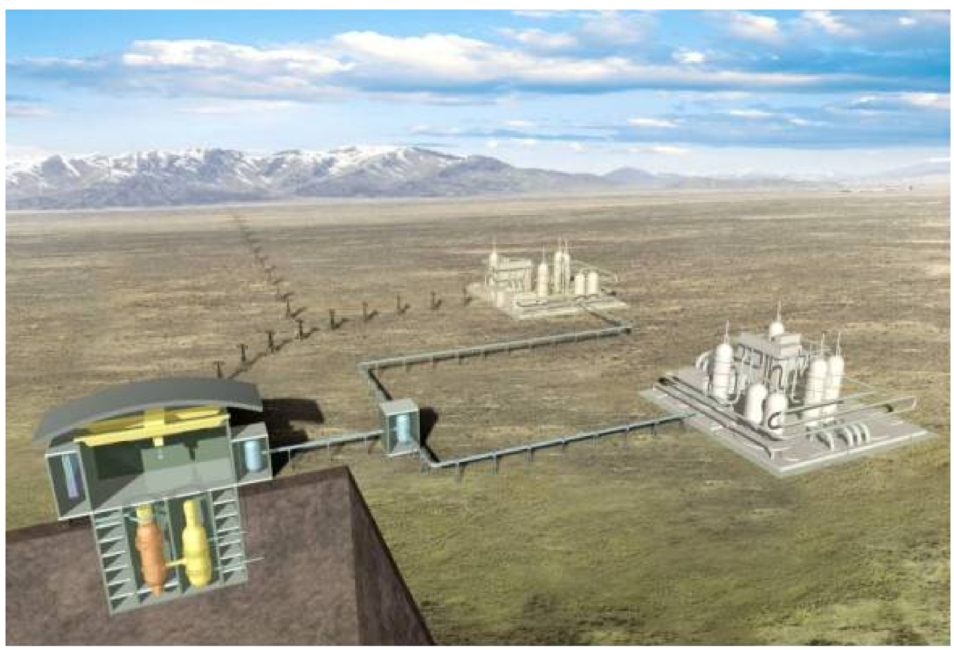
INL/EXT-06-11701
Revision 3

Next Generation Nuclear Plant Materials Research and Development Program Plan

August 2006



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Revision 3

Next Generation Nuclear Plant Materials Research and Development Program Plan

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Executive Summary

The Energy Policy Act (EPACT) of 2005 was enacted on July 14, 2005. This law established that the U.S Department of Energy (DOE) Secretary of Energy shall establish a "Next Generation Nuclear Plant" (NGNP) project. The NGNP project named in the Act was given the following attributes and guiding principles to manage its development:

- The NGNP consists of research, development, design (R&DD), construction, and operation of a prototype reactor to generate electricity and hydrogen
- The project shall be managed by the DOE Office of Nuclear Energy
- The Idaho National Laboratory (INL) shall be the lead DOE laboratory for the NGNP
- The INL shall establish collaborations with selected institutions of higher education, other research institutes and international researchers
- The INL shall organize an industrial consortium of partners for cost-shared R&DD, construction and operation
- The project shall be sited at the INL
- The project shall be licensed by the Nuclear Regulatory Commission (NRC) and by July, 2008 the NRC and DOE shall jointly submit a licensing strategy to Congress
- The project shall be organized to maximize technical interchange with the nuclear power industry, nuclear power plant construction firms, the chemical process industry and to seek international cooperation, participation and contributions
- The Nuclear Energy Research Advisory Committee (NERAC) shall review all program plans for the NGNP
- Phase 1 of the project (selection of hydrogen production technology, conduct R&DD and initial design activities) shall be completed no later than September 30, 2011
- Phase 2 of the project (continue R&DD, develop final design, apply for a license, construct and start operations) shall be completed by September 30, 2021
- Provision for authorization of appropriations was made

As a result of the direction provided, the INL and the DOE issued an NGNP Preliminary Project Management Plan (PPMP), INL/EXT-05-00952, Rev. 1, March, 2006. This document provides planning options for the development of the NGNP Project, a discussion of the project programmatic risks and a discussion of the most important deliverables to support Critical Decision-1 (CD-1) required by the DOE Acquisition Management System. This plan in conjunction with other plan and study documents to be issued will establish the detailed planning basis for the NGNP Project.

Based on these guidelines and other studies performed previously, DOE has selected the Very High Temperature Reactor (VHTR) design for the NGNP Project. The NGNP will demonstrate the use of nuclear power for electricity and hydrogen production without greenhouse gas emissions. The reactor design will be a graphite moderated, helium-cooled, prismatic or pebble-bed, thermal neutron spectrum reactor that will produce electricity and hydrogen in a state-of-the-art thermodynamically efficient manner. The NGNP will use very high burn-up, low-enriched uranium, TRISO-coated fuel and have a projected plant design service life of 60 years.

The VHTR concept is considered to be the nearest-term reactor design that has the capability to efficiently produce hydrogen. The plant size, reactor thermal power, and core configuration will ensure passive decay heat removal without fuel damage or radioactive material releases during accidents. The NGNP Project is envisioned to demonstrate the following:

- A full-scale prototype VHTR by about 2021
- High-temperature Brayton Cycle electric power production at full scale with a focus on economic performance
- Nuclear-assisted production of hydrogen with a focus on economic performance
- By test, the exceptional safety capabilities of the advanced gas-cooled reactors.

Further, the NGNP program will:

- Obtain a Nuclear Regulatory Commission (NRC) License to construct and operate the NGNP. This process will provide a basis for future performance based, risk-informed licensing
- Support the development, testing, and prototyping of hydrogen infrastructures

The NGNP Materials Research and Development (R&D) Program is responsible for performing R&D on likely NGNP materials in support of the NGNP design, licensing, and construction activities. The NGNP Materials R&D Program includes the following elements:

- Developing a specific approach, program plan and other project management tools for managing the R&D program elements
- Developing a specific work package for the R&D activities to be performed during each government fiscal year
- Reporting the status and progress of the work based on committed deliverables and milestones
- Developing collaboration in areas of materials R&D of benefit to the NGNP with countries that are a part of the Generation IV International Forum
- Ensuring that the R&D work performed in support of the materials program supports the PPMP and is in conformance with established Quality Assurance and procurement requirements

The objective of the NGNP Materials R&D Program is to provide the essential materials R&D needed to support the design and licensing of the reactor and balance of plant, excluding the hydrogen plant. The materials R&D program was initiated prior to the design effort to ensure that materials R&D activities are initiated early enough to support the design process. The thermal, environmental, and service life conditions of the NGNP will make selection and qualification of some high-temperature materials a significant challenge; thus, new materials and approaches may be required. The following materials R&D program areas are currently addressed in the R&D program being performed or planned:

- Qualification and testing of nuclear graphite and carbon fiber/carbon matrix composites for use in the NGNP. These components are essential to any VHTR design and the irradiation induced dimensional and material property changes must be properly modeled.

- Development of improved high-temperature design methodologies for application toward the further development, qualification, and selection of high-temperature metallic alloys for potential application in the NGNP. Currently, the data and models are inadequate for many of the high-temperature alloys required for construction of the VHTR.

- Expansion of American Society of Mechanical Engineers (ASME) Codes and American Society for Testing and Materials Standards in support of the NGNP Materials R&D Program. This work is required because of NRC licensing and construction requirements.

- Development of an improved understanding of, and models for, the environmental effects and thermal aging of the metallic alloys for potential application in the NGNP. This work is needed because metallic alloys undergo property changes as a function of exposure to the high temperature, impure gas environments expected in the VHTR.

- Development of a materials handbook/database in support of the Generation IV Materials Program. This effort is required to collect and document in a single source the information generated in this and other previous VHTR materials R&D program

- Performance of materials studies in support of the PPMP

- Support of selected university materials related research and development (R&D) activities that would be of direct benefit to the NGNP Project

- Support of international materials related collaboration activities through the DOE sponsored Generation IV International Forum (GIF) Materials and Components (M&C) Project Management Board (PMB)

- Support of document review activities through the Materials Review Committee (MRC) or other suitable forum

Other program elements may be added or deleted in the future as required by NGNP Project development activities

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Acronyms

AGCNR	Advanced Gas-Cooled Nuclear Reactor
AGR	Advanced Gas-Cooled Reactor
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
AVR	Albeitsgemeinschaft Versuchsreaktor
B&PV	B&PV
CEA	Atomic Energy Commission (France)
C_f/C	Carbon/carbon Composite
CRBRP	Clinch River Breeder Reactor Project
CTE	Coefficient of thermal expansion
CV	Cross vessel
DLOF	Decompression Loss of Fluid Accident
DOE	Department of Energy
EUROFER	Specific European name of a steel alloy
GA	General Atomics
GIF	Generation IV International Forum
GT-MHR	Gas Turbine-Modular Helium Reactor
HFIR	High-Flux Isotope Reactor
HHT	High-temperature helium turbine system
HPC	High-Pressure Compressor
HTDM	high-temperature design methodology
HTGR	High-Temperature Gas Reactor
HTR	High-Temperature Reactor
HTTR	High-Temperature Engineering Test Reactor
IAEA	International Atomic Energy Agency

IHX	Intermediate heat exchanger
INL	Idaho National Laboratory (formerly the Idaho National Engineering and Environmental Laboratory)
ITRG	Independent Technical Review Group
JAERI	Japanese Atomic Energy Research Institute
KAERI	Korean Atomic Energy Research Institute
KFA	Kernforschungsanlage Julich (Institute for Chemical Technology, Germany)
LMR	Liquid-metal reactor
LPC	Low-Pressure Compressor
LWR	Light-Water Reactor
MCNP	Monte Carlo physics code
MRC	INL Materials Review Committee
NDE	nondestructive examination
NE	DOE Office of Nuclear Energy
NGNP	Next Generation Nuclear Plant
NPH	Nuclear process heat
NRC	Nuclear Regulatory Commission
NTD	National Technical Director
ODS	Oxide dispersion strengthened
ORNL	Oak Ridge National Laboratory
PBMR	Pebble Bed Modular Helium Reactor
PBR	Pebble Bed Reactor
PCU	Power conversion unit
PMB	GIF VHTR Materials and Components Project Management Board
PMR	Prismatic Modular Reactor
PNNL	Pacific Northwest National Laboratory
PNP	Prototype Nuclear Process Heat

PWHT	Post Weld Heat Treatment
QA	Quality Assurance
QAP	Quality Assurance Program
R&D	Research and development
RPV	Reactor pressure vessel
SCS	Shutdown cooling system
SG-ETD	Subgroup on Elevated Temperatures Design
SGL	Name of a graphite company
SiC_f/SiC	silicon-carbide/silicon-carbide composite
SIM	System Integration Manager
TBC	Thermal barrier coatings
THTR	Thorium Hochtemperatur Reaktor
TRISO	Tri-isotopic (fuel)
UCAR	Name of a graphite company that is wholly owned by Graftek
UK	United Kingdom
VHTR	Very High Temperature Reactor

NGNP Materials R&D Program Plan

1. Introduction and Purpose

The Energy Policy Act of 2005 was enacted July 14, 2005 (discussed in Section 1.2). This Act established that the Nuclear Energy Research Advisory Committee review NGNP Program development at periodic intervals. This is discussed in Section 1.3. As a result of the direction provided in this Act, the INL and the DOE issued an NGNP Preliminary Project Management Plan (PPMP), INL/EXT-05-00952, Rev. 1, March, 2006 (discussed in Section 1.5). This document provides planning options for the development of the NGNP Project, a discussion of the project programmatic risks and a discussion of the most important deliverables to support Critical Decision-1 (CD-1) required by the DOE Acquisition Management System. This plan in conjunction with other plan and study documents to be issued will establish the detailed planning basis for the NGNP Project. The path forward for development of the NGNP will be discussed in Section 1.6.

