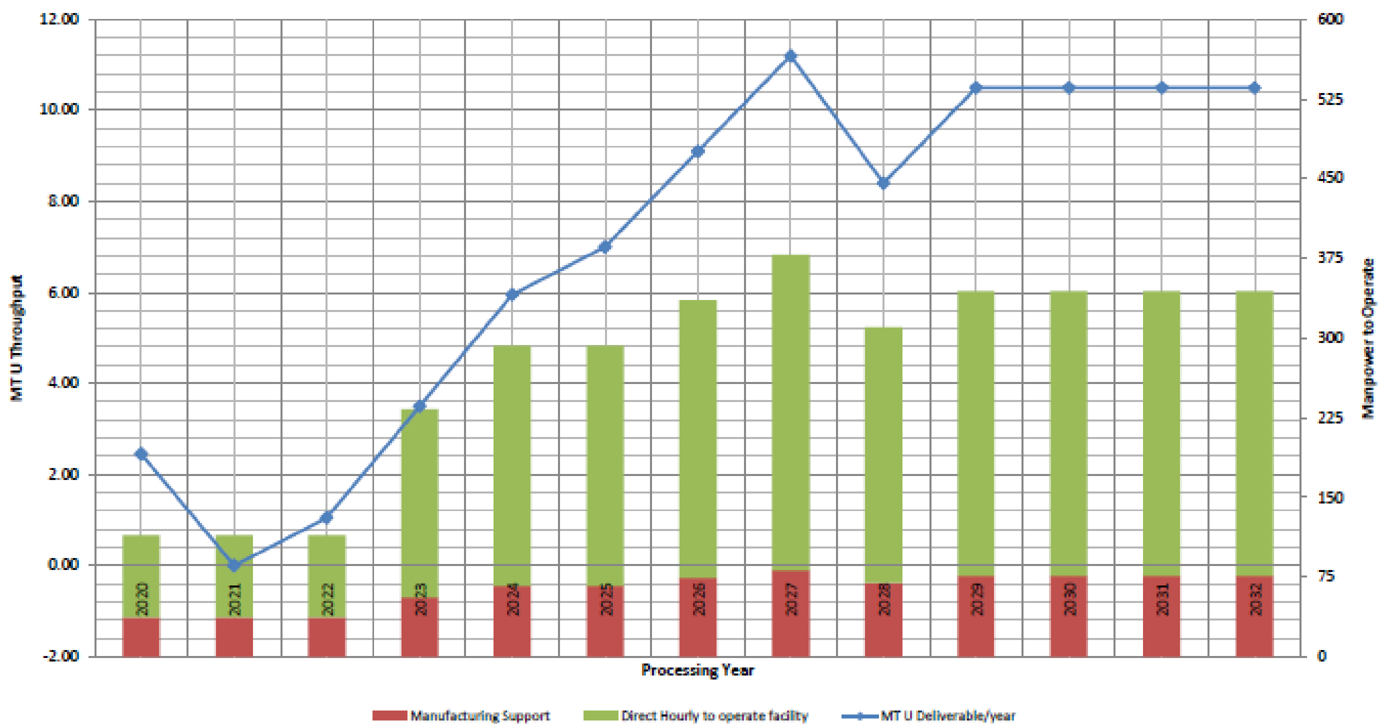
A combination of direct hourly and indirect salary employees are required to maintain production and delivery levels outlined in Table 7-4. Staffing increases as deliverable quantities increase with a maximum staffing occurring in 2027 and tapering off slightly as production efforts normalize in 2029 and beyond. The annual MT Uranium throughput, along with projected staffing levels is shown in Figure 7-6.

The following assumptions are made with respect to staffing:

- Manpower estimates include the direct labor, management, engineering and QC support required for fuel manufacturing and quality control operations. The additional manpower required to operate a commercial nuclear fuel business (product design engineering, marketing, contract management, research, accounting, personnel, etc.) is not included in these estimates. Infrastructure support manpower requirements (security, plant maintenance, grounds, IT, tooling, etc.) are also excluded. It is assumed that all of these ancillary costs would be covered either via an overhead charge that would be recalculated annually and distributed proportionally over the fuel produced in a given year or charged directly to the customer via the fuel contract. (An example of the latter would be fuel cycle engineering costs, which have traditionally been billed directly to the reactor operating entity.)

- Non union operations.
- Fuel Fabrication, Compacting and Loading is 3 shift 5 day operation, 250 working days per calendar year.
- All foreman are salaried as well as those noted in the attached table.

Figure 7-6: Fuel Fabrication Annual Delivery and Staffing Levels



7.2.8 Fuel Qualification

Qualification pebbles, for in-reactor testing is required for deployment of the HTGR Plant 1 and commercial HTGR reactors utilizing pebble bed technology. The following represents activities required for qualification of TRISO coated fuel and fabricated pebbles.

- TRISO coated U-oxycarbide particles will be fabricated at NOG-L. Equipment utilized in the fabrication of the TRISO coated particles is similar to that proposed for the HTGR Plant 1 and commercial HTGR fuel fabrication. Size and scale of the equipment, however, is smaller but still representative.

  TRISO coated U-oxycarbide particles will meet the current AGR-2 specification. TRISO coatings will be applied in the existing 6” AGR coating furnace. This furnace has been used in the AGR fuel fabrication activities since the program’s inception.

- Qualification pebbles will be fabricated using the existing AGR pilot compacting line at NOG-L.

- The over coater and heat treat furnace used in pebble fabrication will be the same as those currently used in AGR compacting development efforts.
- Development of tooling for pebble fabrication is required. Technology exchanges with PBMR and other industry experts is necessary to obtain new technology.
- If necessary, a new press will be procured to accept the pebble fabrication tooling.

- TRISO coated particle qualification will be demonstrated with ongoing AGR 5/6 efforts.
- Pebble irradiation demonstration at an international research reactor (e.g., Petten) is envisioned.
- Fuel fabrication and pebble development efforts, to support fuel qualification, is estimated at $8 M and covers a 16 month period.

  Activities need to occur in CY-2013 to support pebble fabrication equipment efforts in the HTGR Plant 1 fuel facility.

7.2.9 Fuel Acquisition

The following actions are considered to be essential in guaranteeing acquisition of PBR fuel according to the schedule outlined in Table 7-4.

- DOE (or the Public/Private Partnership) commits to fabricate the PBR reactor. In order to guarantee fuel for the HTGR Plant 1, a guarantee to fabricate PBR reactor, to operate for a minimum of 20 years, is required. To fabricate the commercial HTGR fuel module, a guarantee of 10 PBR reactors to operate for 20 years beyond start up of the last unit, is required.
- The identification of a down blended Uranium supplier and securing of that supplier through a long term contract is essential to guaranteeing fuel acquisition. Currently, a suitable supplier of fuel in the identified enrichment ranges (8% and 14% U-235) does not exist. Existing U.S.-based down blended Uranium suppliers are not equipped to handle enrichments greater than 5% U-235.
- Fuel qualification and pebble fabrication development efforts require funding. This allows for adequate development of pebble fabrication equipment, which feeds directly into the activities necessary to start up the fabrication facility.
- B&W is guaranteed as a sole supplier of PBR fuel.
- DOE (or the Public/Private Partnership) commits to cover capital expenses in event of suspension of PBR activities. A “take or pay” agreement would need to be reached before facility modifications and equipment procurement.

7.2.10 Project Risks – Technical and Schedule

Several technical and schedule risks are associated with the fabrication of fuel to support the HTGR Plant 1 and subsequent commercial HTGR modules. These risks are identified below along with the expected risk level.

- The availability of Uranium meeting the enrichment required for the HTGR Plant- 1 and commercial HTGR fuel modules is a high risk. Currently, a supplier who can meet identified annual needs has not been identified.

- Licensing of Dry UF_6 Conversion technology is a medium risk. While this technology is not FOAK, the use is new to B&W NOG-L. Handling of large quantities UF_6 is beyond current license capacities and presents several operational and environmental concerns.

- TRISO coating of UCO particles is a medium technical risk. B&W NOG-L has nearly 10 years experience in TRISO coating activities. However, the largest coater sized used by B&W is 6" in diameter. In order to meet throughputs required for the commercial HTGR modules 10" coating furnaces would be required. Successful coating in a 10" diameter coater has yet to be performed by B&W NOG-L.

- Qualification of fuel to meet HTGR performance is a medium risk. In-reactor testing of AGR fuel is ongoing. Defect types and acceptable levels of TRISO coated fuel is under development. In-reactor testing of pebbles will be required at an acceptable international location (e.g., Petten).

- Licensed containers to ship fuel pebbles is a medium risk. Licensed containers do not currently exist.

- Scale-up of a compact press and furnace technology is a low risk. Neither of these processes represent new or FOAK technology to B&W NOG-L or industry in general. NOG-L continues to develop this technology through current AGR development programs. Pebble fabrication technology is readily available from existing B&W personnel as well as through technology transfers.

- Automated pebble inspection represents a medium risk. These technologies are currently undeveloped. Commercial UO_2 fabrication facilities have developed similar technology but continue to rely on human inspection techniques for maximum accuracy.

- Pebble machining technologies represents a low risk.

7.2.11 Conclusions

In conclusion, Babcock and Wilcox is positioned to support the NGNP program through efforts to develop pebble-based fuel for the HTGR reactor. Past efforts on the AGR programs, coupled with a diverse technical team, provides Babcock and Wilcox the knowledge and experience to safely fabricate pebbles meeting the highest quality standards at a reasonable cost. With modest capital investment, the capabilities to supply fuel for the HTGR Initial Core can be secured in an approximately 5 year time frame. During this time frame, development efforts to optimize the fabrication process would occur. These efforts could then be channeled into the design and construction of a commercial fuel fabrication facility. The design of the facility would be modular. Additional modules, beyond that quoted within, can be added on an as need basis. This modular design allows for efficient scale up of commercial fuel fabrication beyond what is identified within.

As with any project involving the processing of Uranium above 5% U-235, there are risks in securing a suitable Uranium supplier. Beyond that, the risks identified are all manageable. None of the risks identified are believed to be insurmountable.

8.0 PBR GRAPHITE SUPPLY READINESS

8.1 Graphite Qualification

8.1.1 Structural Graphite in the PBR

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 support a number of critical functions:

- The graphite must act as a moderator and/or a reflector; because this depends on the number of carbon atoms per volume, the density is of importance.
- The graphite needs to provide structural support to the fuel and coolant passages and thus, the mechanical strength is of importance.
- The graphite has to be dimensionally stable during fast neutron irradiation and any changes understood.
- The graphite must be resistant to 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 fuel pebbles are considered part of the fuel, rather than core structures and are not considered in this Section.

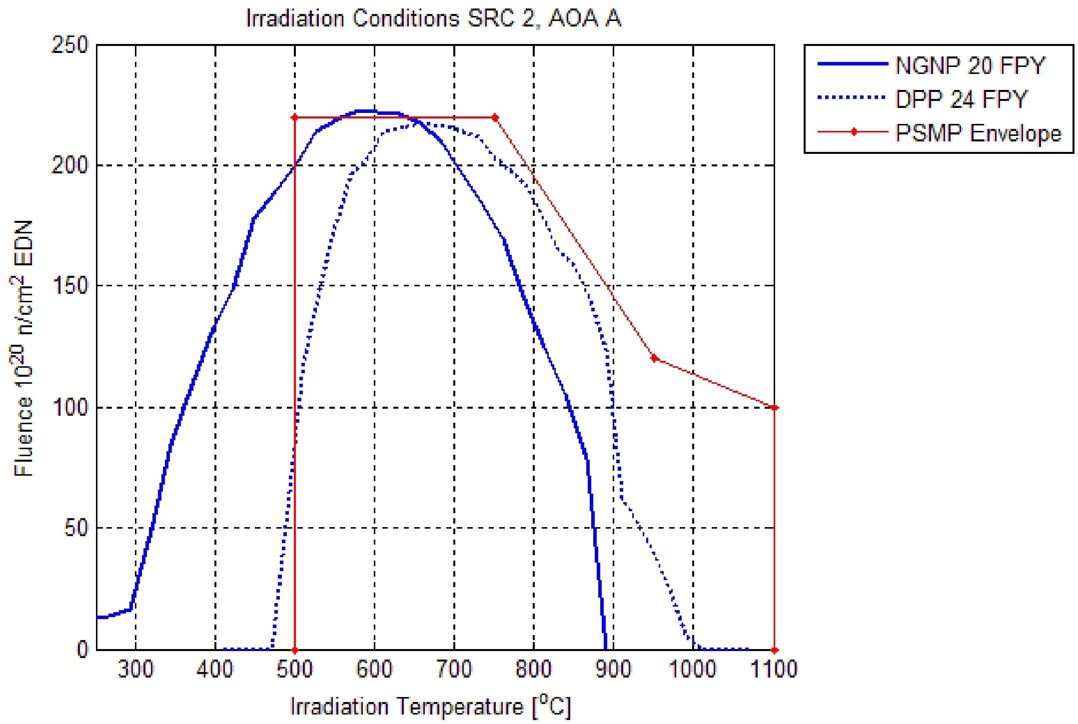
The grades of graphite that were used for previous HTGRs are no longer available. New grades of graphite have been developed based on the strengths and weaknesses of those previous grades.

8.1.1.1 Environment

The nominal operating fuel temperature in the pebble design is expected to be around 850-900ºC, with maximum local fuel temperatures well below 1200ºC. Consequently, the graphite reflector components would be exposed to temperatures ranging from 250 to 900ºC under normal conditions and peak temperatures approaching 1100ºC during accident conditions. [81]

The Figure 8-1 illustrates the relationship between irradiation and temperature for graphite parts. After 20 FPY, the maximum graphite irradiation in the NGNP PBMR is expected to be 220×10^{20} n/cm2 EDN (about 28 dpa), for parts at 600°C.

Figure 8-1: Irradiation Conditions for NGNP and PBMR DPP



8.1.1.2 Lifetime of Components

Pebble-bed concepts have historically had no design provisions for routine replacement of the reflector components, implying that these were expected to last the lifetime of the reactor. Reflector components in the PBMR 250 MWt reactor, which is strongly based on the HTR-Module design, see a substantially lower fluence-temperature regime, resulting in initial estimates of a significantly longer life of just over 40 years, based on the available data and analysis methods [81]. Recent re-evaluation of the data indicated a potentially reduced lifetime, perhaps as low as 10 years [83].

8.1.1.3 Properties

When assessing the essential functions of graphite components and evaluating their lifetime, properties of graphite that need to be considered (in no particular order) are:

- Tensile, compressive and flexural strengths
- Young's modulus
- Poisson's ratio
- Fracture toughness
- Thermal conductivity
- Coefficient of thermal expansion (CTE)
- Dimensional changes under irradiation
- Irradiation creep

8.1.1.4 Oxidation

There are two main considerations to be addressed when evaluating oxidation of structural graphite:

- Air entrance in the core: reaction between graphite and oxygen present in air at high temperature (typically > 600°C) is very exothermic and lead to the rapid consumption of the graphite.

- Reactions between graphite and impurities in helium: The nature of impurities in graphite (such as iron and vanadium) and their amounts are of the first order for the kinetics of these reactions. The behavior of these graphite components is temperature dependant. These reactions may lead to consumption of the external graphite (decrease in volume but internal graphite keeps its initial properties) or structural changes in graphite (more or less without consumption but graphite properties can change).

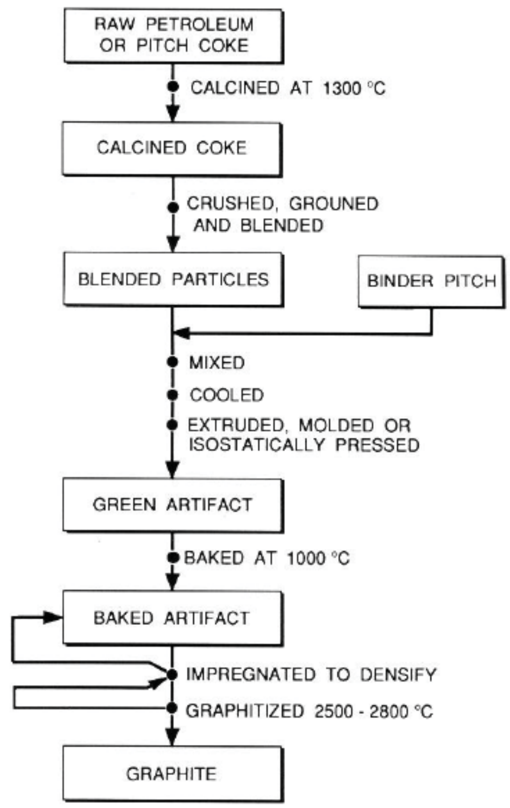
8.1.2 Graphite Manufacture

Graphite products are manufactured for a wide variety of conventional applications. Nuclear applications of graphite date from the Chicago Pile in 1942 and, since that time, over 100 graphite moderated reactors have been constructed, including six HTGRs, of which two (the HTTR in Japan and HTR-10 in China) are presently in operation.

8.1.2.1 Fabrication Technique

Natural graphite is not plentiful and so there is a need to manufacture artificial graphites for large scale applications, such as electric arc furnaces and nuclear reactor cores. There are a number of raw material and process variables that can be combined to produce graphites with the desired properties [84]. Figure 8-2 depicts a generalized process flow for producing graphite.

Figure 8-2: Graphite Manufacturing Process Flow



8.1.2.2 Raw Materials

The raw materials for nuclear graphite include coke, binder, and impregnation materials.

Coke is a solid carbonaceous material, obtained by heating a natural pitch or a residue of crude oil distillation in the absence of air. This process is long and complex, both in the choice of initial stock and its subsequent thermal treatment, and can take weeks to perform. For nuclear application, two kinds of coke are used: pitch coke (from coal-tar pitch) and petroleum coke.

Pitch (generally coal-tar pitch), which is used as the binder, is a solid (at ambient temperature) or is a highly viscous carbonaceous liquid that is also most frequently derived from petroleum crude oil or coal. Pitch is mixed with the coke to provide a material that may be molded or extruded into the desired component shape.

Impregnation materials such as pitch are used for further processing to provide for increased density and strength.

8.1.2.3 Green Pieces

The coke is calcined, crushed and screened to get a specific distribution of particle sizes. Next, the particles are mixed with the binder pitch in heated mixers to obtain a plastic paste at a uniform temperature.

Several processes are available to obtain rough blocks nearly of the shape required:

- Extrusion
- Vibration Molding
- Isostatic Molding

If coke particles are randomly oriented, the aggregate will be macroscopically isotropic, despite the inherent microscopic anisotropy of the graphite's atomic lattice. However, coke particles are seldom spherically symmetric, and the directional forces used in molding or extruding tend to align them in preferred directions, leading to anisotropy on the block scale.

- Extrusion tends to align crystallite basal planes parallel to the extrusion direction
- Pressing in a mould tends to align them perpendicular to the pressing direction.

8.1.2.4 Baking and Impregnation

The green pieces from the forming stage are then baked in a furnace, at a temperature of at least 800°C, for several weeks. After baking, the pieces are allowed to cool very slowly. The baking process changes the binder pitch into amorphous carbon.

As the volatiles are released, an extensive pore network is created. As a result, the apparent density of the graphite is quite low, and so the baked article is generally impregnated with a suitable impregnant, and then re-baked. Like baking, re-baking changes the impregnant pitch into carbon, but now the process is much faster.

This procedure is repeated until the required density is reached. This may involve one, two or three impregnation/baking cycles. As well as increasing the density, there is also a general improvement in mechanical properties, and a decrease in porosity and hence permeability.

8.1.2.5 Graphitization

Finally, the amorphous carbon material is transformed into crystalline graphite, at a temperature between 2800°C and 3300°C, in a graphitizing furnace. The blocks are stacked in close proximity in the furnace and covered completely with carbon particles as a thermal insulation. A large electric current is passed through the bed to raise the temperature of the blocks and maintain it at the required level. After graphitization, the pieces are allowed to cool very slowly.

8.1.2.6 Purification

For nuclear applications, the graphite has to be as free as possible from impurities. Most of the impurities present will become activated during the operating life of the reactor, which will give rise to operational problems, as well as decommissioning and final disposal problems. Extremely low boron levels are important from a reactor physics point of view as it is a very strong neutron absorber.

8.1.2.7 Available size

The graphite blocks are then machined to the dimensions of the finished product.

Depending on process elaboration (extrusion, molding), demonstrated available size for the billets are about (in millimeters):

- For rectangular cross section, ~600×~500 for 3000 maximum in length
- For large cylindrical section (only available by molding), up to 1600 in diameter for ~500 in length

8.1.3 Graphite Grades Description

For every supplier, graphite grades for nuclear components can be roughly defined by several process parameters (see Table 8-1 and Table 8-6):

- Filler (coke) properties
  - Origin (pitch, petroleum…)
  - Size distribution
- Form process: extrusion or molding (vibration or isostatically)

The six candidate graphite grades selected by INL for NGNP (prismatic and pebble bed designs) are the following:

- NGB-17 by SGL
- NGB-18 by SGL
- PCEA by GrafTech
- IG-110 by Toyo Tanso
- IG-430 by Toyo Tanso
- 2020 by Mersen

NGB-18 (chosen by the PBMR Project) and IG-110 (used in the Chinese HTR-10 reactor) are the grades more dedicated to pebble bed reactor.

Table 8-1: Description of Graphite Grades for Nuclear Applications

Grade	Supplier	Key Characteristics	Existing Precedent or Performed Qualification	Availability	Irradiation tests	Remarks
H-327		Needle coke filler	Fort St. Vrain	No longer available		
H-451	SGL (Great Lake)	Coarse grain (~500 µm) Petroleum coke filler Extruded	Fort St. Vrain	No longer available	HTV-1 & -2 AGC-1	Tested as a reference grade
ASR		Petroleum coke filler	AVR			Highly anisotropic

Grade	Supplier	Key Characteristics	Existing Precedent or Performed Qualification	Availability	Irradiation tests	Remarks
ASR-1S		Medium grain Pitch coke filler Vibration molded	HTR-Module			Secondary coking technique
ATR		Petroleum coke filler	AVR			Highly anisotropic
ATR-2E	SGL (SIGRI Elektrographit)	Coarse grain Pitch coke filler Extruded		No longer available	Results synthesized in HTR-M1	
HLM	SGL	Medium grain (~76 µm) Petroleum coke filler Extruded	Fort St. Vrain		AGC-1	Similar to PGX
NBG-10	SGL	Coarse grain (~160 µm) Pitch coke filler Extruded		Available	AGC-1 INNOGRAPH-1A INNOGRAPH-1B INNOGRAPH-2A INNOGRAPH-2B	PBMR's original choice
NBG-17	SGL	Medium grain (~80 µm) Pitch coke filler Vibration molded	No	Production of small batches.	HTV-1 & -2 AGC-1 INNOGRAPH-1B INNOGRAPH-2A INNOGRAPH-2B	
NBG-18	SGL	Coarse grain (~160 µm) Pitch coke filler Vibration molded	Partially Qualified for PBMR Project	Available. Has been in continuous production	HTV-1 & -2 AGC-1 INNOGRAPH-1B INNOGRAPH-2A INNOGRAPH-2B	First irradiation data based on grade ATR-2R
NBG-20	SGL	Petroleum coke filler Extruded			INNOGRAPH-1A INNOGRAPH-1B INNOGRAPH-2A	

Grade	Supplier	Key Characteristics	Existing Precedent or Performed Qualification	Availability	Irradiation tests	Remarks
NBG-25	SGL	Fine grain (~10 µm) Petroleum coke filler Isostatically molded		Available	AGC-1 INNOGRAPH-1A INNOGRAPH-1B INNOGRAPH-2A INNOGRAPH-2B	
PCEA	GrafTech	Medium grain (~76 µm) Petroleum coke filler Extruded	No		HTV-1 & -2 AGC-1 INNOGRAPH-1A INNOGRAPH-1B INNOGRAPH-2A INNOGRAPH-2B	
PCIB-SFG	GrafTech	Fine grain Petroleum coke filler Isostatically molded			INNOGRAPH-1A INNOGRAPH-1B INNOGRAPH-2A INNOGRAPH-2B	
PGX	GrafTech	Medium grain (~76 µm) Petroleum coke filler Extruded			AGC-1	
PPEA	GrafTech	Medium grain (~76 µm) Pitch coke filler Extruded			AGC-1 INNOGRAPH-1A INNOGRAPH-1B INNOGRAPH-2A INNOGRAPH-2B	
LPEB/ BAN	GrafTech	Medium grain (~80 µm) Secondary/Needle (petroleum) coke filler Extruded		Not commercially available	AGC-1 INNOGRAPH-1B INNOGRAPH-2A INNOGRAPH-2B	
LPIB					INNOGRAPH-2B	

Grade	Supplier	Key Characteristics	Existing Precedent or Performed Qualification	Availability	Irradiation tests	Remarks
IG-110	Toyo Tenso	Fine grain (~10 µm) Petroleum coke filler Isostatically molded	HTTR HTR-10	Available. In continuous production.	AGC-1 INNOGRAPH-1A INNOGRAPH-1B INNOGRAPH-2A INNOGRAPH-2B	
IG-430	Toyo Tenso	Fine grain (~10 µm) Pitch coke filler Isostatic-molded		Available. In continuous production.	HTV-1 & -2 AGC-1 INNOGRAPH-1A INNOGRAPH-1B INNOGRAPH-2A INNOGRAPH-2B	
2020	Mersen (Carbone of America)	Fine grain (~15 µm) Petroleum coke filler Isostatically molded	Fort St. Vrain		AGC-1	
2114	Mersen (Carbone of America)	Super fine grain Pitch coke filler Isostatically molded				2020 replacement
2160	Mersen (Carbone of America)	Ultra fine grain Pitch coke filler Isostatically molded		In development		
2191	Mersen (Carbone of America)	Super fine grain Petroleum (sponge) coke filler Isostatically molded		In development		

8.1.4 Codes and Standards

8.1.4.1 ASTM Standards

Two main ASTM standards have been available since 2008 for graphite nuclear grades:

- ASTM C781-08: Standard Practice for Testing Graphite and Boronated Graphite Materials for High-Temperature Gas-Cooled Nuclear Reactor Components

- ASTM D7219-08: Standard Specification for Isotropic and Near-isotropic Nuclear Graphites

In the same time, numerous ASTM standards for graphite (called by C781-08) have been published or reevaluated. They are listed in the Table 8-2.

Some characteristics determination standards are still missing, including those covering:

- Fracture toughness,
- Fatigue,
- Irradiation creep, and
- Friction.

8.1.4.2 ASME Codes

Only minimal guidance has been available from established regulatory requirements or the ASME Code regarding the use of graphite. This situation is expected to evolve, since a consensus ASME code on graphite component design for HTGRs has been prepared by the ASME Subgroup on Graphite Core Components and will be published in 2011 in the 2011 Code Addenda.

The addition of nuclear grade graphite to the ASME Code would be ideal, but it is not required for qualification. Until the graphite code is published and accepted by the NRC, graphite material selection will focus on existing design and operating experience with both past and currently available grades of reactor graphite. Fabrication experience and technical maturity are additional selection criteria that must be considered.

Notwithstanding these criteria, the performance and compatibility of candidate materials with the operating environment will be assessed. Testing is currently underway to qualify the new grades of graphite, including tests to obtain data on physical, mechanical, and oxidation relevant properties (including the effects of irradiation and irradiation-induced creep).

The complete draft of the ASME design code will not be available until early 2011 and this is not expected to be endorsed by ASME (and included in the B&PV code) until later in 2011, when it will be included in the addenda as ASME III, Division 5. Given the existing European and US irradiation programs, the complete data set on the irradiation behavior of the chosen graphites will also not be available until 2021. Nevertheless, work on producing the proposed code, in particular the Data and Method Sheets, can start in the near future.

8.1.5 Qualification Programs

The main objectives of qualification program for graphite supply of core components are:

- Verifying that the behavior of the graphite grade, under operating conditions (fluence, temperature, etc.), 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.

- Verifying that graphite characteristics, in every point of the billet, correspond to the requirements.

- Verifying the reproducibility of graphite material properties during the whole production of the components:

    - For all the billets of the same lot of production (evaluation of the dispersion in one lot);

- o Lot by lot (stability of the process).

The stability of the process can be evaluated by the characterization of properties that are not required by the design but are relevant to the process (such as electric conductivity for example).

8.1.5.1 Behavior under Operating Conditions

The goal of the designer is to design the graphite core structure by taking into account real behavior of the graphite under operating conditions. Unfortunately, at the beginning, only data concerning properties before irradiation are available. And moreover, only characteristics before irradiation will be available as acceptance tests of supplied graphite.

So, the first challenge of the qualification program is to link properties of the graphite under operating conditions with characteristics measured on graphite before irradiation.

Tests under irradiation are performed to make this link. These tests give characteristics of the graphite during irradiation. Nevertheless, in some cases (thermal conductivity for example), it is acceptable to use data from other grades to complete knowledge of the grade.

8.1.5.2 Requirements

Reference [81] gives some ideal requirements for reflector graphites, in a general point of view. The applicable ASTM specification [85] is dedicated to a large spectrum of applications. On the contrary, the PBMR Project has written specifications totally dedicated to their project.

Table 8-2: Key Requirements for Reflector Graphite

Parameter	Standards	Comments
Bulk density	ASTM C838-96 for graphite as manufactured ASTM C559-90 for specimen tests.	High density is indicative of lower porosity, provides for more effective neutron moderation/reflection per unit volume, and in general, also indicates higher strength.
Neutron absorption cross-section	ASTM C1233-09 for Equivalent Boron Content PBMR 041064	A low cross-section is required for neutron efficiency of the core. The limiting neutron absorbency is that of pure carbon (~3.5 mbarn).
Thermal conductivity at room temperature	ASTM E-1461 for thermal diffusivity measurement DIN 51908-2006	High values of thermal conductivity indicate a high degree of graphitization and are required for effective heat transfer in HTGR applications.