Based on these guidelines and other studies performed previously, DOE has selected the Very High Temperature Reactor (VHTR) design for the NGNP Project. The NGNP will demonstrate the use of nuclear power for electricity and hydrogen production without greenhouse gas emissions. The reactor design will be a graphite moderated, helium-cooled, prismatic or pebble-bed, thermal neutron spectrum reactor that will produce electricity and hydrogen in a state-of-the-art thermodynamically efficient manner. The NGNP will use very high burn-up, low-enriched uranium, TRISO-coated fuel and have a projected plant design service life of 60 years. The VHTR concept is considered to be the nearest-term reactor design that has the capability to efficiently produce hydrogen. The plant size, reactor thermal power, and core configuration will ensure passive decay heat removal without fuel damage or radioactive material releases during accidents.

The NGNP Materials Research and Development (R&D) Program is responsible for performing R&D on likely NGNP materials in support of the NGNP design, licensing, and construction activities. The NGNP Materials R&D Program includes the following elements:

- Developing a specific approach, program plan and other project management tools for managing the R&D program elements
- Developing a specific work package for the R&D activities to be performed during each government fiscal year
- Reporting the status and progress of the work based on committed deliverables and milestones
- Developing collaboration in areas of materials R&D of benefit to the NGNP with countries that are a part of the Generation IV International Forum
- Ensuring that the R&D work performed in support of the materials program supports the PPMP and is in conformance with established Quality Assurance and procurement requirements

The objective of the NGNP Materials R&D Program is to provide the essential materials R&D needed to support the design and licensing of the reactor and balance of plant, excluding the hydrogen plant. The mission statement for the NGNP Materials Program and assumptions are discussed in Section 1.1. The materials R&D program was initiated prior to the design effort to ensure that materials R&D activities are initiated early enough to support the design process. The thermal, environmental, and service life conditions of the NGNP will make selection and qualification of some high-temperature materials a significant challenge; thus, new materials and approaches may be required. The following materials R&D program areas are currently addressed in the R&D program being performed or planned:

- Qualification and testing of nuclear graphite and carbon fiber/carbon matrix composites for use in the NGNP. These components are essential to any VHTR design and the irradiation induced dimensional and material property changes must be properly modeled.

- Development of improved high-temperature design methodologies for application toward the further development, qualification, and selection of high-temperature metallic alloys for potential application in the NGNP. Currently, the data and models are inadequate for many of the high-temperature alloys required for construction of the VHTR.

- Expansion of American Society of Mechanical Engineers (ASME) Codes and American Society for Testing and Materials Standards in support of the NGNP Materials R&D Program. This work is required because of NRC licensing and construction requirements. A DOE initiative that addresses this area is discussed in Section 1.7.

- Development of an improved understanding of, and models for, the environmental effects and thermal aging of the metallic alloys for potential application in the NGNP. This work is needed because metallic alloys undergo property changes as a function of exposure to the high temperature, impure gas environments expected in the VHTR.

- Development of a materials handbook/database in support of the Generation IV Materials Program. This effort is required to collect and document in a single source the information generated in this and other previous VHTR materials R&D program

- Performance of materials studies in support of the PPMP

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- Support of document review activities through the Materials Review Committee (MRC) or other suitable forum

Other program elements may be added or deleted in the future as required by NGNP Project development activities

The basic technology for the NGNP was established in former high-temperature gas-cooled reactor plants (e.g., DRAGON, Peach Bottom, Albeitsgemeinschaft Versuchsreaktor [AVR], Thorium Hochtemperatur Reaktor [THTR], and Fort St. Vrain). These reactor designs represent two design categories: the Pebble Bed Reactor (PBR) and the Prismatic Modular Reactor (PMR). Commercial examples of potential NGNP candidates are the Gas Turbine-Modular Helium Reactor (GT-MHR) from General Atomics (GA), the High Temperature Reactor concept (ANTARES) from AREVA, and the Pebble Bed Modular Reactor (PBMR) from the PBMR consortium. Furthermore, the Japanese High-Temperature Engineering Test Reactor (HTTR) and Chinese High-Temperature Reactor (HTR) are demonstrating the feasibility of the reactor components and materials needed for NGNP. (The HTTR reached a maximum coolant outlet temperature of 950 °C in April 2004.) Therefore, the NGNP is focused on building a demonstration plant, rather than simply confirming the basic feasibility of the concept.

Demonstration of hydrogen production may use both electricity and process heat from the reactor. A separate program for development of efficient hydrogen production technologies is operating in parallel with the NGNP Materials Research and Development (R&D) Program.

The operating conditions for the NGNP represent a major departure from existing water-cooled reactor technologies. Although a significant assortment of materials and alloys for high-temperature applications are in use in the petrochemical, metals processing, and aerospace industries, a very limited number of these materials have been tested or qualified for use in nuclear reactor-related systems. Today's high-temperature alloys and associated American Society of Mechanical Engineers (ASME) Codes for reactor applications reach about 800 °C. Some primary system components for the NGNP will require use of materials at temperatures above 800 °C. Such use will require further assessment of existing, well-characterized materials or selection of newer materials for which less data exists. Potential postulated accident conditions with associated temperatures above nominal operational temperatures would dictate the use of composite or ceramic materials within the reactor pressure vessel (RPV). The use of structural ceramics or composites in safety-related reactor components represents a completely new challenge to the nuclear industry.

Qualification of materials for successful and long-life application at the high-temperature conditions planned for the NGNP is a major purpose for the NGNP Materials R&D Program. Few choices exist for metals or metallic alloys for use at NGNP conditions and the design lifetime considerations for the metallic components may restrict the maximum operating temperature.

A materials survey[1], [a] was conducted in January 2003 to identify material requirements that are beyond the limits of current materials technology. That initial look indicated that the materials issues are solvable, but resolution may be expensive and require sustained commitment for multiple years.

A broader review of design features and important technology uncertainties of the NGNP was performed by an Independent Technology Review Group (ITRG) during the period from November 2003 through April 2004. The report[2] provides valuable insight on several focus areas associated with the development of the NGNP and includes a section specifically on materials development. This report is discussed in Section 1.5.

Selection of the technology and design configuration for the NGNP must consider both the cost and risk profiles to ensure that the demonstration plant establishes a sound foundation for future commercial deployments. The NGNP challenge is to achieve a significant advancement in nuclear technology while at the same time setting the stage for an economically viable deployment of the new technology in the commercial sector soon after 2020.

1.1 Mission Statement and Assumptions

The mission of the NGNP Materials Program is noted below:

- Support the objectives associated with the NGNP in the Energy Policy Act. Provide any materials related support required during the development of the NGNP Program

a. Complete bibliographic references appear in numerical order at the end of each Section. Throughout this document, reference notations appear in the normal numerical format.

- Provide materials related support required to meet the goals given in the PPMP
- Provide leadership and support related to international collaboration associated with VHTR materials R&D issues
- Provide leadership and support associated with university materials R&D programs that are of direct benefit to the objectives of the NGNP Program
- Initiate materials related studies as required to support the NGNP Program

The following assumptions are incorporated into these mission statements and are used in estimating the scope, cost, and schedule for completing the materials R&D processes:

1. The NGNP Program including the materials program will continue to be directed by the INL based on the guidelines given in the Energy Policy Act of 2005. The scope of work will be adjusted to reflect the level of congressional appropriations.
2. The reactor design has not been formally selected and this process will take some time to complete. For the purposes of this document, the design is assumed to be either a helium-cooled, prismatic graphite block core design or a pebble-bed core design fueled with tri-isotopic (TRISO)-design fuel particles in carbon-based compacts or pebbles.
3. The NGNP must demonstrate the capability to obtain a Nuclear Regulatory Commission (NRC) operating license. However, the licensing strategy for the NGNP has not been developed to date and this process may take some time to complete. It is, however, assumed that the design, materials, and construction will need to meet appropriate Quality Assurance (QA) methods and criteria and other nationally recognized codes and standards.
4. The NGNP is expected to be a full-sized reactor plant based on the reactor concept selected (400-600 MWt) capable of electricity generation with a hydrogen demonstration unit of appropriate size
5. The demonstration plant will be designed to operate for a nominal 60 years.
6. Application for an NRC operating license and fabrication of the NGNP will occur with direct interaction and involvement of one or more commercial organizations.

1.2 Energy Policy Act of 2005 (H.R. 6) Section 641

Highlights associated with the Energy Policy Act as it pertains to the development of the NGNP are given below:

- Secretary shall establish a project “Next Generation Nuclear Plant”
- The NGNP consists of research, development, design, construction, and operation of prototype reactor to generate electricity and hydrogen
- The project may be combined with the Generation IV initiative
- The project shall be managed by the DOE Office of Nuclear Energy

- The INL shall be the lead laboratory for NGNP
- INL shall establish collaborations with: institutions of higher education, other research institutes and international researchers
- The INL shall organize and industrial consortium of partners for cost-shared RD&D, construction and operation
- The consortium shall give preference to lead industrial partners that retain US technological leadership in the project, while maximizing cost sharing
- The project shall be sited at the Idaho National Laboratory
- The NGNP shall be licensed by the NRC

The major elements of the program shall include high temperature hydrogen RD&D, energy conversion development, nuclear fuel RD&D and qualification, materials RD&D and qualification, reactor and balance of plant (BOP) design, safety analysis, and qualification.

Phase 1 of the program shall extend from fiscal year (FY) 2006 to 2011 and include the selection of the hydrogen production technology, the conduct of R&DD in support of energy conversion, nuclear fuel and materials, and carry out initial design activities in support of the development of the prototype plant including the determination of whether the prototype can combine electricity generation and hydrogen production in one plant. The initial design parameters for the reactor should be set during this phase.

Phase 2 of the project shall extend from FY 2011 to FY 2021 and continue the R&DD required from Phase 1. A final design will be developed in Phase 2. It is anticipated that the final design will be developed by the initial efforts of up to four design teams with a down select to one team made for the final design. The lead industrial partner selected may also perform system integration for final design and construction. This phase will also include the application for an NRC to construct the prototype, construction of the prototype nuclear reactor and its associated hydrogen or electricity facilities and the start of operations.

The NGNP Project shall be structured to maximize technical interchange the nuclear power industry, nuclear power plant construction firms, the chemical process industry regarding the use of process energy for producing hydrogen and integration of technologies into chemical processing environments, and international efforts in areas related to the project. All international activities shall be coordinated with the GIF.

The NERAC shall review all program plans for the NGNP and insure that all important issues receive attention. Within 180 days of the Act, the NERAC shall review the existing program plans for the NGNP in light of the recommendations contained in *Design Features and Technology Uncertainities for the NGNP*, dated June 30, 2004 (INEEL/EXT-04-01816) and address any recommendations not incorporated in the NGNP program. This is the report issued to the DOE from an Independent Review Group (ITRG) established at the request of the DOE by the INL to review technology alternatives for meeting the functional objectives of the NGNP. When Phase 1 is nearly complete, the NERAC shall conduct a comprehensive review and recommend to the Secretary of Energy if the NGNP is ready to proceed to the second phase.

By July 2008, the DOE and the NRC shall jointly submit to Congress a licensing strategy for NGNP that shall include how current Light Water Reactor (LWR) licensing requirements may be adapted, the identification of analytical tools requiring development to verify designs and Safety Significant Conditions (SSCs), and budget needs.

Funding to the support the NGNP Project is currently proceeding as a part of the DOE budget process on a year-to-year basis. It is currently unclear how funding authorization for the NGNP will evolve in the future.

1.3 NERAC Report

A report summarizing a review performed by NERAC of the NGNP Project as required by the Energy Policy Act was issued in January, 2006. The executive summary from this report is given below:

> *In 2002, the Department of Energy (DOE) Office of Nuclear Energy (NE) completed a technology roadmap project that provided an overall plan to the broad vision of enhancing the future role of nuclear energy systems in the U.S. and the world at large. The current NE plan puts a top priority on the successful development of a high-temperature fission reactor system, the Next Generation Nuclear Plant (NGNP), to meet these Gen-IV overall goals. In August 2005, the U.S. Congress passed and the President signed the Energy Policy Act of 2005. One of the key provisions of that legislative authorization was establishment of the NGNP project and the designation of an overall plan and timetable for its research, design, licensing, construction and operation by the end of FY 2021. One of the final directives of the EPACT was to require an initial review of the NGNP project and its associated R&D plan by the Nuclear Energy Research Advisory Committee (NERAC). In September 2005, the NERAC chair and co-chair charged the Gen-IV subcommittee of the full committee with the task of conducting this review. The subcommittee is composed of four members of NERAC along with two additional nuclear engineering experts from the industry, acting as unpaid consultants.*
>
> *The complete charge to the NERAC subcommittee is given in Appendix A (*of the NERAC report). *This initial review focused on the existing NGNP program plan in light of the recommendations from an Independent Technical Review Group (ITRG) and addresses any ITRG recommendations not incorporated into NGNP plans.*
>
> *The subcommittee focused on the first phase of the NGNP program; i.e., between 2005 and 2011.*
>
> *This first phase includes:*
>
> - *Determination of whether the NGNP should produce electricity, hydrogen, or both;*
> - Selection and validation of a hydrogen generation technology;
> - *Conduct of R&D on associated technologies and components (energy conversion, nuclear fuel development, materials selection, reactor and plant systems development); and*
> - *Initial design activities for the prototype nuclear power plant.*
>
> *The subcommittee recommends a series of actions to make the NGNP program as effective as possible.*

> ***Recommendation (1):*** *The current mission for the NGNP is to design and build a reactor that generates electricity and produces hydrogen. The subcommittee recommends that this dual mission be reconsidered and not be accepted without further analysis. The subcommittee further recommends that this analysis be done as outlined in the following discussion.*
>
> ***Recommendation (2):*** *The DoE-NE staff should conduct, with the assistance of key industry representatives, economic and engineering trade studies that consider:*
>
> - *The targets for hydrogen production for various scenarios over the next few decades;*
> - *The DoE target for hydrogen production via nuclear power in this overall context;*
> - *The likely hydrogen production and electricity production alternatives and how those alternatives would be factored into determining the proper mission for the NGNP.*
>
> *The selection of the ultimate NGNP mission can drive the reactor design in different directions. The subcommittee recommends that these trade studies be funded, initiated immediately and completed as soon as possible.*
>
> ***Recommendation (3):*** *The subcommittee recommends that the DoE develop the NGNP as a reactor facility that can be upgraded as the technology advances. Conceptually, the facility would be built using a smaller reactor, carefully choosing the scale to be the smallest reactor that could be reasonably extrapolated to support full size commercial applications, as a 'technology demonstrator'.*
>
> ***Recommendation (4):*** *The DoE-NE staff should update its R&D plans and develop options that can support a reactor deployment much before the 2017-2021 timeframe. EPACT requires the overall cost of the NGNP project be shared with U.S. industry as well as members of the international community. The subcommittee believes that the chances of substantial industrial contributions and international collaborations to NGNP are greatly decreased with a completion target date of 2021. Further, these plans should adopt and enhance the ITRG perspective that to achieve a successful project even in the later time period, less aggressive project objectives must be adopted; e.g., for reactor outlet temperatures, fuel selection and performance.*
>
> *The subcommittee notes that the DoE has already begun to address these recommendations and urges continued refinements and revisions. The subcommittee compares ITRG recommendations to current plans in Appendix B* (of the NERAC report) *and provides more detailed R&D suggestions in Section VIII of the report.*

Further NERAC discussion associated with Recommendation 1 from Section V of their report is given below:

> *The EPACT requires that a range of technology missions be considered for the NGNP: to generate electricity, to generate hydrogen, or to do both. The original NGNP concept was based on the NGNP being a commercial scale demonstration plant for both hydrogen and electricity generation with the additional requirement that these*

capabilities be economically competitive. In our recent meeting with the DOE-NE Staff, they indicated that the working assumption for the NGNP remained the same; i.e., it still had a dual mission of electricity and hydrogen production.

The subcommittee recommends that the NGNP dual mission should be reconsidered and not be accepted without further analysis. There are several reasons for this conclusion:

- *The cost and complexity of the facility will be strongly affected by the mission.*
- *The time that the facility needs to be deployed is strongly influenced by the mission.*
- *The hydrogen mission depends strongly on the parallel hydrogen production research program being carried out in the US National Hydrogen Initiative (NHI). The result of the NHI will likely set the key performance parameters for the NGNP, such as the process heat temperature requirements as well as the associated efficiencies.*
- *The synergy with ongoing activities, and therefore, potential cost share with others will depend on the mission. For example, the South Africans are planning to build an electricity-producer pebble-bed prototype that will startup in the 2011-2013 time frame. Similarly the Japanese are operating the HTTR in Japan, a prismatic core reactor design, to study high temperature reactor operation and develop hydrogen production as well as other industrial applications. Properly choosing the NGNP mission is crucial to obtaining the cooperation, participation and financial contributions of these other programs, as well as potential U.S. industrial collaborators in an effective, cooperative way.*
- *The combined hydrogen and electricity mission is much more challenging than either single mission and will impose a greater burden on current and future funding resources.*

Given that large-scale hydrogen production is a key DoE mission, for which the NGNP can have a significant role, the subcommittee recommends that the DoE-NE staff conduct, with the assistance of key industry representatives, economic and engineering trade studies that consider:

- *The targets for hydrogen production for various scenarios over the next few decades;*
- *The DoE target for hydrogen production via nuclear power in this overall context;*
- *The likely hydrogen production and electricity production alternatives and how those alternatives would be factored into determining the proper mission for the NGNP.*

Because the selection of the ultimate NGNP mission can drive the reactor design in substantially different directions, the subcommittee recommends that these trade studies be funded, initiated immediately and completed as soon as possible.

A NERAC discussion of the ITRG recommendations is given below and is taken from Section VII and Appendix B of the NERAC report:

From the Fall of 2003 through the summer of 2004, a review panel of experts in nuclear fission technology, specifically gas-cooled nuclear reactor technology, conducted a comprehensive review of the NGNP program. This Independent Technology Review Group (ITRG) report was published in the summer of 2004 [3] and provided a series of recommendations to provide additional guidance for the NGNP reactor program; i.e., basic design assumptions, the R&D program as well as the commercial development and associated licensing and eventual commercial operation.

The major ITRG recommendations are outlined in Appendix B along with the current DoE response and R&D plans. It is important to note that because of the timing of this current NERAC review, shortly after the EPACT was enacted, the DoE had not completed a formal revised plan of the R&D program associated with the NGNP. Hence, the NERAC subcommittee met with cognizant personnel to understand current DoE plans and intent. In order to properly compare the ITRG recommendations with the current DoE NGNP program, this NERAC subcommittee reviewed the major ITRG recommendations with the DoE staff and summarized our observations and recommendations in Appendix B. There are additional detailed comments about the NGNP concept by the ITRG in its document that are still being evaluated by the DoE staff. These additional ITRG detailed comments are not addressed in this report.

There are also a number of substantive R&D issues that require more detailed discussion. For these R&D issues, a detailed discussion is provided below. The central theme is that the subcommittee recommends developing R&D plans that can support a reactor deployment much before the 2017-2021 timeframe. Further, these plans should adopt and enhance the ITRG perspective that to achieve a successful project even in the later time period, less aggressive project objectives must be adopted; e.g., for reactor outlet temperatures, fuel selection and performance. The subcommittee notes that the DoE has already begun to address these recommendations and urges continued refinements and revisions.

The Appendix B table from the NERAC report regarding major observations and recommendations regarding ITRG report recommendations is given below:

ITRG Recommendation	***DoE Action***	***NERAC Comment***
A) ITRG recommendations were based on a NGNP schedule for initial operation in 2017	*EPACT legislation provided a schedule with two phases; i.e., 2005-2011 - R&D and 2011-2021 - construction and licensing*	*DoE should re-examine and modify its detailed R&D program plan to meet the target timetable for technology selections*
B) For the two VHTR reactor concepts, neither is more likely to be successful for the NGNP	*DoE has modified its R&D plan to provide a more balanced effort for prismatic & pebble VHTR*	*No specific comments here.*
C) It is impractical for a molten-salt-cooled reactor development effort to be successful for NGNP	*DoE has adjusted its R&D program with only small MSR exploratory feasibility studies*	*There has been an appropriate alignment between DoE plans and ITRG recommendations*

ITRG Recommendation	***DoE Action***	***NERAC Comment***
D) Use of molten-salt in a heat transfer loop in the NGNP may be a desirable design concept	*DoE has adjusted its R&D program and has started a design activity in this technical area*	*There has been an appropriate alignment between DoE plans and ITRG recommendations*
E) Consider alternatives for licensing and purchase of viable technology from offshore vendor.	*This alternative is not consistent with EPACT 2005 and DoE plan*	*No specific comments here*
F) NGNP fuel development should focus on processes that have most successful worldwide experience base (e.g. UO_2 kernel)	*DoE agrees that worldwide fuel experience base be considered, but technically disagrees with ITRG focus only on UO_2 kernel*	*NERAC subcommittee examined this item in detail (F and G), and recommends that the fuel R&D program be reconsidered (VIII.7)*
G) NGNP fuel development plan should incorporate UO_2 & UCO kernels in R&D to determine the influence of fuel manufacturing processes on fuel quality	*DoE agrees and has incorporated this approach in AGR in-pile tests with UO_2 & UCO kernels; it has modifications planned for 2011 technology selection goals*	*NERAC subcommittee recommends that the fuel R&D program reconsider how AGR in-pile tests can be optimized for 2011 technology selection target*
H) Fuel development R&D plan should be consistent with overall NGNP R&D plan and schedule	*EPACT 2005 has provided an overall NGNP schedule and DoE is aligning its R&D schedule to it*	*The overall R&D plan needs to be aligned with the NGNP reassessed mission and associated schedule*
I) ITRG views need to achieve a high outlet temperature in NGNP be justified, and suggested a reactor outlet value of 900-950C	*DoE agrees and aims to set NGNP reactor outlet temperature by hydrogen production needs and the material capabilities*	*DoE approach is consistent with ITRG recommendation; NERAC recommends as wide a range as possible given tech. constraints*
J) An indirect cycle power conversion concept fulfills the high-level functional objectives	*DoE agrees this minimizes risk; this approach is needed for hydrogen production, but may be premature for electricity now.*	*DoE plan is generally consistent with ITRG recommendation, but subcommittee recommends the NGNP mission be reassessed*
K) The development of a high-temperature hydrogen production capability should be accelerated	*DoE agrees and notes that there has been greater R&D activity in FY-05 and will be in FY-06*	*NERAC is pleased with the R&D plan and research activity for hydrogen production*
L) Resource intensive R&D can benefit from direct international and industrial participation	*The GIF provides international input to NGNP and DoE plan is developing to involve industry*	*NERAC subcommittee has some detailed suggestions for industrial input (Section VIII.)*
M) ITRG noted that design uncertainties (IHX, RPV) be addressed with focused R&D	*DoE has identified these and other items for R&D plan for revision to meet 2011 target*	*No specific comments here*

ITRG Recommendation	*DoE Action*	*NERAC Comment*
N) ITRG concern: Electricity, Hydrogen or a dual mission	*EPACT 2005 has given guidance on this issue*	*NERAC subcommittee considers this a key decision (Section VI.)*

A portion of Section VIII of the NERAC Report that includes a discussion of the NGNP Materials R&D Program is given below:

The DoE-NE staff developed an overall R&D program that will enable the full range of possible missions and gas reactor designs for the NGNP. Completion of activities specified in this R&D plan requires funding in the range of 100 million per year and nearly a decade to complete. Current funding for the entire Gen-IV program is in the range of 50 million dollars a year, and the subcommittee believes that it is unlikely that US funding will increase substantially in the near future. Hence, the success of this project requires that a large portion of R&D funds come from industrial and international partners, in-kind and direct, both private and public. In addition to recommending that an early decision be made with respect to the NGNP's mission, the subcommittee recommends that the R&D program be reviewed and an integrated realistic plan be developed that is consistent with the selected mission, that is consistent with potential funding realities, and that can provide the required research results for a possible earlier deployment than the currently proposed 2017-2021 timeframe.