Parameter	Standards	Comments
Ash content (purity)	ASTM C561-91 for graphite DIN 51903-81 for coke DIN 51922-83 for binder and impregnant pitch	High level in purity is required to minimize activation and reduce susceptibility to catalytic oxidation. It is possible to manufacture graphite with much higher purity levels using a dedicated purification step. The selected and specified purity may vary depending on the function of the components.
Strength	ASTM C749-08 or DIN 51914-2006 for Tensile Strength ASTM C695-91 for Compressive Strength ASTM C651-91 or DIN 51944-2006 for Flexural Strength (4-point)	Adequate strength is required for structural component integrity. The strength reserves offered by the material must exceed the allowable operating component stresses. Higher strengths are achievable with isostatically molded, fine grain graphite, but these typically possess lower fracture toughness. Due to natural standard deviation in results for this test in graphite samples, numerous tests are necessary
Coefficient of Thermal Expansion	ASTM E228-06 DIN 51908-2006	A higher value is indicative of the coke isotropy and hence isotropy of the graphite. This implies that the graphite will have better dimensional stability when subjected to fast neutron irradiation. However, lower CTE can be beneficial in terms of thermal stress.
Coefficient of Thermal Expansion Isotropy Ratio	ASTM C781-08 DIN 51937-94	This coefficient is indicative of the bulk graphite isotropy.
Dynamic Elastic Modulus	ASTM C747-93	Higher modulus is typically associated with a higher strength material, but increased sensitivity to thermal stresses. Thus, values at the lower end tend to be more beneficial.

Parameter	Standards	Comments
Oxidation Resistance	ASTM C 1179-91 SGL P/RD/LC/66	This characteristic is usually asked for information.

8.1.6 Characterization of Graphite Grades before Irradiation

8.1.6.1 General

The major new grades (such as NBG-18, PCEA, IG-110, IG-430, NBG-17, 2114) are or have been characterized before irradiation as a standard of comparison for irradiated samples. All the characteristics described in the next section (characteristics after irradiation) have also been performed at the pre-irradiation stage.

For example, in the US, an extensive characterization effort is currently underway at the INL and ORNL laboratories to establish the material properties before irradiation on a series of large graphite billets, from NBG-18, PCEA, IG-110 and 2114 grades.

The CEA has performed a whole characterization before irradiation of the PCEA and NBG-17 grades [86].

Moreover, NBG-18 has been qualified for the PBMR project [87]. This qualification is connected to a whole characterization of this grade before irradiation.

8.1.6.2 Oxidation

As a microstructural component, graphite porosity plays an important role in the dynamics of fracture processes. A low level of porosity is inherent to the new graphite grades developed for nuclear applications.

Although many types of pores, with various shape and size, are present in the graphite structure, it is generally agreed that large, slit-shaped pores are the most damaging to the graphite integrity.

It is generally accepted that the dominant mechanism of graphite oxidation varies with the temperature. Oxidation is controlled by the kinetics of the chemical process at low temperatures, but becomes diffusion-limited at high temperatures, and is strictly limited to mass transfer in the boundary layer at very high temperatures.

When the rates of oxidation are slower than those of in-pore diffusion, oxidation is under kinetic control, and the oxidant penetrates deeper into the bulk of material, leading to (almost) uniform oxidation of specimens. Although with extremely slow rate, this mechanism of oxidation may cause extensive damage of material's strength, because it extends deep into the bulk of graphite.

As the temperature goes higher and the rate of reaction accelerates, diffusion rate becomes the controlling factor and the penetration depth of the oxidant gradually diminishes. This leads to narrow oxidation zones limited to a thin layer at the graphite exposed surface, and leaves the bulk of material practically unchanged. Consequently, in spite of evident surface damage, the mechanical strength of the core material may not be seriously affected [88].

Oxidation tests performed by the KAERI [89] on IG-110, IG-430, NBG-18 and NBG-25 grades, at six temperatures between 600-960°C, have shown little differences on behavior between these selected grades.

French teams (CEA and AREVA) have performed oxidation tests in impure helium [90] and also in the Oxygraph open loop representing the case of a limiting air entrance in the core. Grades tested were NBG-10, NBG-17 and PCEA.

Oxidation behavior of graphite after irradiation has not been studied yet.

8.1.6.3 Tribology

A bibliographic survey concerning tribology has been performed during European RAPHAEL program [91].

. The main conclusions of this survey are:

- No adhesion occurred in graphite-graphite contacts at temperatures up to 800°C.
- There was no systematic effect of gas pressure on friction and wear in pure Helium.
- Friction coefficients were highly dependant upon the moisture level in the gas.
- Graphite-graphite pairs wore little in Helium in the range 400-800°C.

Then, wear tests between two pieces of graphite or between graphite and metal have been performed, at high temperature (Table 8-3).

Table 8-3: Tribologic Tests Performed During RAPHAEL

Material 1 Disk/Rail	Material 2 Pin	Temp. (°C)	Load (MPa)	ANP Tribometer	CEA Tribometer
PCEA	PCEA	1000	10	X	X
PCEA	PCEA	1000	20	X	
PCEA	PCEA	800	10		X
PCEA	PCEA	800	5	X	
NBG-17	NBG-17	1000	10	X	
NBG-17	NBG-17	800	10	X	
NBG-17	NBG-17	800	5		X
800H	PCEA	850	10	X	X
800H	NBG-17	850	10	X	
800H	PCEA	500	10		X
800H	NBG-17	500	10	X	

Four tests with the 800H/graphite pairs have been made, showing a low influence of temperature and differences in friction coefficient between both kinds of graphites. In all cases the wear level was low.

The structure of the debris depends on temperature: at 1000°C, levels of debris was relatively small, and increased in quantity at 800°C and 500°C. At 500°C, debris also spread over larger proportions of the plates.

These results were compared with feedback experience from past HTGRs. This comparison proved to be difficult because the temperatures were generally lower, and the test conditions as well as the graphite properties not exactly known. Values were found to be in good agreement with data that were the most reliable and relevant.

8.1.7 Characterization of Graphite Grades after Irradiation

Public data concerning irradiation behavior of new grades of graphite are only available from the US (HTV and AGC) and the European Union (principally RAPHAEL). Most of the data obtained by other countries are confidential.

8.1.7.1 RAPHAEL Program

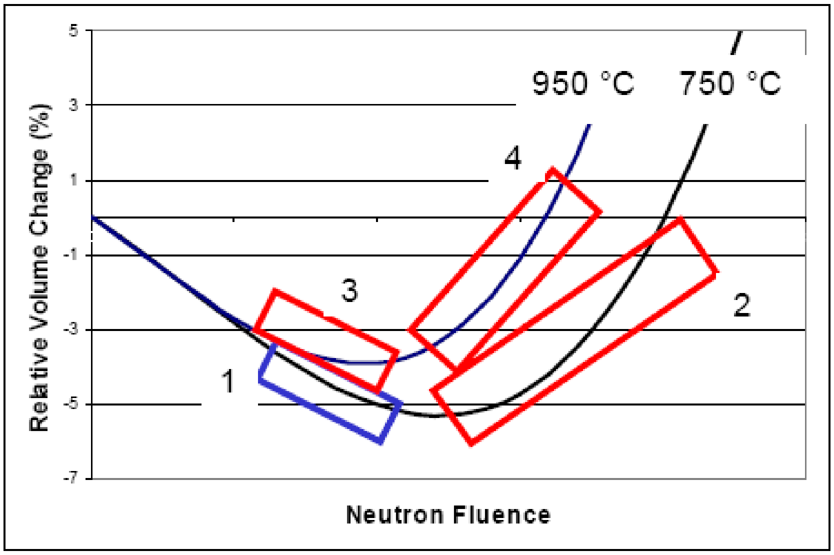
The NBG-18 qualification for PBMR project has been done for irradiation behavior by using data concerning old grades, in particular ATR-2E, which is described as very close to NBG-18. Future data on NBG-18 from European or US programs would complete these data.

During the European program HTR-M, a bibliographic review was written on characterization of irradiated graphite. The graphite data review is concentrated on ATR-2E, which has been tested in great detail under fast neutron irradiation [84]. ATR-2E is not commercially available today.

The EU HTGR Program RAPHAEL was therefore established to include test work on graphites covering graphite irradiation, graphite oxidation, microstructural modeling and the development of guidelines for design. Available commercial graphite grades were investigated and selected and irradiation tested in the HFR at Petten to develop curves of irradiated properties, which can be used for design and to confirm selection.

The list of irradiated grades is given in the Table 8-1 (INNOGRAPH experiments). The general irradiation plan for these experiments is depicted in Figure 8-3.

Figure 8-3: General Plan for RAPHAEL Irradiation Programs



1. INNOGRAPH-1A: 750°C, low/medium dose (FP5)
2. INNOGRAPH-1B: 750°C, high dose
3. INNOGRAPH-2A: 950°C, low/medium dose
4. INNOGRAPH-2B: 950°C, high dose

The maximum dose of interest was considered to be around 25 dpa (estimated end of life dose for the pebble bed design) at 750°C and 16 dpa at 950°C. One irradiation would only give information between 16 to 25 dpa (at 750°C) taking account of the radiation buckling distribution of the HFR reactor, which would be insufficient to construct a design curve. It was decided therefore to have two irradiation experiments to produce each design curve. The first experiment was targeted at one third of the maximum dose; the second, which contains fresh un-irradiated samples and irradiated samples from the first experiment, is targeted at two thirds of end of life dose.

In this way data is generated at a 1/3, 2/3 and 3/3 of end of life dose and data points between these levels will be obtained by making use of the flux buckling of the reactor. The first test was not expected to yield properties up to turn round behavior but provided information on whether the graphite is useful or not thereby providing a first sorting of the available graphites and as the experiment progressed there is the possibility of replacing those that are unsuitable with alternatives.

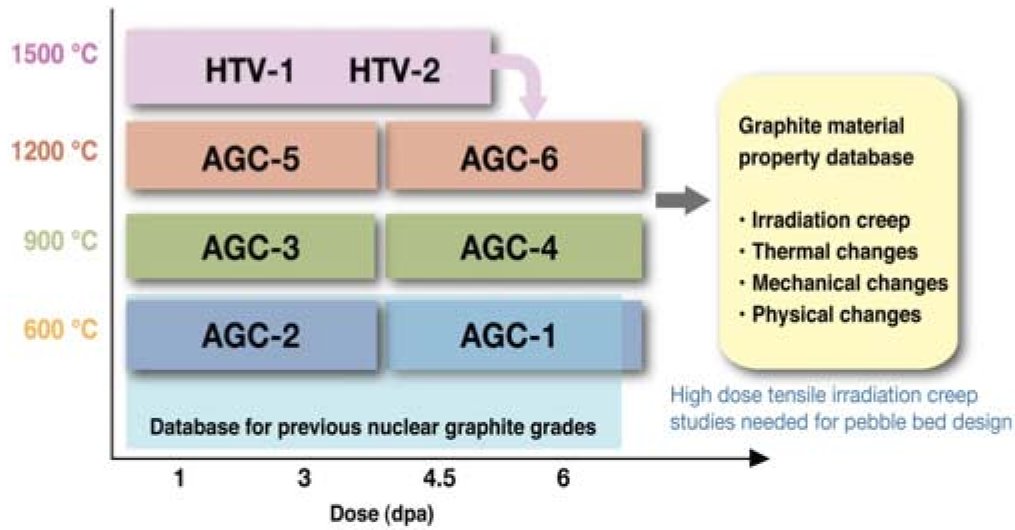
The PIE characterizations have to be performed by the end of 2010. Results will be analyzed and published as a deliverable of the new European program ARCHER (not before 2011). The RAPHAEL program gives numerous data on physical characteristics of graphite after irradiation, but any measurement on mechanical properties (strength, etc.) has not been performed. This will be one of the challenges of the AGC program, to provide data on strength and toughness after irradiation and on irradiation creep.

8.1.7.2 HTV and AGC (US Program)

A series of eight irradiations are planned to establish the thermo-mechanical and thermo-physical response of the major grades of graphite (see Table 8-1 for the list of the grades) as a function of temperature and radiation dose Figure 8-4. Advanced Graphite Capsule (AGC)-1 through AGC-6 will be conducted at INL's ATR to establish the behavior of graphite in the temperature/dose envelope for NGNP.

HTV-1 and HTV-2 will be conducted in the High Flux Isotope Reactor (HFIR) at ORNL to establish graphite behavior under accelerated temperature and damage conditions so that AGC-6 can be designed properly, accounting for shrinkage/swelling and creep anticipated at the high temperature and high dose.

Figure 8-4: AGC and HTV Irradiation Programs



These irradiations will contain specimens of sufficient size, number, and type to support statistical assessments necessary to capture the inherent variability in graphite; to support traditional ASTM requirements for sample analysis; and to more completely characterize the physical, thermal, and mechanical properties of the irradiated graphite.

Table 8-4: AGC and HTV Program Schedule

Experiment	Irradiation	Post-Irradiation Examination
AGC-1	Sept. 2009 – March 2011	June 2011 – Nov. 2013
AGC-2	March 2011 – Feb. 2012	Feb. 2012 – Aug. 2014
AGC-3	Feb. 2012 – Jan. 2013	Jan. 2013 – June 2015
HTV-1 HTV-2	May 2012 – Oct. 2012	Oct. 2012 – April 2015
AGC-4	Jan. 2013 – Dec. 2013	March 2014 – Sept. 2016
AGC-5	Dec. 2013 – Nov. 2016	Jan. 2015 – June 2018
AGC-6	Dec. 2016 – Oct. 2018	Jan. 2019 – July 2021

8.1.7.3 Summary of Graphite Qualification Programs

Table 8-5 presents a summary of the information that will be developed during the course of the AGC and RAPHAEL programs.

Table 8-5: Summary of Graphite Irradiation Programs

	Before irradiation	**After irradiation** Low fluence (<7 dpa)	**After irradiation** High fluence (7-20 dpa)	**After irradiation** Very high fluence (>20 dpa)
Shrinkage	NA	Data at 750°C and 950°C (RAPHAEL) Irradiation in progress for 600°C and 900°C (AGC) **No scheduled irradiation program between 400 and 600°C**	Data at 750°C and 950°C (RAPHAEL) **No scheduled irradiation program between 400 and 750°C**	**No scheduled irradiation program.**
Physical properties (thermal conductivity…) + Dynamic Young's Modulus	Data from RAPHAEL or US programs	Data at 750°C and 950°C (RAPHAEL) Irradiation in progress for 600°C and 900°C (AGC) **No scheduled irradiation program between 400 and 600°C**	Data at 750°C and 950°C (RAPHAEL) **No scheduled irradiation program between 400 and 750°C**	**No scheduled irradiation program.**
Mechanical strengths	Data from Europe and US programs	Irradiation in progress for 600°C and 900°C (AGC) **No scheduled irradiation program between 400 and 600°C**	**No scheduled irradiation program.**	**No scheduled irradiation program.**
Irradiation creep	NA	Irradiation in progress for 600°C and 900°C (AGC) **No scheduled irradiation program between 400 and 600°C**	**No scheduled irradiation program.**	**No scheduled irradiation program.**

8.1.8 Manufacturing Program

Graphite characteristics (before and after irradiation) are very dependant on the manufacturing process. After the qualification of a graphite grade, it is mandatory to follow a manufacturing program (through quality assurance) describing the whole process in order to avoid deviation in the quality of the products.