The multiple missions and designs currently considered by NGNP R&D has led to a very broad research program that may not even yield the required data for deploying an NGNP by 2021. Our committee believes that a careful review should be completed (with industry, regulatory, and international participation) after the mission is selected to confirm that NGNP research primarily supports the selected mission, has appropriate industry, regulatory, and international support, has considered the ITRG recommendations, and is consistent with EPACT.

The subcommittee offers a number of specific suggestions for the DoE-NE staff to consider that can provide more focus to the current R&D plan

1. *Develop an integrated schedule of all planned activities, similar to the computer-based scheduling program currently used by DoE staff for the Nuclear Hydrogen program. This integrated schedule and associated work breakdown structure can be used to identify a baseline R&D plan for highest priority R&D activities (and assess the impact of alternative/additional R&D tasks). Such a schedule should facilitate the adjustments that will be needed to the NGNP R&D program after its mission is selected.*

2. *Conduct a series of structured workshops with industry, regulatory, laboratory, and international representatives to discuss the following:*

    - *Trade study results to select the NGNP's mission*

    - *Design optimization studies to meet the selected NGNP mission (e.g., plant power level, fuel configuration, fuel material, operating temperatures, structural materials)*

- *R&D program elements (analysis codes and associated data needs, materials research, fuels development and certification). As noted below, steering committees may be required to ensure that appropriate parties provide continued input in some of these areas.*

- *Appropriate cost sharing by NGNP stakeholders (industry, international, regulatory agencies, DOE)*

- *Materials R&D: The subcommittee is aware that there is also research being conducted in-this area in the nuclear hydrogen program and recommends that the work for NGNP be better coordinated with that work to avoid overlap and assure the work is complementary. After trade studies are completed and an NGNP mission selected, the following items should be considered:*

- *Focus on key material research needs. For example, if hydrogen production were selected as the key mission and an Intermediate Heat Exchanger (IHX) concept were included in the optimized design for that mission, the use of developmental materials should be limited to the IHX (and a systematic evaluation should be conducted to identify an appropriate material for the IHX operating conditions and develop a "code case" for the identified material). To the extent possible, the remainder of the plant should rely on conventional, proven materials.*

- *Graphite certification activities, which are required irrespective of the NGNP mission and reactor design, should be reviewed and accelerated so that an appropriate material is certified within the required timeframe for deployment.*

A more detailed discussion of the materials related ITRG report recommendations is given in Section 1.4.

1.4 ITRG Materials Issues and NGNP Materials Program Response

As noted previously, the ITRG report entitled *Design Features and Technology Uncertainities for the Next Generation Nuclear Plan*t was issued on June 30, 2004 by the INL An analysis of the materials related risk issues noted in this report and comments associated with these risk issues are given below and in Appendix E-2 of the PPMP report; however, some of the comments have been updated to reflect additional information currently available.

The ITRG performed their review from November 2003 through April 2004. Their final report was completed on June 30, 2004 and the INL cleared it for limited distribution on June 30, 2004. The purpose of the ITRG review was to provide a critical review of the proposed NGNP project and identify areas of R&D that needed attention. As stated in the report, the ITRG observations and recommendations focus on overall design features and important technology uncertainties of a very-high-temperature nuclear system concept for the NGNP. The risk issues discussed in the ITRG report related to the materials development program will be discussed below.

1.4.1 Upper Temperature Limit

Technical Observation. The specified NGNP gas outlet temperature of 1000°C is beyond the current capability of metallic materials. The requirement of a gas outlet temperature of 1000°C will result in operating temperatures for metallic core components (core barrel, upper shroud, control rod drive assemblies), and intermediate heat exchanger (IHX) components that will approach 1200°C in some cases. Metallic materials that are capable of withstanding this temperature for the anticipated plant life will not be available on the NGNP schedule, if they can be developed at all. Nonmetallic materials capable of this temperature will require a development program that cannot support the NGNP schedule.

Associated Risk. The requirement for a gas outlet temperature of 1000°C represents a significant risk that ITRG judges cannot be resolved consistent with the schedule for the NGNP.

Recommendation. It is recommended that the required gas outlet temperature be reduced such that metallic components are not exposed to more than 900°C for the base NGNP design. Raising the gas outlet temperature to 950°C may be considered but at the potential expense of a reduced life (<60 year) for key components (e.g., the IHX).

NGNP Materials Program Response. This recommendation was implemented and the NGNP materials program now focuses on a reactor outlet temperature for the NGNP of 950°C maximum.

1.4.2 Pressure Boundary Time-Dependent Deformation

Technical Observation. Several of the NGNP concepts that were reviewed require many of the irreplaceable Class I boundary components (pressure vessel, piping, etc.) to operate at temperature and stress combinations that will result in significant time-dependent deformation (creep) during the component life. While there is allowance (American Society of Mechanical Engineers [ASME] Code Case) for the inclusion of time-dependent deformation in pressure vessel designs, this has not been a part of commercial nuclear pressure vessel and piping systems in the past and represents a very significant departure from current practice. Moreover, it is likely that creep as well as fatigue-related time-dependent deformation will be present. Creep-fatigue interaction represents the most complex form of high temperature behavior, often requiring component-specific design rules. In addition, the regulatory infrastructure does not have experience with including significant time-dependent deformation in the licensing and safety evaluation process.

Associated Risk. The allowance of time-dependent deformation in the irreplaceable Class I boundary represents an unacceptable risk for the NGNP program.

Recommendation. The ITRG recommends that time-dependent deformation be limited to "insignificant," as defined by the ASME Code, during the life of irreplaceable (60-year life) components for the NGNP. Time-dependent deformation for replaceable Class I components (portions of the IHX, interface heat exchanger for the hydrogen system, etc.) can be allowed, but only with the addition of significant additional levels of inspection and monitoring. However, the fraction of the Class I boundary that experiences significant creep deformation must be limited as much as possible.

NGNP Materials Program Response. The program agrees with this recommendation and this philosophy is currently reflected in materials design approaches.

1.4.3 Fabrication, Welding, Inspection, and Monitoring

Technical Observation. The NGNP pressure vessel represents a significant increase in size over previous systems because of the 600-MWt power rating and the relatively lower power density. The large size of the pressure vessel will require a significant amount of field fabrication, including welding, post-weld heat treatment, and machining. Fabrication, field welding, and post-weld heat treatment represent significant extensions of current technology. New inspection technology will have to be qualified for field inspection of welds.

Associated Risk. The risk is associated with the possibility that field-related fabrication and inspection of these large vessels, especially if advanced pressure vessel materials (2.25Cr-1Mo, 9Cr-1Mo class) are employed, will be beyond the limits of current technology and will not be achievable within the NGNP time frame. The qualification of advanced materials represents a significant risk for the NGNP program.

Recommendation. The ITRG recommends that the pressure vessel and associated irreplaceable piping be fabricated using materials for which the current database now exists. If this is not possible, then the ITRG recommends that a focused R&D program be initiated at the earliest possible date to evaluate the key issues related to fabrication, welding and inspection to determine whether it will be possible to qualify advanced pressure vessel materials in time for NGNP service. Reduction in power rating might be an additional option to the vessel size within acceptable limits.

NGNP Materials Program Response. The reactor pressure vessel (RPV) transportation and fabrication project was initiated in FY-05 and shutdown later that year at the request of DOE. Therefore, this risk area has not been fully evaluated, however, the materials development program agrees with the risk issues noted. It is true that there is greater risk associated with fabrication and inspection of the RPV at a remote construction site. It is also true that these risk issues are increased if a more advanced RPV steel is utilized rather than the normal steel used for LWR RPV construction (SA508/533) A study was initiated in FY-06 to partially resolve the issue of RPV steel selection from an NGNP design perspective. The results of this study will be available early in FY-07 and will be carefully considered in light of the ITRG recommendation.

Several analyses performed indicate that additional RPV cooling flow will be required to insure that the average wall temperature of the steel remains below 371°C to utilize LWR RPV steel for VHTR applications. This design feature will be used in the design of the Pebble Bed Modular Reactor (PBMR) demonstration plant being designed and constructed in the Republic of South Africa (RSA). However, the reliability of this system for all potential transient events is currently unclear. The issue of RPV cooling appears to be a greater consideration for potential prismatic designs than for pebble-bed designs.

Other potential issues have now been identified that directly affect the use of advanced steel (Grade 91, Modified 9 Cr-I Mo, Grade 22, 2-1/4 Cr-1Mo steels) for the VHTR RPV application. The Grade 91 steel is a higher strength steel compared to the normal SA 508/533 LWR steel, has better creep resistance and its use allows the RPV to be made with thinner wall sections and therefore lower thermal stresses. However, these superior properties of Grade 91 steel depend on the creation (by heat treatment) and maintenance of a uniform tempered martensitic microstructure throughout the service life of the RPV. Failure to maintain this microstructure following fabrication including hot bending, welding, forging or repair operations would seriously degrade the high temperature properties of the material and these issues have caused several failures in the field. In response to these issues, an ASME task group has recently developed more rigorous rules for postweld heat treatment of Grade 91 components. These rules have been incorporated as mandatory requirements in the 2005 Amendment to ASME Section I. The type of post weld heat treatment described, however, is difficult to correctly apply at a remote RPV fabrication site on a very large component.

Currently, Grade 91 steel is not available in large forged sections of the type required to fabricate a VHTR RPV of the size discussed for the NGNP. Therefore, assembly of plate material by welding would need to be used as the primary assembly method. Assuming that issues with thick section welding techniques associated with this material can be resolved, uniform heat treatment is still an issue that needs to be resolved for the reasons noted above.

There currently is little incentive to use Grade 22 steel for the VHTR RPV application because ASME Section III only allows the use of this material for Class I applications in the annealed heat treatment and this results in a material that does not have adequate strength for the RPV wall thicknesses being discussed.