The PBMR Project published a document which includes all the requirements for quality assurance provision. It includes the following:

- Characteristics of the filler (coke origin, chemical analysis, size distribution)
- Characteristics of the binder and the impregnant (origin, chemical analysis, coking value)
- Characteristics of the other additives
- Process parameters of the mixing (quantities of each compounds, temperature and time of mixing)
- Process parameters of the forming (extrusion or molding)
- Requirements on green billets
- Process parameters of the baking (furnace, time-temperature chart)
- Requirements on billets after baking
- Process parameters of the impregnation and re-baking (temperature and pressure of impregnation, furnace, time-temperature chart)
- Requirements on billets after the second baking
- Process parameters of the graphitization and the eventual purification (furnace, heating-power chart, purification method, cooling-down method, graphitization temperature)
- Final graphite inspection of every billet (mass, dimension, bulk density, electrical conductivity)
- Sampling plan and charge acceptance criteria

8.1.9 Boronated Graphite

Boronated graphite (really boronated baked carbon that is not fully graphitized) is produced by mixing submillimetric B_4C particles with the coke before baking. The amount of B_4C depends on expecting boron content in the final product.

Some articles have been published in the 1960s and 1970s (essentially in oxidation resistance and irradiation behavior due to the helium production) but only a few recent papers concerning boronated graphite for HTGR are available, including:

- Irradiation Behavior of Boronated Graphite for the HTTR, from the JAERI (1991), in which irradiation behavior of laboratory-made boronated graphite is studied (3 and 30 wt% of boron). Boronated graphite was produced by mixing B_4C particles with graphite particles. Then samples were sintered. There is no information concerning the graphite used for the preparation of the samples.

- Characterization of Boron-doped Graphite for Possible Nuclear Applications, presented by GrafTech during the 7th INGSM in 2006. Two grades have been tested before irradiation (no data on boron content). By comparing results with unboronated graphite, they found a dramatic decrease in the CTE and change in Young's modulus. Moreover, microstructure changed a lot with boron addition.

The HTR-Module SAR only mentions the need of "boronated graphite" but gives no technical data on it.

In the Chinese HTR-10, in the side reflector, boron content depends on the location of the graphite part, no information has been found about the manufacturing process to obtain this graphite with fluctuations of boron content.

For use in the US program, if boronated graphite is required, a new manufacturing plant will be needed to produce the material. Machining capabilities on the necessary scale will also need to be developed.

The South-African PBMR Project only needs (unboronated) NBG-18 graphite grade.

Nevertheless, testing boronated graphite is included in the ASTM standard practice (C781) concerning graphite for high-temperature Gas-Cooled Nuclear Reactor.

8.2 Graphite Acquisition

Today, there are only a few graphite suppliers actively considered as suppliers by potential NGNP reactor vendors:

- GrafTech (US)
- SGL Group (Germany and France)
- Toyo Tanso (Japan)
- Mersen (former Carbone-Lorraine, France) and subsidiaries in the US like Carbone of America

These graphite vendors are experienced at producing graphite for nuclear applications. They understand and are able to meet the quality requirements for nuclear components. Table 8-6 indicates the major nuclear grades produced by these vendors.

Table 8-6: Major Nuclear Graphite Grades

	Pitch coke	Petroleum coke	Other
Extruded	NBG-10 (SGL) PPEA (GrafTech)	NBG-20 (SGL) PCEA (GrafTech) PGX (GrafTech)	LPEB/BAN (GrafTech) (secondary needle petroleum coke)
Vibration molded	NBG-17 (SGL) NBG-18 (SGL)	----	----
Isostatically molded	IG-430 (Toyo Tanso)	NBG-25 (SGL) PCIB-SFG (GrafTech) IG-110 (Toyo Tanso) 2020 (Mersen)	----

These vendors are able to control the properties of the finished product by controlling the feedstocks used and maintaining appropriate process controls. They are experienced at keeping properties within the expected specifications and impurities below their limits.

All graphite has some variability in properties. The key requirement is to understand the variability and to set appropriate requirements to bind the variability. Then the reactor vendors can design components for the specified variability, while vendors ensure that the actual variability is within the specification. While the variability of a material such as graphite may be larger than some more common materials, this fundamental nature of the design process is not that different.

8.2.1 Sustainability of the Grades

8.2.1.1 Toyo Tanso

The IG-110 graphite grade has been in production for over 35 years from development to the present, and it is used not only for the nuclear graphite but also as global standard grade for various isotropic graphite applications.

The IG-430 graphite grade has been developed as a material for the next generation of nuclear applications based on the sufficient production experience and know-how obtained through IG-110 production.

Toyo Tanso is confident with the ability to supply stable next generation graphite for long term projects [92].

8.2.1.2 SGL

All recipes and production processes are frozen and in all detail defined in a quality plan. The raw material supply is secured by a long-term supply agreement. SGL is prepared to deliver nuclear graphite as of today; however, long term outlooks, in particular for NOAK plants, can only be based on bilateral agreements as NOAK plants would definitely require considerable expenditures and the allocation of resources [92].

8.2.2 Raw Materials Issues

8.2.2.1 Pitch coke

Pitch cokes are made from coal tar, which is produced as a by-product in coke ovens. Previous German developments focused on pitch coke for their graphite development program, following the oil crisis of 1978.

Because of economic and environmental factors, Japan is currently the only source of pitch coke [81].

8.2.2.2 Petroleum coke

Petroleum coke accounts for by far the largest tonnage of coke made worldwide, and is available in the US. Oil refineries are run to optimize the production of fuels, so petroleum cokes made from the heavy end of the distillation process will have variable quality and properties dependent on the crude source and refinery operation.

However, on the west coast of the United States, certain smaller refineries have developed a niche business supplying specialty isotropic cokes made from sweet light crude [81].

8.2.2.3 Toyo Tanso

According to Toyo Tanso, their raw materials used for graphite production have a stable supply source secured for a long term basis. They are confident with their stable supply chain in the future. An alternative product grade to IG-110 is IG-430, raw material supply of which has been also prepared [92].

8.2.2.4 SGL

Alternative supplies for raw materials and also significant changes in the specification of the raw materials would normally require a new qualification of the grade. Understanding of the influences of exchanging the raw material for the nuclear properties, in particular behavior in irradiation is not fully understood by many parties.

Thus, any new raw material might lead to a new graphite grade in terms of nuclear properties.

However in Germany this problem was already solved and the information is available to SGL as we were the chosen supplier for the German HTGR project at that time. In any case a new grade requires qualification via a program as currently initiated by the DOE.

Therefore it is essential to freeze the recipes and production processes and bind the suppliers with long-term agreements [92].

8.2.3 Procurement and Available Quantities

The required quantity of graphite for the FOAK NGNP is small compared to the total production volume of any of the graphite vendors. However, the nuclear graphite specifications require production steps that have much more limited capacities. The existing infrastructure is believed to be adequate to produce the quantity of the selected grade of nuclear graphite on the planned NGNP production schedule. However, this assumes that the required quantity of graphite is ordered in a timely manner.

8.2.3.1 Toyo Tanso

Current production capacity for nuclear applications is approximately 1000 tons annually.

Under the assumption that the NGNP HTGR uses up to 2000 tons, it would take approximately 2.5 to 3 years to produce the whole quantity, with a lead-time of approximately 6 month [92].

8.2.3.2 SGL

The quantities for Westinghouse UK are more than 500 MT (metric tons) annually, but further details cannot be disclosed. For NBG-17 and NBG-18 SGL produces between 5 and 10 tons annually.

Assuming approx. 1,200 MT of graphite parts, it would take 3 years to produce the required material. Depending on ones needs, more resources could be allocated to reduce delivery times. A business case would need to be completed to determine how the additional resources would change delivery times and their effect on pricing.

It will take roughly 3 years to establish a new location to manufacture the graphite if the graphite manufacturing is to be located near the reactor construction site. This time would be dependent on using an existing SGL site and revamping it or if we need to build a new facility. If delivery is initiated from our current production facility it will only take 1 year. A machine shop would be set up locally no matter where the material is produced, but this would not add extra time because the production of the material is the long lead time item in the system. A machine shop can be set up within the timeframe to produce the graphite itself [92].

8.2.4 Graphite Acquisition Conclusions

The graphite infrastructure is believed to be adequate to produce the quantity of the selected grade of nuclear graphite on the planned NGNP production schedule. This assumes that the required quantity of graphite is ordered in a timely manner. The main issue on graphite acquisition is that every change in raw materials (and more specifically in filler coke origin) will involve the qualification of a new grade. After qualification, in order to secure graphite supplying, 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 stock for pitch coke graphite, like NBG-18, because pitch coke sources are rare.

9.0 PBR CONSTRUCTABILITY AND TRANSPORTABILITY ASSESSMENT

9.1 Building Construction

9.1.1 Reactor Building and Reactor Auxiliary Building

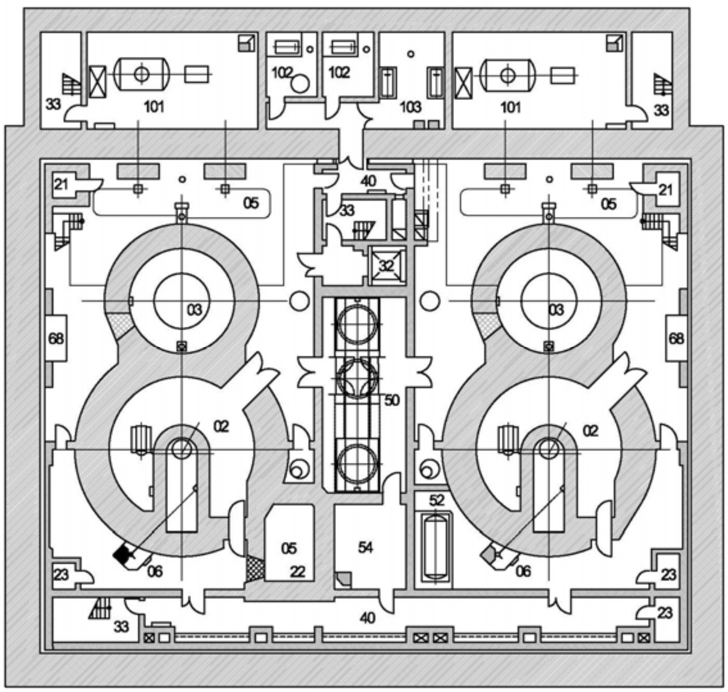
The Reactor Building and the Reactor Building Annex are structurally a single complex structure that shares a common foundation mat. In the reference HTR-Module design, the bottom of the mat is located approximately 15.5 meters below grade and the top of the mat is located about 12.7 meters below grade. The Reactor Auxiliary Building is a separate structure; however it abuts the reactor building. The bottom of the Reactor Auxiliary Building mat is about 11.5 meters below grade and the top of the mat is about 9.5 meters below grade. This close configuration requires that both buildings be considered together from a construction standpoint.

9.1.1.1 Overview

The Reactor Building is structurally a "building within a building" as seen in Figure 9-1. The Reactor Building includes two spectacle shaped concrete support and shielding structures for the reactor pressure vessels and steam generators. The massive support and shielding structures are surrounded by a rectangular building that contains various rooms and passages housing equipment, piping and electrical services as well as HVAC and other building services. The "interior" building is a combination of concrete chambers as well as structural steel framing as required to meet the shielding and ventilation requirements in the various locations. The "exterior" building shares a foundation mat with the interior building and structures; however the vertical surfaces of the two structures are separated from one another by a small "rattle space." The "exterior" building is designed for aircraft impact and has concrete walls and a concrete roof, each about 1.3 meters thick. These "inner" and "outer" buildings are depicted in Figure 9-3.

The Reactor Auxiliary Building is a rectangular concrete and steel structure that contains various process equipment rooms in the lower elevations, locker rooms and changing rooms on an upper elevation with HVAC and building service equipment on the top elevation. This building is also used to control entrance into the Reactor Building.

Figure 9-1: Reactor Building Configuration



9.1.1.2 Construction Methods

It appears that concrete placement in the Reactor Building and Reactor Auxiliary Building can likely be accomplished by a combination of conventional forming techniques, slip forming and modular leave-in-place forming systems as further described in Sections 9.1.1.2.2 and 9.1.1.2.3. It would appear that in the Reactor Building, about 25% of the total volume of concrete could be placed using slip forming techniques, 50% using leave-in-place modular forming systems and about 25% using conventional forming. In the Reactor Auxiliary Building, it would appear that about 60% of the concrete could be placed using leave-in-place modular forms with the remaining 40% using conventional forms.

While many elevated floor slabs could be formed with leave-in-place modular forms, they still require rebar for strength unlike the leave-in-place modular forms used for walls that are sandwich panels with interior bracing that does not generally require rebar for structural strength.

9.1.1.2.1 Effects of Subsurface Conditions

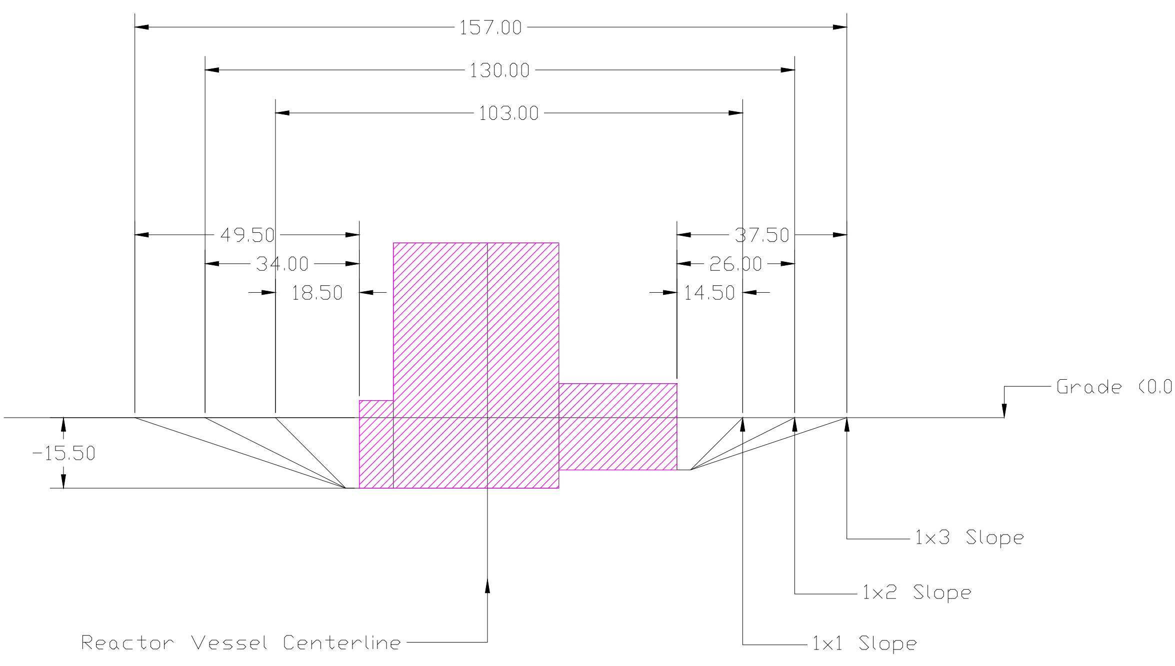
In the HTR-Module design, both the Reactor Building (including the Reactor Building Annex) and the Reactor Auxiliary Building are founded well below grade with the base mat of the Reactor Building about 15.5 meters below grade. For the purposes of this review, the facility is located somewhere along the southern coast of the United States bordering the Gulf of Mexico. The selection of the specific site as well as the final decisions that are made regarding embedment depth will play a large part in the methods that must be employed to excavate, dewater and prepare the area where these buildings are to be located. With the project sited on Gulf Coast it is somewhat likely that that the geology will include loose soils that will have to be “over excavated” and partially backfilled with structurally suitable materials before any building construction can take place. It is also very likely that because of the working elevation below grade, significant dewatering will be required to maintain a dry work area during construction.