1.4.4 ASME Codes and Standards

Technical Observation. Several of the NGNP concepts require either that existing materials be qualified for Section III service or that entirely new materials be developed. In addition, some of the NGNP designs allow for creep deformation in the Class I boundary.

Associated Risk. The risk is associated with the time that will be necessary for the qualification of existing or new materials for use in the Class I boundary. In addition, there will be risk associated with the allowance of creep deformation as a part of the Class I boundary design, both technical and regulatory. It is the judgment of the ITRG that the development and qualification of new materials for Section III, Class I service cannot be achieved in the time frame for the NGNP. Further, it is our judgment that the qualification of existing materials for Class I service where creep deformation is allowed represents an unacceptable risk for the program.

Recommendation. The ITRG recommends that the NGNP irreplaceable Class I boundary components be constructed using materials that are currently qualified for Class I service or that can be qualified without an appreciable data gathering program. The ITRG further recommends that qualification of new materials be limited to those that are either currently listed in Section II for Section VIII service or for which a database currently exists.

NGNP Materials Program Response. The materials development program agrees with these recommendations assuming that an aggressive schedule for plant construction is an over riding consideration; however, as noted in the discussion above the use of SA-508/533 LWR RPV steel for the NGNP RPV will cause limitations on the design of the plant which will need to be addressed. The other important issue involves a NERAC recommendation to design a prototype VHTR plant with lesser initial capabilities but with the ability to upgrade the capabilities of the plant over time. This could be more easily accomplished if the Class I boundaries were fabricated with a more advanced alloy because the Class I boundary components are not easily replaceable and the material used for these components establish long term limitations on possible higher power, higher temperature plant operation that may represent a desirable modification in the future. While the higher alloy content steel 9Cr-1MoV is included in ASME Section III, Subsection NH for Class I service, it is not necessarily adequately qualified for the full ranges of time and temperature of service that the plants may require. It may be possible to address these issues with design changes over time but this approach represents a tradeoff of reduced initial risk versus potential additional long term risk associated with possible long term plant upgrades.

1.4.5 Corrosion and Oxidation

Technical Observation. High-temperature operation of metallic components for times up to 60 years will result in the possibility of corrosion damage. This damage will most likely be associated with coolant contamination within the primary system or air oxidation on external surfaces. In addition, the development of oxidized surfaces may affect the overall thermal resistance of the system as a result of changes in emissivity.

Associated Risk. The risk is associated with potential changes in material properties as a result of corrosion-induced embrittlement.

Recommendation. This risk is judged to be minimal assuming that adequate coolant contaminant control is exercised.

NGNP Materials Program Response. The materials development program does not consider this risk to be minimal because it is currently unclear how effective coolant contamination control measures will be. Therefore, the program is planning to perform adequate testing to characterize changes that take place from corrosion induced embrittlement.

However, it is now clear based on a literature review that has been performed that this issue has not been a significant problem with VHTR plants that have operated in the past. These plants may not be representative of the NGNP design; however, the results noted from the review are encouraging. The program also believes that continued R&D in this area will be useful during licensing discussions with the NRC if questions arise concerning this issue.

The new issue of particular concern for NGNP is the impact of impure helium corrosion on the very thin sections that may be required within compact heat exchangers, if they are used to minimize system cost.

1.4.6 Microstructural Stability

Technical Observation. The exposure of metallic materials to high temperature for long periods of time may result in microstructural changes in pressure boundary materials that result in a degradation of material properties.

Associated Risk. The risk is associated with embrittlement of materials. However, with an adequate monitoring and inspection program, this risk is judged to be acceptable.

Recommendation. The ITRG recommends that the development program conducted for the NGNP include tests to identify thermal aging issues. Special consideration should be given to welds and heat-affected zones.

NGNP Materials Program Response. The materials development program agrees with this recommendation and investigation of these issues have been integrated into the program.

1.4.7 Graphite

Technical Observation. The NGNP may require use of new sources of graphite, the source of graphite for earlier gas reactor systems no longer being available. The qualification of new sources of graphite will require an extensive R&D and qualification program.

Associated Risk. The risk is associated with the potential that graphite performance under irradiation at high temperatures may limit the life of key core structural components. In addition, radiation-damage-induced distortion of core structural materials may result in an increase in bypass flow. The amount of bypass flow will have an important effect on fuel temperatures during operation. If excessive bypass flow is present, then the viability of a prismatic concept may be influenced. This risk is judged to be significant but not unacceptable with respect to safety.

However, cost risk associated with early replacement of core components may become significant and affect the overall economics (e.g., for structural components in a prismatic concept and reflector structures in the PBMR concept).

Recommendation. The ITRG recommends that additional work be done to quantify the expected evolution of bypass flow for a prismatic core concept more accurately. The relationship between graphite performance and bypass flow should be quantified. Radiation-induced graphite degradation generally increases with increased temperature and has to be investigated in detail with regard to maximum allowable coolant temperatures in the core region.

NGNP Materials Program Response. The materials development program agrees with these recommendations; however, compromises for graphite R&D may be required due to NGNP schedule limitations and budget. The optimal qualification program for graphite is very expensive and includes a lengthy commitment associated with test reactor irradiations required to fully characterize the new graphites under all possible conditions expected during the life of the plant and to predict the graphite performance based on validated modeling. The required irradiation schedules appear to be in partial conflict with the proposed license application schedule given in the PPMP. Possible ways to resolve these issues will be discussed in Section 4.1.

1.4.8 Extrapolation of Limited Data

Technical Observation. Use of advanced materials will require the development of an adequate database for materials used in the NGNP design. The current databases for advanced pressure boundary materials do not extend to the temperature ranges where some of the proposed NGNP concepts will operate. Use of materials at high temperatures may result in new deformation mechanisms (creep, creep-fatigue) becoming issues. The extrapolation of time-dependent data where fatigue is present represents a very significant challenge to the design.

Associated Risk. The risk is associated with the uncertainty in extrapolation of existing data to higher temperatures.

Recommendation. The ITRG recommends that, where possible, minimal extrapolation be used. In addition, care must be taken to ensure that extrapolation remains within the database deformation regime.

NGNP Materials Program Response. The materials development program agrees with these recommendations. The issues of maximum long-term usage temperature and negligible creep for Grade 91 steel are currently being investigated under contract from DOE to the ASME (see Section 1.7 for further information). However, it is believed that data extrapolation and determination of the temperature of on set of negligible creep are minimal issues compared to other issues noted previously regarding the use of Grade 91 steel for the NGNP RPV.

1.4.9 Advanced Materials Development

Technical Observation. Operation at a gas outlet temperature of 1000 °C will require use of nonmetallic materials in core structural applications and will require development of new materials for heat exchanger designs.

Associated Risk. The risk is associated with the time it will take for the development and qualification of new materials. It is judged that sufficient new material development and qualification cannot be achieved in time to support the NGNP construction permit application (2009).

Recommendation. The ITRG recommends that NGNP materials be limited to those currently qualified or that can be qualified with minimal effort. As discussed in earlier sections, this will require that the gas outlet temperature be reduced.

NGNP Materials Program Response. The materials development program agrees with this recommendation. This is the reason that the program has selected only the most mature high temperature metallic alloys, RPV materials, and non-metallic materials for qualification within the NGNP. No inherently new materials are being developed, though small variations to existing materials are being evaluated, including graphites from currently available coke sources, controlled chemistry variations of high temperature alloys, like Alloy 617, and minor architecture and processing variations of C-C composite already developed for other applications.

However, it has been concluded that the use of a large complex Intermediate Heat Exchanger (IHX) system regardless of the design that operates at a temperature greater than 850^0C may represent a significant risk for long term plant operational costs and plant reliability if it is fabricated using any mature metallic alloy currently available. The cost, the possible necessity of periodic replacement of the IHX, and the fact that plant operation (if an indirect cycle design is used) is dependent on reliable operation of the IHX are the primary reasons that the PBMR plant will operate with a direct cycle. These risk issues were determined to be more important for the PBMR plant design than other risk issues regarding possible nuclear contamination in the BOP. These issues need to be carefully considered when a direct or indirect cycle NGNP design is evaluated.

It has been concluded that it is impractical in the PPMP planning horizon for the NGNP to utilize C-C composite materials for construction of a large capacity IHX system.

A smaller heat exchanger that would isolate the plant from the hydrogen production facilities would be required if electricity and hydrogen production continue to be a part of the NGNP design. However, the failure or necessity for replacement of this smaller heat exchanger would not normally result in the extended shutdown of the plant in a direct cycle system.

1.5 NGNP Preliminary Project Management Plan

The *NGNP Preliminary Project Management Plant* (INL/EXT-05-00952, Rev 1, was issued in March 2006. The primary objective of this document is to identify those activities important to Critical Decision (CD)-1. CD-1 will be used as the primary basis for a decision on proceeding or not proceeding with the NGNP Project. The report contains a discussion of three planning options that involve a range of programmatic risks and approaches to mitigating risk; high level schedule logics for design, construction, licensing and acceptance testing associated with these planning options; an assessment of current technology development plans to support CD-1, a rough order of magnitude cost evaluation for each of the planning options and a project management concept built around capabilities at the INL.

It is recommended that Option 2 (Balanced Risk Approach) be used as a project management basis. The high level schedule logic and rough order of magnitude funding profile required to support this option is given below.

The focus of Option 2 is to reduce the number of design alternatives as early as practical so that R&D activities may be focused on the reference design, thereby reducing the scope and time for completing necessary R&D.

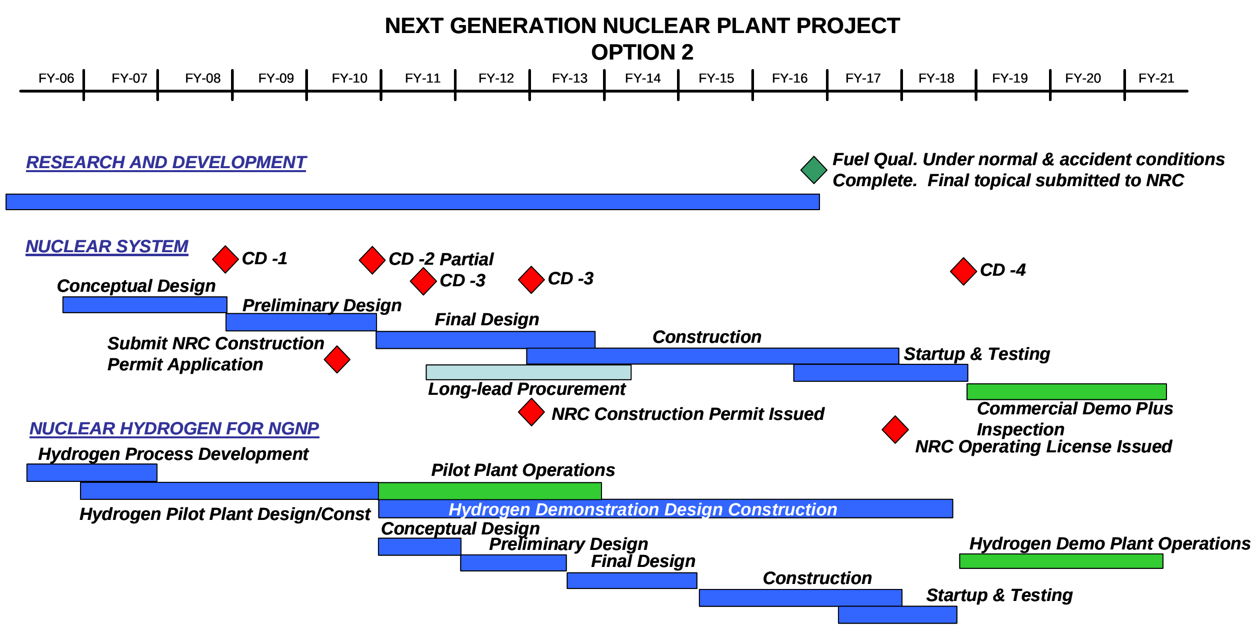
The design and construction schedule is conservative. The approach is to design and then construct, with limited overlap, to minimize project cost risk. There is about one to two years margin in the preliminary design and construction schedule that subsequently might be removed once the technology development is more mature to support accelerated system and equipment design, and procurement activities.