Additional issues arise if these structures were to be sited near existing structures such as within the confines of an existing industrial complex in that dewatering could affect the existing foundations. This would require additional measures such as slurry walls to avoid disturbing the existing foundations with the dewatering operations.

Composition and stability of the soils in the area will have a profound affect on the overall dimensions of the excavation that will be required. In a typical Gulf Coast location, it is very likely that a slope of 1 (vertical)/2 (horizontal) or even 1/3 will be required to provide a safe working environment within the excavation. This could result in an excavation measuring from about 130 meters x 130 meters to nearly 160 meters x 160 meters (see Figure 9-2 below) or even greater if over excavation is required to improve the native soils.

In addition to the volume of material that has to be removed and replaced, the large excavation will have an effect on the sizes of the cranes required during construction since greater distance from the crane to the load reduces the crane's lifting ability.

Figure 9-2: General Excavation Requirements for the Reactor and Reactor Auxiliary Buildings



9.1.1.2.2 Opportunities for Slip Forming

Slip forming is a concrete placement process that uses either a continuously moving or a periodically jacked form system. Slip forming can be used in many applications but is a more effective technique when constructing relatively simple, tall structures such as chimneys. The spectacle shaped reactor and steam generator support structure is a potential candidate for slip forming since it is a relatively consistent profile for much of its height; however since significant amounts of large rebar will be required for this heavy walled structure it is not likely that rebar installation could progress fast enough to support a continuously moving form system and a periodically jacked form system may have to be employed.

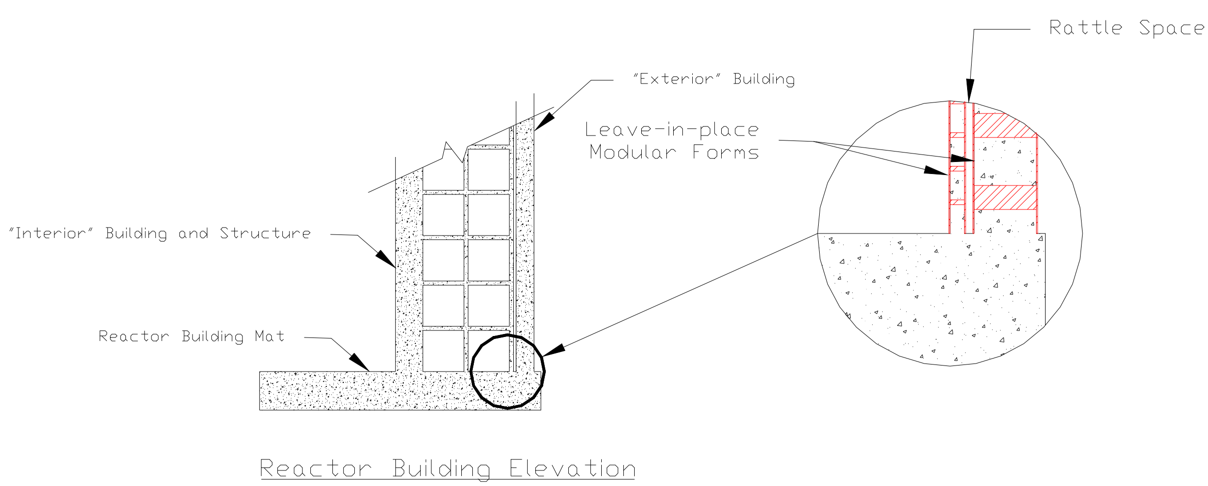
9.1.1.2.3 Opportunities for Leave-in-Place Modular Steel Forms

This concrete forming system employs sandwich panels of steel plates that are connected together with steel members. The forms are manufactured in an offsite shop and shipped to the site in large modules, the sizes of which are determined by a variety of factors including shipping and rigging considerations. In some cases, smaller modules may be further pre-assembled into larger assemblies at the jobsite prior to installation and can be formed into extremely complex shapes when required. Once the modular leave-in-place forms are installed in their final location, they are welded to the adjacent modules and then filled with concrete. When used for vertical surfaces, the forming system of plate surfaces and the interior bracing generally does not require any rebar. Rebar is required for horizontal surfaces and at certain interfaces such as between a foundation mat and a wall.

The exterior building of the Reactor Building appears to be an excellent candidate for the use of the leave-in-place modular steel form system. It is an independent structure with limited interaction with the interior building. The modular leave-in-place forms would also allow the exterior building to be constructed after the interior structures and building as well as minimize the "rattle space" between the two parts of the structure since no working space is required to remove forms. Figure 9-3 shows an elevation section of the Reactor Building and how the modular leave-in-place forms can be applied. Constructing the exterior building after the interior building has been completed should also provide additional opportunities to install larger modularized equipment in the interior building since it would afford additional access routes into the structure that would not be available if the exterior building were to be constructed first. This will be discussed in greater detail in Sections 9.1.1.3 and 9.1.1.4.

Certain walls and floors within the interior building of the Reactor Building also appear to be candidates to be constructed with modular leave-in-place forms. Elevated floors would generally use the leave-in-place forms to form the ceiling of the room below and require a rebar mat. The surface of the floor would then be cast and toweled similar to any conventional concrete slab floor.

Figure 9-3: Modular Leave-in-Place Forms Applied to the Reactor Building



The exterior concrete walls as well as many of the interior concrete walls in the Reactor Auxiliary Building could be constructed effectively with modular leave-in-place forms. Again, in some cases additional strengthening may be required either in the fabrication facility or in the field to accommodate piping and other loads that may be required to bear upon the walls.

9.1.1.2.4 Embedments for Equipment and Piping

Fastening of piping and other loads to concrete surfaces presents certain challenges and requires significant early planning in buildings with concrete walls and ceilings. Embedments cast into concrete result in fixed locations for attachments early in the building construction process and provide very little flexibility for future changes. Drilled-in expansion anchors also present installation challenges because it is difficult to locate and avoid rebar in a concrete wall while drilling for the anchor. The cast-in-place forms present a solid steel surface to which attachments can be welded; however, in some cases additional strengthening or stiffening of the surface may be required. This may be accomplished either in the fabrication facility if the high stress area is known or in the field to accommodate piping and other loads that may be required to bear upon the walls.

9.1.1.3 Construction Sequence

Due to their proximity to one another, the interaction of the Reactor Building and the Reactor Auxiliary Building construction sequences will have to be considered very carefully. The Turbine Building and the Switchgear and Emergency Supply Buildings, while having significant embedment depths appear to be spaced far enough from the Reactor Building to be considered independently to a large degree. Figure 9-4 and Figure 9-5 show elevation views of the relative positions of these buildings.

Figure 9-4: Elevation View – Relationship of the Reactor Building to the Reactor Auxiliary Building and the Turbine Building

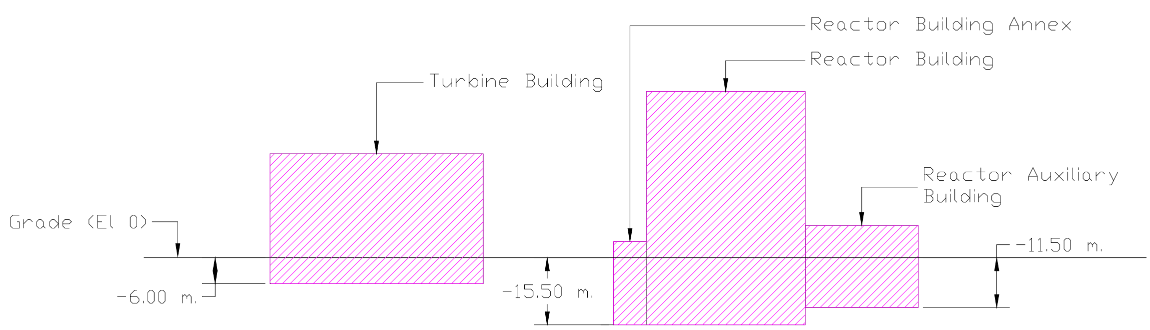
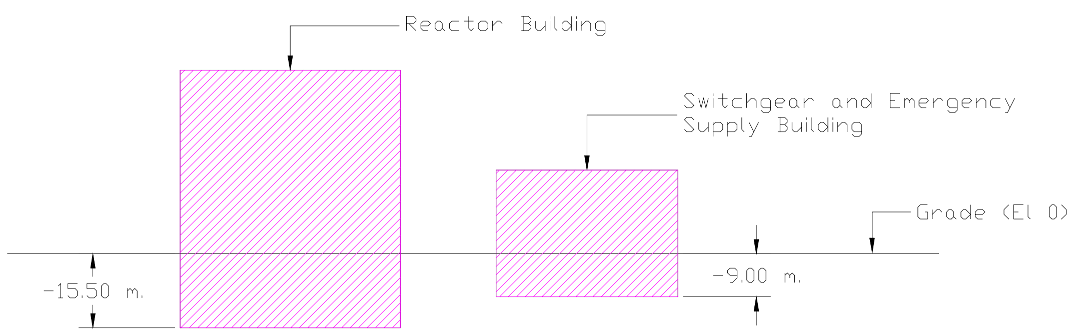


Figure 9-5: Elevation View – Relationship of the Reactor Building to the Switchgear and Emergency Supply Building



As discussed in Section 9.1.1.2.1, the excavation required to construct the Reactor Building and Reactor Auxiliary Building will be dependent upon the specific geological conditions at the site selected as well as the Reactor Building embedment depth decided upon to suit other conditions such as radionuclide shielding and seismic considerations. Once excavation and soil improvement has been completed, the Reactor Building mat can be constructed. It is anticipated that this mat will be conventional reinforced concrete about 2.5 to 3 meters thick, and will likely be cast as one mass concrete placement. After the Reactor Building mat concrete has been placed and cured, construction of the Reactor Vessel/Steam Generator concrete support structures can begin using a jacked forming system. This would be followed by construction of the concrete portions of the “interior” building of the Reactor Building using leave-in-place modular forms to the greatest extent possible. The design should also employ interior steel structures rather than concrete structures wherever possible. It is anticipated that certain access openings can be engineered into this structure to allow certain equipment modules to be inserted into the structure prior to erection of the “exterior” building of the Reactor Building. It is necessary to complete rigging of the large reactor vessels and steam generators into their final locations before the “exterior” building roof has been installed and consideration must be given to completing backfill around the Reactor Building before this rigging operation is attempted as it may reduce the amount of preparation that is required to site the required heavy lifting equipment. A detailed rigging study will be required to make the final rigging determinations.

The Reactor Building contains two complete nuclear reactors. Due to shielding as well as worker safety and operational considerations, it appears likely that the entire Reactor Building have to be completed before the plant can become operational.

9.1.1.4 Equipment Modularization opportunities

In general, smaller equipment components, piping and cable tray banks (without cable installed) as well as some structural components such as stair towers are good candidates for assembly into larger modules at a fabrication facility. The module size is dependent upon certain conditions including the ability to ship the module to the jobsite as well as the locations in the structures where the modules are to be installed. The current HTR-Module design of the “interior” Reactor Building appears to consist of numerous rectangular shaped passages designated to contain piping and equipment or act as personnel access passages as seen in Figure 9-3. At this time, insufficient design details are available to determine what portion of these passages are totally encased on concrete and what portion may be structural steel framing. By carefully designing these passages and leaving access openings, it would be possible to modularize some of the piping runs, cable tray runs and small system components (pumps, motors, small tanks, etc.) and rig the modules into the “interior” structure before the “exterior” structure is constructed.

9.1.2 Turbine Building

The Turbine Building is a conventional design with the turbine generator located atop a 9.5 meter high pedestal. The low pressure steam turbine exhausts downward into a condenser that is located beneath the pedestal. The remaining area below the pedestal is used to house piping as well as the mechanical equipment and electrical equipment to support operation of the turbine generator. The area above the turbine generator is open space reserved for lifting the large components of the turbine generator with the permanently installed bridge crane. While the original HTR-Module design appears to show a “tuned” pedestal with light weight steel legs, an American installation would more likely employ a more massive total concrete pedestal design.

9.1.2.1 Construction Methods

Generally, conventional construction methods will be employed in constructing the Turbine Building foundations and structures. There is significant history of Turbine Building construction with little indication that any savings in cost or schedule can be realized using other than the already proven methods of construction.

9.1.2.1.1 Opportunities for Slip Forming

The Turbine Building is essentially a large concrete mat and a concrete pedestal surrounded by a steel frame structure with no opportunity to employ slip forming.

9.1.2.1.2 Opportunities for Leave-in-Place Steel Forms

The only portion of the Turbine Building that may have any application for leave-in-place steel forms is the turbine pedestal legs if concrete rather than tuned steel legs are used. This represents about 10% or less of the total concrete volume.

9.1.2.1.3 Embedments for Equipment

Generally, concrete embedments (other than machinery base plates) are not required in a conventional Turbine Building since most of the piping and other services are fastened to the structural steel building framing.

9.1.2.2 Construction Sequence

As seen in Figure 9-4, the Turbine Building has a shallow foundation (6 meters below grade) relative to the Reactor Building (15.5 meters below grade). Horizontal separation between the two buildings appears to be sufficient to allow both building foundations to be constructed at the same time. The Turbine Building will likely be constructed by first installing the foundation mat followed by construction of the turbine pedestal. If the turbine generator is not available early in the erection process, it will be necessary to leave an unimpeded access route to the top of the turbine pedestal for future installation of the turbine and generator. Once the turbine pedestal is completed, the steel frame building can be built around it, after which mechanical and electrical equipment can be installed.

9.1.2.3 Modularization Opportunities

Much of the equipment located in the Turbine Building is already packaged by the vendors into skids or small modules. Some limited opportunities do exist to modularize low energy pipe runs such as lubricating oil, service water, or component cooling water. However, much of the high energy piping within the Turbine Building cannot likely be routed together with other piping orderly banks for stress considerations, and therefore is not a likely candidate for modularization. There is potential for modularizing some cable tray banks where multiple cable trays are routed together, and some structural components such as stair towers could be factory assembled and shipped as modules. Both cable tray banks and stair towers have been successfully modularized on other construction projects.

9.1.3 Switchgear and Emergency Supply Building

The Switchgear and Emergency Supply Building is a multi-floor concrete structure that contains the main control room, the electronic cabinets that house the instrumentation and control systems and electrical switchgear. Space has also been allocated within this building for the emergency diesel generators.

In the HTR-Module design, the control room layout and equipment rooms appear to be sized and arranged to support control systems that were in use in an earlier time and are now obsolete. Older control systems required cabling of control and instrumentation signals to a central location or directly to the main control board as well as between devices to provide the necessary control logic. The layout does not appear to consider modern distributed control systems that employ remote input-output cabinets and communicate with the central electronic systems to provide control logic and use digital signals and fiber optic technology to communicate between the various portions of the system. In addition, modern distributed control systems (DCS) have reduced the size requirements for the control room since most of the plant control is accomplished digitally with a few computer monitors and pointing devices rather than hundreds of “hard wired” control switches, indicator lights and analog meters requiring large control boards.

9.1.3.1 Construction Methods

The construction methods that are to be employed will be dependent upon the final configuration that is required once all factors have been considered. Generally, it would appear that a certain amount of masonry construction will be required to provide safe separation of redundant control systems as well as for fire safety within the building; however it is likely that certain portions of this structure can employ steel framing with metal exterior siding and drywall or other interior wall paneling material forming interior walls. All below grade portions of the building would be concrete.

9.1.3.1.1 Effects of Subsurface Conditions

As seen in Figure 9-5, the Switchgear and Emergency Supply Building has a moderate depth foundation (about 9 meters below grade) relative to the Reactor Building, which is 15.5 meters below grade. Horizontal separation between the two buildings appears to be sufficient to allow both buildings to be constructed at the same time. The Switchgear and Emergency Supply Building will likely be constructed by first installing the foundation mat followed by construction of the exterior walls up to just above grade. The foundation mat is somewhat complex because it includes tunnels to route cables to the Reactor Building and elsewhere. Since the specific site geology is not known at this time, it is also unknown if any over-excavation and replacement with structural fill or other soil improvement will be required.

9.1.3.1.2 Opportunities for Slip Forming

The design of the Switchgear and Emergency Supply Building does not appear to lend itself to slip forming.

9.1.3.1.3 Opportunities for Leave-in-Place Steel Forms

Leave-in-place steel forms may be able to be employed to some advantage in construction of the foundation walls and cable tunnels. Above grade, a steel framed building is anticipated with concrete floor slabs. Where masonry walls are required for separation or fire protection in the upper elevations it is anticipated that cement block construction can be employed. It would appear that leave-in-place modular forms could be used for up to about 40% of the total volume of concrete required.

9.1.3.1.4 Embedments for Equipment

Equipment installed in the Switchgear and Emergency Supply Building is limited to electrical and electronic component as well as building service equipment (i.e., HVAC). Much of the equipment is mounted on housekeeping pads on the floors, which are installed after the concrete for the floors is placed. Some medium voltage switchgear employ roll-in circuit breakers that require the bottom of the switchgear be flush with the floor. This equipment is generally fastened to steel base plates that are embedded in the concrete floor. Both of these systems of attachment are easily modified after initial installation to accommodate later changes to the equipment.

Cable trays located in the concrete cable tunnels and the concrete below grade portions of the Switchgear and Emergency Supply Building will require embedded fastening systems to secure them to the structure. This can easily be accomplished by embedding commercially available metal framing systems such as those manufactured by Unistrut® into the concrete in a regular pattern if conventional concrete forming is used. If a modular leave-in-place forming system proves to provide additional construction and cost advantages, a pattern of the metal framing channels can be welded to the steel surfaces of the forms in the manufacturing facility as part of the fabrication process.

9.1.3.2 Construction Sequence

Construction of the foundation mat and underground cable tunnels would first be completed followed by construction of the concrete portions of the walls. This would be followed by erection of structural steel and construction of floor slabs, completion of interior and exterior walls and installation of the roof. Openings must be left in the walls to allow the equipment to be installed.

9.1.3.3 Modularization Opportunities

Modularization opportunities in the Switchgear and Emergency Supply Building as currently envisioned appear to be limited mostly to construction of the cable tunnels, cable tray banks and stair towers. Modularization of other electronic or electrical components does not appear to have any consequential construction or cost benefit at this time.

9.1.4 Spent Fuel Store

The Spent Fuel Store appears to consist of a shallow concrete mat or slab with rows of concrete boxes to contain the spent fuel casks installed at grade. The structure also includes a permanently installed bridge crane to load and unload the spent fuel casks into the concrete boxes and install the concrete box covers.

9.1.4.1 Construction Methods

9.1.4.1.1 Effects of Subsurface Conditions

This is a light weight structure on a shallow concrete slab. It is anticipated that minimal excavation, preparation and dewatering will be required regardless of the geological conditions in the area.

9.1.4.1.2 Opportunities for Slip Forming

Slip forming does not appear to be suitable for this application.

9.1.4.1.3 Opportunities for Leave-in-Place Steel Forms

Construction of the rows of concrete boxes appears to be a possible application for leave-in-place steel forms. Each box is about 10 meters high and 5 meters wide with a wall thickness of about 0.5 meters. Due to the amount of welding required to join the form sections together, a detailed economic evaluation would be required to determine if conventional forming and rebar installation would be less costly than the leave-in-place steel forming system, which does not require rebar other than doweling at the connections of the walls of the boxes to the base slab. Leave-in-place modular steel forms could potentially be used to form about 50% of the total volume of concrete.

9.1.4.1.4 Embedments for Equipment

None required.

9.1.4.2 Construction Sequence

This is a simple structure consisting of a concrete slab with a series of concrete boxes and a structural steel runway to support a bridge crane. The slab will be constructed first followed by the structural steel runway for the bridge crane. If the bridge crane were to be installed early, it could be used to support erection of the concrete boxes.

9.1.4.3 Modularization Opportunities

There are no other opportunities for modularization in this structure other than the possible use of leave-in-place steel form modules to form the concrete boxes to contain the fuel casks.

9.1.5 Other Structures and Installations

The remaining facilities consist of conventional components and buildings including cooling towers, tanks, miscellaneous yard structures and foundations.

9.1.5.1 Construction Methods

9.1.5.1.1 Effects of Subsurface Conditions

Most if not all of the remaining structures will be relatively light weight and supported by relatively shallow foundations; therefore they will likely require only minimal excavation, ground preparation, soil improvements and dewatering.

9.1.5.1.2 Opportunities for Slip Forming

There are no apparent opportunities for slip forming in the remaining structures.

9.1.5.1.3 Opportunities for Leave-in-Place Steel Forms

There are appears to be no significant opportunities for leave-in-place steel forms in the remaining structures.

9.1.5.1.4 Embedments for Equipment

No significant equipment embedments are required in the remaining structures other than conventional foundation attachments for machinery and equipment.

9.1.5.2 Construction Sequence

Mostly conventional construction techniques will be employed; therefore conventional construction sequences for the various items will be employed.

9.1.5.3 Modularization Opportunities

Some modularization opportunities may exist in the gas supply system building as well as the cooling towers. Additional modularization opportunities may exist for the installation of overhead and underground cable tray banks and pipe racks that are run between buildings as discussed in more detail in Section 9.2.1.1 and 9.2.2.2.

9.2 System Design and Installation

9.2.1 Mechanical Systems

Some mechanical system components, primarily in the Reactor Building and the Reactor Auxiliary Building have the potential to be modularized to a certain degree. This will be largely dependent upon the design of these two buildings and if it will allow the large modules to be "loaded" into the building during the erection sequence. A system of openings must be left in the structures in line with the final locations of the modules to make this an effective method of installation. Once the exterior walls of the two buildings are constructed it will be impossible to rig the large modules into most locations.

9.2.1.1 Potential Modularization of Piping between Buildings

Elevated pipe racks between the buildings are excellent candidates for modularization. The module framing can be designed to become the steel framing of the pipe rack and large modules can be rigged into place with relative ease.

In certain underground locations where pipe runs are located in tunnels, piping could be installed on steel framing similar to an above grade pipe rack and rigged into the tunnel after the floors and walls have been installed but before the roof has been installed. This installation could also benefit from leave-in-place steel forms, in that wall forms and steel framing for the pipe supports could be factory fabricated into a module that is rigged onto the foundation mat and the wall forms filled with concrete. The module could also include left-in place formwork for the ceiling, and could even include the rebar required to reinforce the roof. Figure 9-6 shows the cross section of a completed pipe tunnel and Figure 9-7 shows a pipe tunnel module installed on a concrete base prior to placement of concrete in the walls and ceiling. Bracing required for rigging and certain details have been omitted for clarity.

Figure 9-6: Cross Section of a Completed Pipe Tunnel using Modular Construction

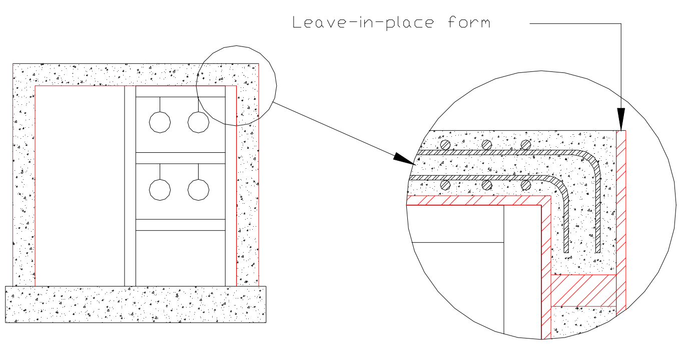
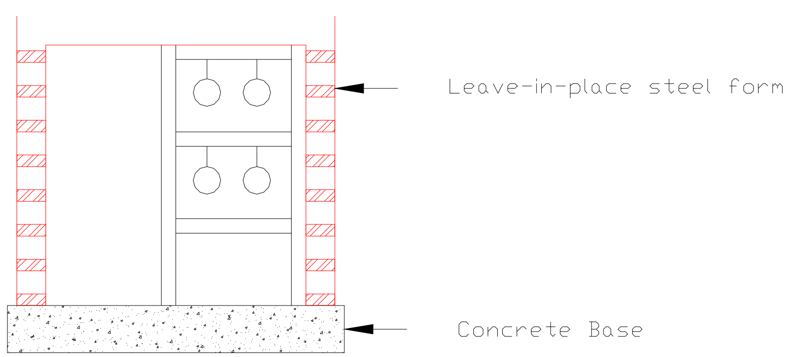


Figure 9-7: Pipe Tunnel Module (temporary bracing not shown)



9.2.2 Electrical Power and Control Systems

It is anticipated that state of the art technology would be employed for the electric power and control systems. This would include current design switchgear, power distribution panels and motor control centers as well a distributed control system to control and monitor all plant functions.

9.2.2.1 Modernization of the Control System from HTR-Module Design

The HTR-Module design appears to employ an old technology control system with many of the control functions directly wired to control switches on a large main control board. A modern plant is typically controlled with a distributed control system (DCS) with the human interface being a group of computer monitors and pointing devices. The DCS architecture includes but is not limited to:

- Input/output (I/O) cabinets to interface with electrical switchgear, various field sensors and other devices that require control signals;
- Digital signal data processing equipment to provide the control logic;
- Control room operator workstations for primary central control of the plant;
- Engineering workstations for programming an system monitoring.

A DCS will pass data between the various components via data highways, which would likely be a combination of fiber optic cables and copper cables designed for digital data transmission.

By strategically locating I/O cabinets in close proximity to groups of signal sources or devices that are being controlled, the amount of field-installed cable required is reduced significantly from conventional (and now mostly obsolete) “hard wired” control systems, which directly wire various devices and switches together in a certain sequence to form the control logic, to the digital processing that is now in common use. I/O hardware is now being integrated directly into switchgear, motor control centers and other equipment by the manufacturers permitting digital data communication over fiber optic or copper data highways between the switchgear and the central control system thereby reducing numbers of individual field cables required and reducing the field installation labor required.

9.2.2.2 Modularization of Electrical Raceway Systems

Electrical raceway systems include electrical conduit, cable trays and underground duct banks. Typically cable trays are used for longer cable runs and where large quantities of cable must be routed in the same direction for a large portion of the run while electrical conduit is typically used to support cable between cable tray and equipment or where additional physical protection of the cable is required. Underground duct banks are employed where a limited number of cables must run between buildings or outdoor equipment and where large quantities of cables must be run between two places, underground cable tunnels may be employed.

Where significant numbers of cable trays and/or conduits are to be routed together in banks, they can be assembled into modules that also form the support structures for the raceways. Buildings interconnected with overhead cable tray banks are excellent candidates for modularized cable tray installations, and certain cable tray banks within buildings may also be modularized as discussed previously.

Cable trays installed in underground cable tunnels could also be installed using a process similar to the modular underground pipe tunnels described in Section 9.2.1.1.

9.3 Transportation and Heavy Haul Considerations

Site location is a key element to be considered in the transportation of the large and heavy components of any facility. This may become the overriding factor in the feasibility of a project if the component shipping size and weight prohibits shipment from the manufacturing facility to the jobsite. Most of the concern is placed upon the project site since manufacturers of the large and heavy components would already have the required facilities in place to ship their products from their factories.

The largest component of the PBR appears to be the reactor vessel, which is estimated to measure about 25 meters high and 7 to 8 meters in diameter with a weight expected to be in excess of 800 metric tons. Transportation of this component over water either on an ocean going vessel or on a barge on a navigable inland waterway is a relatively simple task.

Off-loading of a large and heavy component from the ocean vessel can be accomplished in several ways and will be dependent upon the facilities that are available in the vicinity of the construction site. Ocean going vessel choices include large conventional haulers, self-loading heavy lift vessels with heavy lift cranes installed on the vessels, and roll-on roll-off vessels that allow heavy haul equipment a to drive the loads on and off the vessel in a manner similar to a ferry boat.

Land transportation of the large components, which includes road and railroad, presents the greatest challenge as established infrastructure may not be able to bear the loads or allow the load dimensions to pass. The fabricated reactor vessel as well as the core barrel exceeds absolute maximum railroad dimensions for height and width so this method of transport cannot be considered. The likelihood of finding an over road route to transport the large components decreases as the distance from the point where the load is offloaded from water transport to the site increases since the likelihood of encountering height, width, length or weight restricting obstructions increases. In some cases, convoluted travel routes as well as costly upgrades to roads and/or bridges may be required. In some locations, transport permitting authorities may not approve the transport and in other locations, the underlying geology of the road bed may severely restrict weights of the loads that may be transported.