In Option 2, construction permit application is made to the Nuclear Regulatory Commission at the earliest practical date consistent with having a sufficient technical basis for the anticipated performance of the TRISO fuel and plant materials. This schedule for a construction permit application requires early development of design to about 15% completion following early trade-off studies to settle fundamental design concept issues (e.g., nuclear system operating temperature; direct v. indirect cycle; core design concept; intermediate loop heat transfer concept) and establish detailed design input requirements. With this approach, formal resolution of the licensing basis for NGNP can be achieved at the earliest practical time, greatly reducing the risk of proceeding with final design and construction activities. Similarly, operating license application is made at the earliest practical time consistent with qualification of the TRISO fuel design and plant materials.

Option 2 includes a two to three year operating period following initial operations to represent an operating cycle of a commercial power reactor, followed by disassembly and inspection to directly confirm design performance. This proof-of-principle operating period provides the basis for subsequent commercialization decisions by commercial industry. This period of confirmatory operation is not available in Option 1 prior to 2021 due to the later design and construction start.

Table 1. Option 2 – Rough Order of Magnitude Estimate Funding Profile

	Prior Year	**FY 06**	**FY 07**	**FY 08**	**FY 09**	**FY 10**	**FY 11**	**FY 12**	**FY 13**	**FY 14**	**FY 15**	**FY 16**	**FY 17**	**FY 18**	**FY 19**	**FY 20**	**FY 21**
Nuclear System	44	45	100	155	159	198	155	233	284	163	126	146	134	26			
Hydrogen Production	4	25	25	25	29	37	36	32	26	16	17	20	20	5			
Total	48	70	125	180	188	235	190	265	311	180	142	166	154	31			



1.6 Clarification of High level Issues

The DOE Acquisition Management System is a systematic management process that translates mission needs and technological opportunities into reliable and sustainable facilities, systems, and assets to meet a required mission capability. This system is organized by project phases and "Critical Decisions" (CDs) or quality gates. The CDs provide a review and approval mechanism to determine if a project is sufficiently matured to proceed to the next project phase. The Deputy Secretary of Energy serves as the secretarial acquisition executive (SAE). As the SAE, he promulgates Department-wide policy and direction, and personally makes critical decisions for major system projects.

The project phases (Initiation, Definition, Execution, and Transition/Closeout) represent a logical maturing of broadly stated mission needs into well-defined technical, system, safety, and quality requirements, and ultimately into operationally effective, suitable, and affordable facilities, systems, and other end products. Each phase provides at least one iteration of requirements definition and refinement, trade studies and alternatives evaluation, design, identification of uncertainties and/or needed technology development, testing, and mockup, resulting in a relative mature level of system architecture at the completion of the project. Each iteration increases in maturity from the previous until a mature deployment of the system architecture or finished useable end product is completed.

The Initiation Phase for the NGNP Prototype Project is essentially complete, based on approval of the Mission Need Statement and approval of CD-0, *Approve Mission Need*, October 18, 2004.

With CD-0 approval, the project has moved into the Definition Phase, where alternative concepts based on user requirements, risks, costs, and other constraints will be analyzed to arrive at a recommended alternative. Because design and project activities were not funded during FY-05, for planning purposes, project activities normally subsequent to CD-0 approval will be assumed to be at the beginning of FY-06.

Beginning in FY-06, the reactor technology alternatives will be analyzed through design and trade studies and conceptual design studies. The Conceptual Design Report, due at the end of the definition phase will document the conceptual design for the selected alternative, including a rough order of magnitude range for the project cost and schedule. During this phase, more detailed planning and analysis will be performed, which further defines the required capability. These efforts include conceptual design, requirements definition, risk analysis and management planning, a value management assessment, preliminary project execution plan development, and development of the acquisition strategy.

The GIF provided initial screening of technology alternatives. From this evaluation, a very high-temperature gas-cooled reactor was identified as the nearest-term Generation IV technology for producing both electricity and hydrogen. The primary gas-cooled reactor alternatives include a prismatic or a pebble bed reactor. Alternative and trade studies that evaluate these technology alternatives, power conversion, and balance of plant design will be conducted during the definition phase through trade studies and Conceptual Design. These studies are necessary to develop the project technical and functional requirements. The Conceptual Design, in addition to providing requirements analysis, alternatives evaluation, hazards analysis, project risk evaluation, and information needed for R&D, also provides the first basis for a cost range, schedule, and performance requirements for the project.

Reactor technology trade studies and balance of plant and power conversion studies are planned to be subcontracted with vendors and industry consultants. Various stakeholders will be consulted for reviews of the study reports to ensure adequate industry involvement. It is anticipated that design and analysis work performed previously for gas-cooled nuclear systems will be used to the maximum practical extent.

In addition to the gas-cooled reactor technologies, a liquid-salt-cooled technology was considered during the Initiation Phase. This option was considered by the GIF and rejected for immaturity of technology. The Independent Technology Review Group also concluded in 2004 that a liquid-salt-cooled reactor was too immature for consideration as an NGNP technology.

A major activity that will also be started during the Definition Phase is risk planning and management. Certain significant technology risks have already been identified by the technology groups and the ITRG, and a minimal, selected level of risk planning management has been initiated. This process will be further implemented both for technical risks and project risks during Conceptual Design. Many of the risks will not be apparent until some design work is accomplished.

1.7 DOE Initiative to Address ASME Code Issues

1.7.1 Introduction

The DOE Generation IV Nuclear Energy Systems Program includes six reactor systems: the Supercritical Water Reactor (SCWR), the VHTR, the Lead-cooled Fast Reactor (LFR), the Gas-cooled Fast Reactor (GFR), the Sodium-cooled Fast Reactor (SFR), and the Molten Salt Reactor (MSR)[3]. The NGNP is a helium-cooled VHTR reactor concept with an outlet temperature in the range of 850-950°C, and is of primary interest to DOE. The reactor life is targeted for 60 years. Design options may permit and/or require numerous components to be replaceable; however, the reactor pressure vessel is among those that will not be replaced. Reactor internals will experience very high temperatures, while temperatures of out of core structural metallic components may be reduced by innovative design concepts. Nuclear structural component construction in the U.S. complies with Section III of the ASME Boiler and Pressure Vessel Code[4], although licensing is granted by the NRC. The extensive use of graphite, particularly its use as a structural material in the NGNP is novel. A number of additional technical topics were identified by DOE, ORNL, INL, and ASME to have particular value with respect to the ASME Code. A three-year collaboration between DOE and ASME was established that addresses twelve topics in support of an industrial stakeholder's application for licensing of a Generation IV nuclear reactor. Efforts to address the first five tasks are currently initiated. The majority of these tasks are relevant to action items within ASME Section III Subsection NH, and the nature of the topics inherently include significant overlap, and in some cases parallel activities on the same issue. These tasks are summarized below.

1.7.2 Task 1: Verification of Allowable Stresses

Stress allowables for Alloy 800H include data for Alloy 800, and differences exist for stress allowables found in Subsection NH and RCC-MR for Grade 91. There are also discrepancies between material properties as implemented in allowable stress values in Subsection NH and Section II Part D which should be in agreement. The review of stress allowables for base metal and weldments of Alloy 800H and extension of time-dependent allowables to 900°C (which would required data at least 25°C above the temperature at which allowables are set in order to achieve more reliable extrapolation to longer times) are desired, if possible. Similarly, review of stress allowables for Grade 91 Steel is desired.

The task will require formal access and use rights of the various materials databases. The original database needs to be assembled and reviewed, including methods used to set the time-dependent allowables. Comparison of European and Japanese databases and the methods and procedures used by these sources to set allowable stresses are needed to provide guidance and comparison on how to set allowable stresses for ASME Section III Subsection NH. For Alloy 800H, the U.S. database needs to include existing data produced up to 925°C, including both creep and stress rupture data. An updated compilation of the creep and rupture data for Grade 91 needs to be assembled, especially for up to 300 mm thick plate, forgings, and heavy wall piping. Consideration of post-weld heat-treatment (PWHT) effects needs to be made. Assessment of alternate procedures for describing creep and stress-rupture for conditions of concern to the NGNP, namely 60 year plant life, is also required; procedures developed by the Pressure Vessel Research Council need to be considered as well. The current allowables need to be compared to the results of the reassessment, and a recommended course of action made with respect to ASME Section II-Part D and III-NH, including identification of supplementary testing required to address the needs outlined in the High Temperature Metallic Materials Test Plan for Generation IV Nuclear Reactors [5].

1.7.3 Task 2: Regulatory Safety Issues in Structural Design Criteria for ASME Section III Subsection NH

The NRC licensing review of the Clinch River Breeder Reactor Plant (CRBRP) in the 1970's and 1980's identified a number of concerns, including but not limited to weldment safety evaluation, notch weakening, and creep-fatigue evaluation[6]. The major fundamental regulatory safety need was improvement of the criteria to prevent creep cracking. The need to build confidence in the regulatory community that the resulting designs will have adequate safety margins is critical. Compilation and storage of reports describing confirmatory program plans that were jointly developed by the NRC and the CRBRP are needed. A review of all the safety issues relevant to Subsection NH is required, including the generation of a historical record that includes a detailed description of how Subsection NH currently addresses these issues or not; further, identification of additional safety concerns within NH for application to very high temperature service is needed. The review will serve as a foundation to initiate communications with the NRC on these issues, and facilitate future consultation with the NRC in improving, developing, and confirming design and fabrication procedures, strain limits and material design curves.

1.7.4 Task 3: Improvement of ASME Section III Subsection NH Rules for Negligible Creep & Creep-Fatigue of Grade 91 Steel

Significant differences in prediction of creep strain under monotonic loading exist between Subsection NH and RCC-MR[7]. These differences are critical in satisfying the insignificant creep criteria and are likely due to extrapolation of data from 500-600^{o}C to lower temperatures applicable to the NGNP RPV, 370-450^{o}C. The current approaches available to define negligible creep need to be reviewed and their applicability verified for use to Grade 91 steel. Material data available in France and the U.S are needed. The methodology, data, and additional tests required to support the definition of negligible creep conditions for Grade 91 steel need to be identified.

The damage envelopes for creep-fatigue are significantly different for Grade 91 steel in Subsection NH and RCC-MR codes[8]. The differences have yet to be explained by scatter in data, variation between heats, whether the damage envelope is procedure dependent (e.g., definition of stress and creep damage during hold times), material softening, or other yet unidentified causes. Hence, a critical comparison of ASME Section III Subsection NH and RCC-MR creep-fatigue procedures is needed. Comparisons are desired on the basis of experimental test results available from Japan, France and the U.S.; particular attention is required in the definition of safety factors and creep-fatigue damage envelope procedures. Assessment of whether or not the material data presently available in nuclear codes are thought to be sufficient and valid is required, including recommend improvements to existing procedures and definition of a test program to validate the improved procedures.