From a heavy transportation prospective, the ideal site would be located on a waterway that can be navigated by ocean going vessels. It is possible that if the project were to be sited at or near an established heavy industrial facility that the marine infrastructure may already be in place to offload heavy and oversized loads.

The site location and its accessibility to over water transport will also affect the size and weight of pre-fabricated modules that can be transported to the site.

The largest components of the plant that will require heavy transport considerations include the reactor vessel; the reactor core barrel; large heat exchangers such as the steam generator; the turbine generator and the large transformers. Although the reactor vessel is the largest and heaviest component, if a viable shipping route for a completed assembly cannot be established, it may be possible to assemble the individual forgings that make up the vessel at site. A large and sophisticated fabrication facility with state of the art production welding equipment would be required along with the skilled workers to complete the fabrication. The reactor core barrel, while smaller than the reactor vessel, will likely present similar challenges. It appears feasible to also assemble the core barrel in a workshop on site. The steam generator, while expected to be about 22 meters in length, appears to be about 4 meters in diameter, which may be small enough to pass under many overpasses. The steam generator may also be transported on railroad (using a Schnabel car) if it meets the height, width, length and weight restrictions of the eventual route.

The turbine generator and transformers will likely be commercial models that have been engineered to be able to be more easily transported to a jobsite with only limited special handling considerations.

9.4 NOAK Plants Constructability Assessment

This section discusses a constructability assessment that was conducted to examine issues that might impact large-scale production of HTGRs. As indicated earlier, initial assessment concludes that the existing infrastructure should be able to accommodate the FOAK plant with early planning and execution of long lead procurements. As such, this constructability assessment is primarily focused on the follow-on NOAK plants.

This review is based on experience and lessons learned from the nuclear power plant construction in the United States in the 1970s and 1980s. The review also considers the current practices being used in Europe and China along with the plans for the Generation 3 nuclear power plants being proposed for North America. The constructability assessment is a general high level review, and the results presented are dependent on timing, location, and the final design.

9.4.1 Constructability Considerations

The conclusions of the Constructability Assessment are influenced by several basic conditions that are indeterminate at this stage of the project. Therefore, several observations are qualified based on the actual site conditions.

- The decision of building one unit versus multiple units and the time sequence of building multiple units obviously plays a major role in the excavation plan, the unit interfaces, and in the approach to security for the operating unit and the unit under construction. Other issues impacted include: attracting and maintaining labor, and mobilization and infrastructure cost.

- The type of foundation will determine the impact of subsequent units. Since the reactor pit could be of significant depth, excavation itself is a major consideration. (The proposed depth continues to be a variable, as the most cost effective elevation is studied.) Whether the excavation is in soil or rock is an obvious cost and schedule consideration. Blasting impacts, excavation, and backfill interfaces between units must be considered. The use of slurry walls or freezing the surrounding ground with liquid nitrogen to avoid excessive step-backs in the excavation are expensive techniques that can't be ruled out at this time.

- Labor availability is always a consideration. This will be further impacted by the length of the project (one unit versus four), open shop or closed shop (employment restricted to trade union members only), and the skills required on site.

- Whether the units are built continuously or spaced out over time, security of the existing operating unit must be considered. The approaches and requirements may vary depending on the final configuration and construction sequence.

- Firm regulatory requirements must be established and maintained in order to plan and execute the project. The regulatory requirements must be established early to be considered in the planning phase of the project.

9.4.2 Constructability

9.4.2.1 Potential Issues

9.4.2.1.1 Excavation

Typically the PBR concept will not consider a fully embedded configuration. However some issues related to excavation may warrant consideration. The reactor pit excavation depth presents several issues that must be addressed. The issues and their resolution will vary depending if the excavation is in solid rock, soft soil, or a mixture. Most sites will need to have a robust de-watering system (for construction and operations). De-watering sites with wet soil conditions may prove especially difficult.

The excavation at this depth may require terracing and sheet piling, along with extensive tie backs, to stabilize the embankments and allow access for excavation and construction. The excavation of subsequent units will impact the foundation support and backfill of the existing units.

If the excavation requires blasting, subsequent units will require a plan to minimize impact on the existing units. This could vary from minimum size blasting charges to “pre-splitting” blasting to isolate the resulting vibrations. Pre-splitting entails drilling blast holes the depth of the excavation along the pattern of the near walls of the next unit. The holes would be spaced to allow the force of the charge to “split” the rock from hole to hole along the future excavation line. This would be accomplished for each subsequent unit while the blasting was performed for the previous unit.

The approach to excavation needs to be studied in more detail for each site to avoid or minimize impact to the construction of critical path structures and existing units. The best approach from a construction point of view is to excavate all the reactor pits for a site before the first unit goes into operation. However, this may not be realistic from a logistics or economical point of view.

The consideration to raise the reactor and auxiliary buildings to reduce excavation presents the issue of more exposed structure to be “hardened.” With the potential need to “harden” certain structures, it may be determined that raising and hardening more reactor and auxiliary building structure is not overly difficult. If this is the case, the reactor and auxiliary building could be raised to an elevation that is most compatible with the interfaces with the power conversion system or client stem supply headers. Raising the structure may increase seismic and wind/tornado loadings; however, it is believed that the penalty for a deep excavation (construction cost and designing the walls for lateral soil loads) are higher than those for seismic, wind, and hardening if the elevation is raised. Of course, there are other functional and operational considerations that may dictate the desirable elevation.

9.4.2.1.2 Subsequent Units

Not only does excavation impact the construction approach for subsequent units, the interface of structures and systems must also be considered. The interfaces identified thus far include the turbine building, and the condenser cooling water intake and discharge pipes.

<u>Turbine Building</u>

Security considerations may not be as restrictive as other areas, but access control will still be required. The ventilation requirements and their interface/impact with other units must be considered in the construction approach. If the turbine building crane(s) are used for multiple units or moving heavy equipment or materials through the buildings is required, separate foundations could lead to differential settlements that are unacceptable. The structural steel may also need to be tied together.

Intake and Discharge Piping

The plant design will use either cooling towers to cool the circulating cooling water or a once-through circulating cooling water system with intake and discharge structures on a large surface water feature. The construction issues associated with the intake and discharge piping systems serving each module are similar for both the designs. For sites with multiple plants, the circulating cooling water intake and discharge piping will probably run from the intake structure through or near the series of turbine buildings to the discharge structure. The pipes may not specifically interfere with construction of subsequent units. However, there are several scheduling issues to be resolved as subsequent units are anticipated.

- Do the site conditions warrant initially building the intake and discharge structures large enough to support the total number of units? Although building the structures large enough for all anticipated units would require a capital investment in concrete structure, it is recommended these structures be initially built to handle all anticipated units. Each unit would need to be capable of isolation to allow the installation of pumps, valves and piping at a later time. This approach would be much less expensive from a construction view point.

- Should the intake piping for subsequent units be installed with the first unit? Again, this would require an early capital investment. However, it would avoid the construction disruption of the existing plant yard and access corridors caused by multiple excavation projects to install the pipe for subsequent units. If multiple units are anticipated, it would be more convenient to install the intake pipe for all units with the first unit.

- The intake piping for subsequent units that is located under previous units must be installed as each unit is constructed.

- The discharge pipe for existing units must be supported to allow the excavation necessary for installation of the discharge pipe for subsequent units. Adequate spacing of the pipe must be considered in the design.

- Similar to the intake pipe, should the discharge pipe between the units and the discharge structure for subsequent units be installed with that of the first unit? The installation of these discharge pipes is not as disruptive as the intake. They do penetrate the security fence, which must be considered.

- If the pipe and structures for subsequent units are installed, special care needs to be taken to isolate them from the water. And there needs to be a plan to “dewater” the structures to complete the piping and remove the isolation barriers.

From a construction perspective, it would be best to construct the intake and discharge structures along with the associated yard piping with the first unit. This approach would avoid large excavations across existing roads, pipes, buried cables, security fences. It would also reduce the overall cost associated with adding onto an existing structure.

9.4.2.1.3 Security

Security requirements for operating nuclear stations continue to be developed as vulnerabilities are determined. Requirements for security during construction are also receiving more attention. If construction workers are required to have background checks and security clearances, the physical security between operating units and units under construction may only be access monitoring and work orders versus actual control barriers.

It may be more economical to require security clearances for the core construction labor and escorts for short term subcontractors. The interior yard fencing could be access control versus intruder detection. The turbine building walls could be designed to meet the HVAC requirements with minimum access control requirements for personnel. The tunnels may require additional barriers for radiological protection and restricted access control. This could be accomplished with a reinforced and monitored concrete block wall between units under construction. If the period between construction of subsequent units was extended, a concrete barrier could be constructed to seal the tunnel (in addition to the block wall). The intake and discharge piping would need similar security barriers.

A more difficult issue is the potential need to harden certain above ground structures. The construction issues with the depth of excavation were discussed previously. Raising the reactor and auxiliary buildings to reduce the excavation depth, presents more issues with hardening these structures. Assuming that these structures will generally remain below ground, hardening the enclosure possibly could be accomplished with a concrete structure instead of a structural steel structure. The concrete structure could resemble more of an igloo shape versus the traditional cylinder and dome structure. Such a configuration may limit the access between units for the fuel crane.

9.4.2.1.4 Equipment Access

Special consideration must be given to the method of installation of the reactor vessels and the steam generators. Two options have been identified for installing the reactor vessel: (1) final assembly in a remote or on-site fabrication shop and then installation in one piece inside the reactor cavity; (2) final assembly within the reactor cavity itself. A large construction crane will be used to install the reactor vessels and steam generators. The sequence of installation and the construction of the surrounding buildings will determine the size and setup location of the installation crane. This does not appear to be a major issue, but to minimize impact on the reactor building and auxiliary building progress, this sequence needs to be detailed and considered in the design. Large surcharge loads could impact the design of the outer walls of both the reactor and auxiliary buildings.

The same issue may exist for other equipment in the auxiliary building that needs to be installed with the progress of the concrete.

9.4.2.2 Potential Opportunities

As with any major construction project, early constructability input to the design provides the opportunity to optimize the construction cost and schedule. Some of the opportunities include:

9.4.2.2.1 Pre-Assembly of Components

Optimization through pre-assemble of components, pre-fabrication, and modularization. In general, this approach is minimizing the piece by piece construction in the confinement of the buildings and the critical path and optimizing the amount of work that can be accomplished away from the structures and off the critical path. There is not one magic answer to optimization. However, there is consensus in the belief that the more work that can be accomplished out of the power block or away from the site, the larger the opportunity to save cost and schedule. Each opportunity to optimize has its own benefits and pitfalls, and therefore must be evaluated individually. Some of the benefits include the following:

- Critical path schedule improvement
- Better productivity
- Reduced labor cost
- Reduce site labor
- Improved safety and quality
- Production line capability for next unit

There are some pitfalls to be avoided, such as the following:

- Early commitment of expenditure
- Additional engineering cost
- Additional support and framing steel
- Extra transportation cost
- Configuration control during transportation and storage

Pre-assembly of components includes reinforcing steel mats and cages, liner plate assemblies, welding valves to equipment before installation, and completed electrical cabinets. Pre-fabrication is larger pipe spools, steel frames and platforms, and cable tray modules. The conservative approach to modularization is more and larger equipment/system skids. Installing a completely "dressed-out" Reactor Vessel head assembly in a LWR is an example of a module. Modularization could be expanded to composite concrete and steel assemblies barged to the site and heavy lifted into position. This more liberal approach will require considerable study, investment in engineering, and early funding. The more liberal approach is not proven and introduces additional risk into the project.

9.4.2.2.2 Open-Top Construction

"Open top" construction is the approach of installing the previous discussed optimization techniques with the structure as it is being built. This approach allows the benefits of optimization to be realized to their maximum extent. Conventional construction has always installed large tanks before the next concrete floor was placed. "Open top" construction simply expands that approach to allow bulk quantities and system installations to proceed where the concrete structure is being constructed. This allows the construction schedule to be expedited.

9.4.2.2.3 Pre-Fabricated Concrete

There is also the opportunity to explore prefabricated structural concrete walls and floor where the metal formwork becomes part of a composite modular design. Once the modules are set in place, concrete can be placed inside the composite module. This choice of design and construction must be implemented in the conceptual phase of the project to be successful. This is another way to optimize the amount of work that can be accomplished away from the structures and off the critical path further reducing on-site labor and taking work off of the critical path.

9.4.2.2.4 Power Block Design

If there is an option to build multiple units, the interfaces discussed under Constructability should be considered in the design of the first unit. This will avoid back-fitting and allow a more cost effective approach.

The Project Team should concentrate on the "power block" design and construction and subcontract the conventional and specialty buildings and systems. It is recommended that these buildings and systems be subcontracted on a fixed price basis to design/build firms that specialize in the particular type of building or system. This will enhance the opportunity to subcontract to more local and minority firms (U.S. consideration).

9.4.3 Techniques and Sequence

The critical path for last generation of nuclear plants started with the reactor building concrete and proceeds through the piping and electrical systems. After reviewing the interface between the reactor building and the auxiliary building, there is no reason not to expect the same for the HTGR. However, additional details need to be studied to schedule the buildings adjacent to the reactor building. The reactor building foundation could be over 45 meters below grade, which will monopolize the schedule for several months. The adjacent buildings will not be able to start until the reactor building concrete is near complete, cured, and backfilled. Some early work on these structures may be feasible on the sides away from the reactor building and though preassembly, prefabrication, and modularization.

The schedule for system completion needs to match the turnover schedule for operations start-up. Initially piping and electrical cables will be installed in bulk and by area. System completion will probably start at about 75-80% bulk installation completion. Special consideration with technique and sequencing needs to taken with the large bore main steam piping, the reactor vessel cavity cooling pipe, and the associated structural steel supports.

Open Top Construction- is the practice of installing as much pipe, equipment, skids, tanks, and material as possible as the structure is being built before the next floor is installed. This allows the opportunity for the system work to start early. Properly planned this approach will save cost and schedule. It also will allow more work to be accomplished off site and off of the critical path.

Early planning for pre-assembly, pre-fabrication, modularization, and multi-purpose skids is critical to allow design to accommodate the "Open Top" approach. Larger and more integrated pieces may require additional support structure for handling and transportation. The design must also consider erection and maintenance access. One of the most critical design issues will be designing the support shoring system for the concrete above. Each design will be special since the floor space available for shoring will vary with the type and amount of materials and equipment being installed early.

Scheduling the remaining support systems and buildings off the critical path and spreading them across the schedule will help minimize the workforce peak on site.

9.4.4 Labor Initiatives

The nuclear renaissance being created by the deployment of the Generation 3 nuclear power plants has highlighted a major concern and risk in the availability of an experienced construction workforce. Several initiatives are being adapted by the industry to mitigate this concern. Industry leaders are working with their local community colleges and vocational schools to create interest and develop the skills required to support the renaissance. Engineering schools are seeing a renewed interest in the degrees associated with heavy industrial projects. Contractors are engaging the trade unions early in the planning process for proposed plants to ensure alignment. In addition to these types of local efforts, the industry needs to extend these types of initiatives across the country to reach suppliers, fabrication facilities, engineering firms, and other support groups.