1.7.5 Task 4: Updating of ASME Nuclear Code Case N-201

The scope of Code Case N-201 needs to be expanded to include materials with higher allowable temperatures, extend the temperature limits of current materials if possible, and to confirm whether the design methodology used is acceptable for design of core support structure components at the appropriate elevated temperatures. The maximum operating temperatures required for HTGR metallic core support structures must be identified in a review of data made available by AREVA, GA, PBMR, DOE, and other available sources. Operating parameters including but not limited to temperature, pressure, time, and environment need to be defined. Candidate materials need to be identified and prioritized for use as metallic core support structures within the defined operating parameters. The design methodology used, which is primarily based on ASME Section III Subsection NH, needs to be critically reviewed for application to Generation IV reactors, specifically the NGNP. Recommendations for inclusion of new materials or extension of times and temperatures for current materials are required. If necessary, recommended changes and/or additions to design methodologies should be made; note, Task 7 closely parallels this portion of Task 4 - namely the evaluation of Subsection NH and Simplified Methods. Gap analysis on material data needs is required, including definition of supplemental testing required to support determination of material design curves (e.g., creep rupture, creep-fatigue, etc.).

1.7.6 Task 5: Creep-Fatigue Procedures for Grade 91 Steel and Hastelloy XR

Task 3 and 7 include intentional parallel activity in evaluation of creep-fatigue procedures for Grade 91 (Task 3) and Alloy 617 & 230 (Task 7). The intent in this task is to compare procedures used on two different classes of alloys, and develop modified or new procedures for application to universal creep-fatigue modeling and alloy specific procedures. The task will require that formal access and use rights of the various materials databases be secured, followed by an evaluation of the state of existing data to determine if more data are necessary. Candidate database sources include Grade 91 steel data generated for the Japanese demonstration fast breeder reactor and Hastelloy XR used in the High Temperature Test Reactor (HTTR). Creep-fatigue criteria need to be summarized based on existing international nuclear codes, e.g. Subsection NH, RCC-MR, and Monju Design Codes.

A comparison of creep-fatigue damage evaluation procedures is required, including assessment methods of strain range, initial stress and relaxation behavior, formulation of creep damage, strain rate and hold time effects, methods used in partitioning plastic, elastic, and creep strain, environment effects, and wave shape effects. A recommended testing program to generate required supplemental data needs, and model verification is needed.

1.7.7 Task 6: Graphite and Ceramic Code Development

Several new graphite and ceramic materials are being identified in new nuclear power plant designs because of high temperature operating environments[3]. Requirements must be established, and environmental effects must be considered, including irradiation to support emerging designs. Support for development of ASME Code requirements for qualification of non-metallic components, including carbon-carbon composites and ceramics such as SiC/SiC composites for new nuclear power plant designs is required. In ASME Section III, the Project Team on Graphite Core Components is currently drafting design rules for the graphite core internal for a VHTR, and will eventually develop design rules for the composite and ceramic core internals. An independent peer review of the draft code rules, including a framework of probabilistic design approaches, fracture mechanics assessment methodologies, qualification of components, material specification/traceability, examination or inspection methods and acceptability criteria, and irradiation considerations in design rules are needed. Workshops are needed to inform ASME staff and committee members of the technical and background information required regarding the behavior and properties of brittle materials such as graphite and ceramics, including the effects of neutron damage on their properties. Codification of such brittle materials within ASME has yet to be accomplished, and will require a review of the adequacy of the current data base, recommendations regarding further data needs to support the design rules, and a thorough review and endorsement of test plans and programs, both within the U.S. and internationally.

1.7.8 Task 7: NH Evaluation and Simplified Methods

The design of Class 1 nuclear structural components that are deemed to operate at elevated temperature, ~371 ^{o}C or greater for ferritics and 427 ^{o}C or greater for austenitics, are designed in the U.S. with ASME Section III Subsection NH Code[4]. Subsection NH is also the foundation for Part B of Code Case N-201, which addresses elevated temperature design of core structures. With the demise of new nuclear programs in the late 1980's after the 1979 Three Mile Island incident, activity and development of elevated temperature code within Section III virtually ceased. Nations other than the U.S. continued R&D activities and gained valuable experience in elevated temperature design. As such, a review and comparison of current design methods in ASME Section III Subsection NH, RCC-MR, BC5500, and Japanese codes are needed. Other assessment codes and standards, such as API 579 and R5, should also be included. Topics should include a range of design and analysis methods consisting of various levels of simplified analysis techniques: elastic analysis, limit load analysis, simplified inelastic shakedown and ratcheting analysis, and full inelastic analysis. Methods that either eliminate stress classification or provide a consistent and acceptable methodology for properly classifying stresses are desired.

Code developers and researchers worldwide generally recognize that the current linear damage accumulation rule for creep-fatigue has significant shortcomings, particularly at higher temperatures and longer times. Various improvements, such as those based on ductility exhaustion and damage rate concepts, have been proposed, but none have been backed by sufficient R&D to allow their adoption as a replacement for linear damage in ASME Section III Subsection NH. These shortcomings should be remedied for Generation IV systems[9]. Advances in thermo-mechanical fatigue life prediction methods in the last two decades have resulted from addressing mechanisms of damage nucleation and propagation; alternative creep-fatigue models should consider aging effects, formation and repeated fracture of protective oxides, internal oxidation, crack growth, and strain-based methods. Application to Grade 91 steel and Alloy 617/230/800H materials is required. Also, a critical evaluation of the data and methodologies is needed in light of likely VHTR operating conditions. The adequacy/inadequacy of existing life prediction methods, extrapolation techniques, and test data need to be summarized, and subsequent recommendations made to address shortcomings.

1.7.9 Task 8: Identification of Testing Needed to Validate Elevated Temperature Design Procedures for the VHTR

ASME Section III Subsection NH Code requires a combination of component testing, testing of coupons samples, and testing of basic structures with component like features in order to validate design procedures. A critical review of VHTR design features (block and pebble designs) from the viewpoint of elevated temperature design is needed, which summarizes necessary design procedures that require validation. The focus should be on major components such as reactor vessel, internal structures, piping, etc. The review should include international accomplishments in validation of design procedures and related activities for which results would contribute to the validation. Recommendations on future tests required for validation are needed, and should address all known failure modes addressed in NH and any additional failure modes identified from the review.

1.7.10 Task 9: Environmental and Neutron Fluence Effects

The ASME Code currently does not address environmental and neutron fluence effects; these effects must be addressed in any nuclear design, particularly the design of metallic core structures. Inclusion in relevant design procedures must be deemed acceptable by the NRC in order to obtain construction and operating licenses. Historically, industry and the DOE have addressed these issues outside of the ASME framework; as such, the approach taken for liquid metal fast breeder reactor components may provide a basis for supplementing the rules that are now in ASME Section III Subsection NH.

A critical review of available VHTR conceptual design information is essential to the development of supplemental design criteria, including materials of construction, ranges of expected temperatures, gas compositions, neutron fluences, capabilities for component replacement, and the like. The review should include assessment of the current U.S., European, and Japanese materials selection criteria, design codes, and conceptual designs of reactor pressure vessels and metallic core internal structures for gas-cooled reactor applications. Other sources such as the ITER Structural Design Code should also be considered[10]. A summary of recommendations should be provided, including an assessment of the adequacy of the U.S. Generation IV materials program to develop the materials performance characteristics needed to formulate supplemental rules, and if applicable, a supplemental testing programs to support design criteria that address environmental and neutron effects.

1.7.11 Task 10: ASME Code Rules for Intermediate Heat Exchangers (IHX)

Several versions of VHTR reactors include an intermediate heat exchanger (IHX)[11]. The IHX in the NGNP reactor concept serves to transfer energy to a secondary plant dedicated to the production of hydrogen and/or electricity. The intent of this task is to determine how and where within ASME codes and standards the IHX, safety valve, and similar components should be addressed. Many technical questions need to be addressed to determine how the function of such components affects the plants, safety, etc. The specific type of IHX has yet to be selected, e.g. tube/shell, plate/fin, or a micro-channel type IHX. All aspects of possible IHX concepts must be identified, including materials, design, fabrication, examination, testing, overpressure protection and in-service inspections that are used in the construction and operation of representative heat exchanger pressure boundaries and internals designs. Heat exchangers with working fluid temperatures at the upper end of the creep regime for their materials of construction need to be emphasized, e.g. those requiring the use of Alloys 617 and 230. Identification of alternative concepts, if available, is desired. Emphasis needs to be placed on (a) design criteria including methods, if any, for evaluation of cyclic life, (b) construction codes of record and designated pressure boundaries, (c) qualification of materials and fabrication techniques for the intended service, and (d) environmental effects.

Once the lead industrial system reactor design and candidate component suppliers are selected and the conditions to be evaluated are representative, an evaluation of IHX design approaches with respect to the evaluation of past experience accomplished in former paragraph is needed. This includes scoping analyses to identify critical design configurations and loading conditions. All operational and safety aspects of a gas-cooled reactor design need to be evaluated in light of past practice and experience. Recommend changes and additions to current construction codes or features of a new construction code must be identified. Candidate Codes include ASME Section III Subsection NB, NC, ND, and NH, and their respective elevated temperature Code Cases, and Section VIII Div 1 and Div 2.

1.7.12 Task 11: Flaw Assessment and Leak before Break (LBB)

A review and status report of international design criteria and fitness for service rules presently available is desired. A synthesis of approaches available for LBB assessment and more generally for fracture mechanics methods (crack growth and stability calculations) is needed. This shall include clarification of the extent that existing methods are applicable to VHTRs. Available material properties need to be identified, including but not limited to SA508/SA533 and Grade 91 steel. Identification of specific test programs to be carried out to establish a set of material properties for use in design rules and fitness for service assessment methods and validation of LBB and assessment methods is needed. The results are needed to serve as a basis for discussion with U.S. NRC before launching significant activities on the subject.

1.7.13 Task 12: Improved NDE Methods for Metals

Identification of appropriate new construction and in-service NDE methods for examination of metallic materials (e.g., acoustic emission, ultrasonic) is desired. This will include (1) defining maximum acceptable flaw types and sizes based on the Load and Resistance Factor Design (LRFD) approach and material properties of candidate materials that are available, and (2) defining nondestructive examination methods needed to detect sub-critical flaws of the size and type defined in (1) above in pressure equipment during initial construction and for periodic examination during the life of the equipment.

The need for new methods is anticipated to reliably detect smaller discontinuities than those of concern for the current generation of pressure equipment. The methods will include the characterization of uncertainties in a manner that is suitable for reliability-based LRFD development for overall plant design. Some methods to be considered include but are not limited to (1) ultrasonic time-of-flight-diffraction, including detailed guidance for application, and (2) ultrasonic phased arrays, including definition of requirements and guidance for application.

Overall, the outcome of this task will provide technical information and background to resolve concerns and assist codes & standards committees and jurisdictional authorities in adopting improved NDE methods into ASME codes and standards.

1.7.14 Conclusions

Numerous activities of value to the DOE Generation IV Reactor Systems Program have been identified. A collaborative agreement between DOE and ASME has been established to facilitate technical review of code allowables, material databases, regulatory needs, and design procedures, expansion of ASME Section III Code to include brittle materials such as graphite and SiC/SiC composites, and an independent critical review is also planned. Initial efforts are also planned to consider environmental and neutron fluence effects and intermediate heat exchangers in appropriate ASME Code for nuclear applications. Recommendations to additional testing requirements to support design procedures, as well as modification/additions of design procedures are expected to accelerate the codification process to support Generation IV reactor systems.