Most noted is the concern for skilled craft workers such as welders and electricians. However, there should be equal concerns to attracting experienced construction managers, engineers, and technicians. There was a great sense of pride in the construction industry during the boom in the last generation of nuclear power plants. Re-establishing this type of interest in construction as an occupation is necessary to ensure an adequate workforce for the future development of nuclear power.

Generation 3 nuclear power plants are expected to renew the interest in construction that should result in an experienced workforce for the NOAK. It is that renewed interest that the NGNP must sustain and grow, not allowing the industry to digress. This task will be more likely to be successful if it is an industry initiative.

9.4.5 Construction Pre-Planning Activities

There are several follow-up activities that are recommended to enhance a constructability review of the proposed NGNP.

- Once the decision is confirmed on the selected design, number of units and the depth of excavation required, a more detailed excavation plan can be developed to determine the recommend approach for excavation and backfill.
- There is a need to determine the requirement and method of "hardening" the structures. As these studies materialize, constructability reviews need to conducted.
- Once the elevation of the reactor building is finalized, a detailed major equipment erection and crane plan needs to be developed. Although there does not appear to be an access issue, specific crane and external loads on the perimeter walls needs to be considered in their design.
- To take full advantage of any optimization opportunities to pre-assemble, pre-fabricate, and modularize components and system, the opportunities must be considered in the design. Construction and Design should identify potential opportunities early in the design process. This will enhance the "open-top" construction approach.
- To complete the constructability review, a detailed schedule of sequence of construction would be beneficial. Recommend that a Level III schedule be developed.

9.5 Conclusions

As with any major construction project, there are risks and opportunities. The key is to identify both the risks and the opportunities early and plan their fate during the design phase of the project. Although there are challenges with the construction of the HTGR, we do not believe there are issues that extend beyond those experienced and resolved on similar projects. The current challenges with new-build nuclear projects have and will continue to prepare the industry for the deployment of the NGNP.

9.5.1 Building Design and Construction

Ease and cost of construction will be dependent upon several factors including embedment depth, and the ability to provide a design that can take advantage of a certain amount of modularization and leave-in-place forming systems, particularly where the leave-in-place forming systems can replace the massive amount of large diameter rebar that is found in conventional nuclear structures such as reactor containments on light water reactors and provide a more efficient method of fastening equipment to these walls and ceilings without the need for conventional embedment plates and drilled-in concrete anchors.

These techniques must be carefully considered during the detailed design phase and should result in reduced construction costs.

Table 9-1 below shows the approximate percentages of the volume of concrete in each structure that could potentially be placed using the various forming systems.

Table 9-1: Application of Concrete Forming Methods in Buildings and Structures

	Leave-in-place Modular Forms	Slip Forms	Conventional Forms
Reactor Building	**50%**	**25%**	**25%**
Reactor Auxiliary Building	**60%**	**-**	**40%**
Switchgear and Emergency Supply Building	**40%**	**-**	**60%**
Spent Fuel Store	**50%**	**-**	**50%**
Other Structures and Installations	**-**	**-**	**100%**

9.5.2 Affects of the Design on Construction Schedule

9.5.2.1 Multiple Reactors in One Structure

Locating multiple PBR reactors in a single Reactor Building, while reducing some installation costs because the reactors can share certain common structures and services also reduces the flexibility of the construction process since both nuclear reactors must be completely installed before either reactor can operate. The construction progress of both reactors must progress nearly simultaneously so that both are completed at about the same time. Much of the erection process can progress simultaneously for both reactors; however certain operations such as rigging the reactor vessel and steam generators will likely be completed sequentially due to the availability of the required lifting equipment.

9.5.2.2 Common Services

The use of certain common services will likely reduce the overall cost of installation.

9.5.2.3 Equipment Modularization

Modularization of certain equipment and hardware may provide some savings in site erection time; however the benefit must be weighed carefully against the cost of preparation and shipping of the modules as well as providing structures and a construction sequence that will support the modules. Modularization of pipe racks, cable tray racks, and stair towers that are to be located in accessible above areas around and between buildings should prove to be beneficial while modularization of the same items located within buildings may prove to be less beneficial due to rigging costs and affect on the construction sequence.

Modularization of underground pipe and electrical tunnels appears to be beneficial as it appears to provide an easy way to construct these structures since the leave-in-place forms would eliminate the requirement for shoring during construction. While cable trays could be installed in the cable tunnels at the factory where the leave-in-place form modules are fabricated, it appears unlikely that the piping design would have progressed far enough for this to be done with much of the piping; however, additional design time could be gained if pipe was able to be “loaded” into the module at the site after it has been installed on its foundation mat.

The design of the plant should consider from the very beginning the concept of modularization, where systems and structures are designed as modular components of the plant, modularizing the completed design to the greatest extent possible. This requires more of an “area” or “zone” approach to the design. It may include multiple systems or even electrical and mechanical services to be included in the same module, which may also serve as the final structure. It is recognized that certain systems and components may not lend themselves to modularization due to routing requirements, stress considerations, etc. but a comprehensive modularization program can effectively reduce the field construction durations and installed cost as well as provide a quality product with significant fabrication occurring in the more ideal environment of a shop rather than the less ideal conditions of a construction jobsite.

9.5.2.4 Transportation of Large/Heavy Components

In general, sites located on or very near navigable waterways are most suitable for transportation of the large and heavy components. Ship transport will allow extremely large items to be transported with relative ease. Land transport vehicles can move extremely large and heavy loads but their operation is sometimes constrained by roadbed conditions or overhead obstructions, the most common of which are bridges. Upgrading infrastructure along public right of ways can be extremely expensive, and may not be politically feasible. Surreptitious routes and special handling considerations such as transferring the load to a barge to cross a river are often required and at times, the permitting agencies may choose not to grant a permit or may not be able to grant a permit under the enabling legislation.

The ability to transport very large fabrications to the jobsite can play a significant role in reducing the overall cost of the project. It appears that if they cannot be transported to the site fully assembled, the largest components (the reactor vessel and the reactor core barrel) could possibly be shipped to the jobsite in smaller pieces. A sophisticated shop facility would then be required to complete the assembly. This on-site fabrication facility would have to be able to perform this work of the highest quality; therefore highly skilled workers would be required. The cost of establishing the fabrication facility could be significant, but if multiple PBRs were to be located at a site, the cost per PBR would be reduced.

10.0 PBR TECHNOLOGY READINESS STUDY CONCLUSIONS

10.1 Design Status Assessment

The HTR-Module design is based on a German design that uses largely proven technology. This design met all necessary requirements of German nuclear regulatory authorities. Review of pertinent German design documents indicate that the HTR-Module was well into the final design stage. Foremost among the challenges of deploying this design for the NGNP is the need for the design to accommodate U.S. regulatory requirements, codes and standards (sometimes called Americanization of the design). It is clear that the NGNP PBR design must necessarily undergo some degree of regression from the near final design stage of the German HTR-Module on which it is based. It has been estimated that an NGNP based on the HTR-Module should be considered to be in the late conceptual design stage. In order to progress to the point of early preliminary design, a reconciliation of the design to NGNP requirements and an initial round of “Americanization” would be necessary. Though this design would be considered to be in the late conceptual design stage, it has certain advantages over other designs at this stage. Because the HTR-Module had progressed much further in Germany, a defined success path for major design decisions is largely available, which should eliminate or greatly reduce the need for multiple design iterations going forward. This could be of significant potential benefit, in terms of reduced schedule duration, engineering costs, and overall project risk.

10.2 Key PBR Issues

10.2.1 Stochastic Core

Based on experimental and analytical results, the stochastic nature of the fuel pebble’s movement in the PBR core is well understood and is predictable with statistical methods. Studies on power peaking effects due to artificially introduced fresh fuel spheres serve as a simple and conservative way to study statistical variations in fuel loading, flow speeds and clustering of more reactive fuel pebbles. These studies show that although the maximum power delivered in a fuel pebble may increase due to clustering, the maximum fuel temperature increases only moderately in normal and accident operation. The AVR melt-wire experiment provides valuable information on the maximum fuel temperature distributions as fuel pebbles pass through the core. Although the temperature measurements appeared to be higher than expected, detailed 3D CFD studies show that the temperature differences were mainly due to coolant bypass flows and radial power distributions, which were not included in the original analysis. The uncertainties in the pebble bed core are well understood, and the core design margins adequately compensate for these uncertainties.

10.2.2 Core Compaction

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.

10.2.3 Dust

Based on the experimental and analytical results of operations at AVR and THTR, dust did not cause problems affecting the reliability of either PBR. The AVR provided a valuable experimental data base on dust issues, including steady state conditions and dust remobilization in accident conditions. In that reactor, it was demonstrated that the dust at inner surfaces of the primary circuit forms a closed layer with strong binding on the surfaces. As such, the estimated amount of remobilized dust in the primary circuit of the HTR-Module in the DBA Depressurization Phase supports an enveloping value of 1 kg in the safety evaluation of the HTR-Module. In addition, dust explosions and dust cloud interference with RCCS function were demonstrated to be not credible concerns. The graphite dust generated in PBR plants should pose no real safety risks during normal operation and following a DBA.

10.2.4 Broken and/or Lost Pebbles

The experiences learned from the operation of AVR and THTR, have shown that the scrap pebble production rate can be kept to a very low level and that it does not represent a safety problem in any way for the operation of PBR.

The experiences have also shown that pebble damage occurring in the PBR can be minimized with optimal design of the graphite bottom support structure and fuel handling system, and by keeping the control rods inside the graphite reflector. Therefore, 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.

10.2.5 Proliferation

Though PBR concept has some inherent features that make proliferation of fissile materials from this type of reactor more difficult, it is not significantly more resistant than other reactor designs. The proliferation resistance of a PBR should be built from the integration of safeguards concerns in the details of the design in order to minimize the possible diversion paths and to facilitate safeguards inspections and measurements and from the development of a robust safeguards approach adapted to this type of reactor.

Although the PBR safeguards approach and the criteria are still in development, a hybrid scheme has been proposed and seems to offer the required robustness, with sufficient defense in depth features. The PBR design is no more attractive than other types of reactors spread in the world in terms of quality of the fuel fissile materials that can be diverted from it. Moreover, using a PBR is not the optimal solution for production of fissile materials for nuclear explosives.

The question of whether the PBR concept addresses the Generation IV goal of improved proliferation resistance cannot be assured at the level of the design concept. It is mainly by taking into account proliferation concerns in the details of the design and by developing advanced safeguards measures that proliferation resistance can progress.

10.2.6 Shutdown Margin

The available shutdown margin for high temperature pebble bed reactors is an issue that must be addressed by the core design. 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. Furthermore, it has been shown that the HTR-Module reactor geometry, with its reflector rod worth and positioning, can address the Xenon stability issue, which could pose a problem for any long core configuration.

The problem of core recriticality is at low temperature (< 50°C). This prevents the core from being cooled down to ambient temperature for maintenance. However, this is not an 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.

Overall, the safety benefits of benefits of the PBR reactor concepts, including HTR-Module, far outweigh the low temperature shutdown margin design issue even in a startup accident. Design characteristics of that support this are a large negative temperature coefficient, a large heat capacity, and no coolant phase change,

10.2.7 Online Refueling

Past PBR Fuel Handling Systems availability experience feedback and performance of the BUMS technology by ^{137}Cs spectroscopy has been examined and assessed regarding the current PBR technology based on the HTR-Module.

It has been concluded that past experience of frequent Fuel Handling System unavailability should not be considered as an intrinsic feature of PBR Fuel Handling Systems and that careful consideration of past experience as well as appropriate qualification test programs should largely support the successful operation of a new Fuel Handling System design.

As for the BUMS, particular care should be granted to the qualification tests of the highly demanding design specification of the HTR-Module. However, it has been shown that if the expected performances could not be reached, practical solutions would exist to tackle the difficulty, like the use of multiple BUMS in parallel.

Overall, this analysis has shown that the performance and reliability limits of PBR Fuel Handling Systems would not constitute a potential show stopper for efficient on-line refueling of future designs of PBR, including for process heat applications and associated high availability requirements.

10.2.8 Tritium

The mechanisms by which Tritium is produced in the HTGR are well understood and are not unique to the PBR design concept. Uncertainties in the Tritium production rate in the HTR-Module are mainly associated with an imprecise knowledge of fission yield, and accurately assessing Tritium-forming impurity levels.

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 the steam reboilers before subsequent process heat applications such as steam reformers 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.

10.3 Supporting Technology Database

An assessment of the design data needs for the PBR reactor type, based on the HTR-Module design was conducted. It was based on an analysis of the DDNs issued by the Westinghouse team for the development of the NGNP 750°C, steam cycle version of the PBR, excluding DDNs devoted to the hydrogen production process. Specific consideration of the HTR-Module design led to removal of some of the DDNs proposed by the Westinghouse team, which are relevant for parts of the less mature design, but not for a design fully developed and tested. Additionally, some DDNs that could not be found in the Westinghouse list have been added that correspond to the views of AREVA experts, which are often, but not always, parallel to the judgment of Westinghouse team. The result of this assessment is set of DDNs identifiable as applicable to the PBR technology based on the HTR-Module design. This activity did not generate new DDN documents for those that were identified as new DDNs.

10.4 Fuel Supply

A fuel design and associated qualification strategy was developed based on the current AGR program being conducted by the INL. 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.

Babcock and Wilcox is positioned to support the NGNP program and produce the selected fuel design. With modest capital investment, the capabilities to supply fuel for the HTR Initial Core can be secured in an approximately 5 year time frame. During this time frame, development efforts to optimize the fabrication process would occur. These efforts could then be channeled into the design and construction of a commercial fuel fabrication facility. The design of the facility would be modular. Additional modules can be added on an as need basis. This modular design allows for efficient scale up of commercial fuel fabrication beyond what is identified within.

As with any project involving the processing of Uranium above 5% ^{235}U, there are risks in securing a suitable Uranium supplier. Beyond that, the risks identified are all manageable. None of the risks identified are believed to be insurmountable.

10.5 Graphite Supply

The graphite infrastructure is believed to be adequate to produce the quantity of the selected grade of nuclear graphite on the planned NGNP production schedule. This assumes that the required quantity of graphite is ordered in a timely manner. The main issue on graphite acquisition is that every change in raw materials (and more specifically in filler coke origin) will involve the qualification of a new grade. After qualification, in order to secure graphite supplying, 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 stock for pitch coke graphite, like NBG-18, because pitch coke sources are rare.

10.6 Constructability/Transportability

As with any major construction project, there are risks and opportunities. The key is to identify both the risks and the opportunities early and plan their fate during the design phase of the project. Although there are challenges with the construction of the HTGR, we do not believe there are issues that extend beyond those experienced and resolved on similar projects. The current challenges with new-build nuclear projects have and will continue to prepare the industry for the deployment of the NGNP.

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