1.7.15 Affiliated Organizations

The following organizations have been selected by the ASME to lead Tasks 1-5 activities:

- Task 1- University of Dayton Research Institute
- Task 2- O'Donnell Engineering
- Task 3- AREVA-Framatome ANP
- Task 4- Westinghouse Electric
- Task 5- Japan Atomic Energy Agency

2. NGNP Materials Program Scope

The objective of the NGNP Materials R&D Program is to provide the essential materials R&D required to support the design and licensing of the NGNP and balance of plant excluding the hydrogen plant. The following materials R&D program areas are currently addressed in the R&D workscope being performed or planned in the approximate order of priority based on current DOE NGNP direction:

1. Qualification and testing of nuclear graphite and carbon fiber/carbon matrix composites for use in the NGNP. Adequate models of the irradiation induced dimensional and material property changes are needed.

2. Development of improved high-temperature design methodologies (HTDMs) for the NGNP metallic alloys. This activity includes support for development of ASME Code Cases relevant to the license application of the NGNP and research into the complex creep/fatigue/environment interactions and joining technologies associated with the use of these alloys in the NGNP, and development of guidance not covered specifically in ASME Code Cases. Materials issues associated with the intermediate heat exchanger (IHX) and the metallic components within the RPV are covered in this task.

3. Expansion of ASME Codes and American Society for Testing and Materials (ASTM) Standards in support of the NGNP Materials R&D Program.

4. Development of an improved understanding of the environmental effects and thermal aging of the high-temperature metallic alloys to be used in the NGNP.

5. Qualification and testing of the silicon carbide fiber/silicon carbide matrix composite materials needed for the NGNP. This area will not be continued in FY-07.

6. Development of a materials handbook/database in support of the Generation IV Materials Program.

7. Support of a program to study, design, test, and qualify potential candidates for use as NGNP metallic internals.

8. Support of a program to study, design, test, and qualify insulation, valves, bearings, seals, and other components as required.

9. Support of program(s) for materials research that directly supports the development of the NGNP that should be performed at universities

These issues are being addressed by direct DOE funding to national laboratory organizations (primarily the INL and ORNL), within the collaborative context of the GIF and by university funded programs. The subsections noted below on the program organization, materials R&D issues included in the program, Materials Review Committee, relationship with the GIF, International Nuclear Energy Research Initiative (INERI) related programs and university programs are intended to further clarify the scope of the materials program.

2.1 Program Organization

The organizational structure currently used for the management of the Materials R&D program is given in Figure 1, though other organizational structures may evolve as the program matures. The NGNP

Program is the primary interface with DOE. The NGNP Materials R&D Program Manager interfaces with the Generation IV Materials National Technical Director (NTD), the NGNP Program System Integration Manager (SIM) and the principal investigators to establish the program elements. Input, interface, and recommendations are obtained from the MRC, the Materials Quality Assurance Program (QAP), and the Generation IV Materials and Components PMB. Work Packages and detailed program elements are based on DOE guidance and available funding. Program execution is performed by principal investigators who are scientific specialists related to the specific program area being addressed.

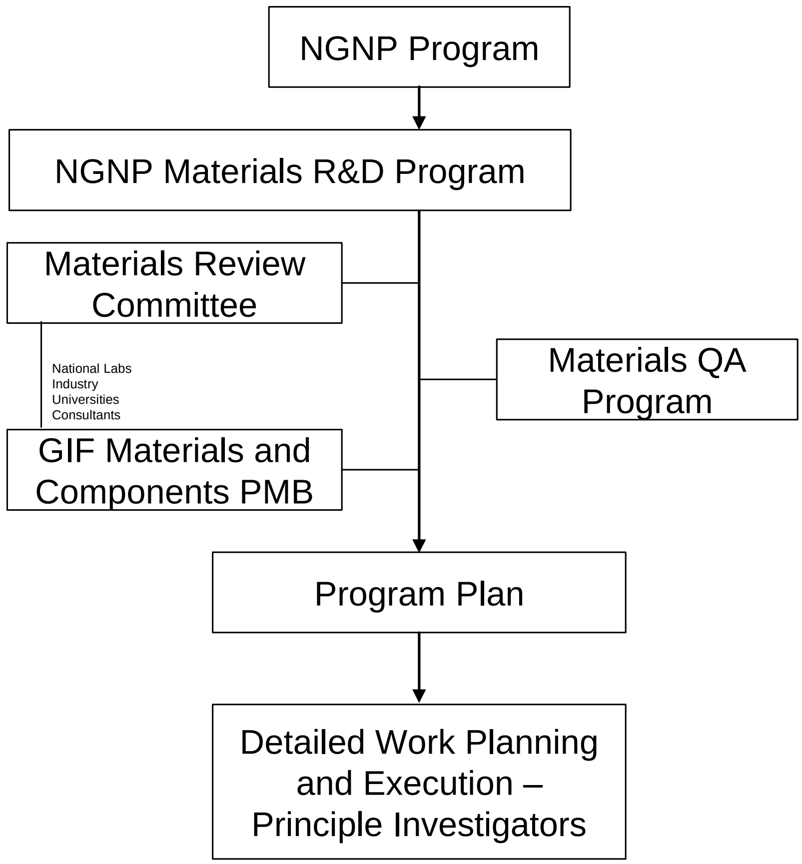


Figure 1. NGNP Materials Organization Structure.

2.2 Significant NGNP Materials R&D Issues Included

As noted above, there are four significant materials R&D issues currently being addressed in the program and these are noted below:

1. Qualification and testing of nuclear graphite and carbon fiber/carbon matrix composites for use in the NGNP. Adequate models of the irradiation induced dimensional and material property changes are needed.

2. Development of improved high-temperature design methodologies (HTDMs) for the NGNP metallic alloys. This activity includes support for development of ASME Code Cases relevant to the license application of the NGNP and research into the complex creep/fatigue/environment interactions and joining technologies associated with the use of these alloys in the NGNP, and development of guidance not covered specifically in ASME Code Cases. Materials issues

associated with the intermediate heat exchanger (IHX) and the metallic components within the RPV are covered in this task.

3. Expansion of ASME Codes and American Society for Testing and Materials (ASTM) Standards in support of the NGNP Materials R&D Program.

4. Development of an improved understanding of the environmental effects and thermal aging of the high-temperature metallic alloys to be used in the NGNP.

These areas are changed or modified as required to support the NGNP Program and will be discussed in greater detail in Section 4.

2.3 Materials Review Committee

A Materials Review Committee (MRC) was established in FY-04 as a senior independent review body for the materials R&D program. The MRC is chaired by Russell Jones who is an independent consultant and is operated under a charter approved by the NGNP Program and the MRC. The committee meets as required to review NGNP materials R&D reports, discuss program direction and make recommendations. The committee is composed of specialists who are independent consultants or employed in by private industry, universities or national laboratory organizations. Information inquiries concerning MRC activities and meeting minutes can be obtained by contacting Russell Jones at matsci@aol.com.

2.4 Relationship with the Generation IV International Forum

The primary mechanism for international collaboration for materials R&D activities in support of the VHTR is through the GIF. The GIF is an international effort to advance nuclear energy to meet future energy needs of ten countries–Argentina, Brazil, Canada, France, Japan, the Republic of Korea, the Republic of South Africa, Switzerland, the United Kingdom, the United States–and the European Union. Russia and China have petitioned to join GIF and the GIF Policy Group has voted unanimously to accept both countries. There will be an exchange of letters and each will sign the GIF Charter early in FY-07.They then will have a year to sign the GIF Framework agreement, after which they can join System and Project Arrangements.

These partners have agreed on a framework for international cooperation in research for a future generation of nuclear energy systems, known as Generation IV. Generation 1 nuclear reactor systems are considered to be early prototype plants such as Shippingport, Dresden, Fermi I and Magnox. Generation II plants are considered to be the current generation of commercial nuclear plants that are currently producing electricity today. These plants include current PWR, BWR, Canadian Deuterium-Uranium, and AGR plants. Generation III plants are considered to be advanced LWRs and include Advanced Boiling Water Reactors and System 80+ PWR plants. Generation IV plants have not been commercially operated to date and are envisioned to have the following general characteristics: highly economical, enhanced safety, minimal waste and proliferation resistant.

The GIF partners noted above have joined together to develop future generation nuclear energy systems that can be licensed, constructed and operated in a manner that will provide competitively priced and reliable energy products while satisfactorily addressing nuclear safety, waste, proliferation, and public perception concerns. The objective is to have these systems available for international deployment by about 2030 when many of the worlds currently operating nuclear plants will be at or near the end of their operating lifetimes.

Nuclear energy research programs around the world have been developing concepts that could form the basis for Generation IV systems. Many concepts have been developed including the VHTR concepts that include the NGNP. Collaboration on R&D to be undertaken by GIF partners will stimulate progress toward the realization of such systems.

The primary mechanism for collaboration of materials R&D for the VHTR is through the Materials and Components PMB. This board is currently composed of members from France, Switzerland, Japan, Korea, South Africa, the United Kingdom, the United States, and the EU and meets on a nominal semi-annual basis in various locations in the world. This board has the charter to address each materials R&D program area noted previously and is developing detailed collaboration plans for each of these areas. These plans are being developed in the same approximate order of priority noted in Section 2.2. It is currently envisioned that this process will not be fully developed and implemented until 2007, however, as each plan is developed, implementation of collaboration activities will begin immediately. Currently, the collaboration plan for nuclear graphite R&D has been developed. Plans related to ceramics and composites, high temperature alloys and design methodology and component testing are in draft form.

It is currently envisioned that collaboration will involve (as a minimum) the establishment of coordinated test and irradiation programs, coordinated purchase of testing materials, coordinated use of specialized testing facilities, coordinated support for the establishment of an integrated Generation IV materials data base and coordinated support of codes and standards committees. Other collaboration areas may be developed as the materials R&D program fully matures.

It is expected that these collaboration activities will result in a spirit of cooperation between the participating countries, the acceleration of design and licensing activities of VHTR systems and the reduction of the cost for materials R&D.

Two GIF M&C PMB meetings were held during FY-06. The first meeting was held in Daejon, Korea in April and was hosted by the Korean Atomic Energy Research Institute (KAERI). The second meeting will be held in Johannesburg, South Africa, in September and will be hosted by the Pebble Bed Modular Reactor (PBMR) Company. The focus of PMB activities in FY-06 was to finalize the collaboration plan for graphite R&D and to develop similar plans for high temperature metallic materials and design methods, structural composites and large component testing systems. Minutes of meetings and other information are available by contacting the secretary.

2.4.1 FY-07 Activities

Meetings during FY-07 have not been planned to date.

2.5 INERI Collaboration Programs

International Nuclear Engineering Research Initiatives (INERI) are designed to allow a free exchange of ideas and data between U.S. and international researchers working in similar research areas. This international agreement encourages strong collaborations between research institutions where a benefit to both countries is anticipated. One INERI collaboration has been approved by the DOE between the United States and France. Other INERI collaborations are being discussed.

A three-year INERI program 2005-2007 between U.S. and French research institutions (INL, ORNL, PNNL, CEA, and University of Bordeaux) was approved for R&D of SiC/SiC composites. The proposed research has investigated the issues surrounding the development of tubular geometry SiC/SiC composite material for control rod and guide tube applications. Mechanical, thermal, and radiation-damage response of the French fabricated tubular composites will be studied during this time. A progress report for the period from January, 2005 to October, 2005 is given in Appendix A.

It is probable that the US will withdraw from this program in 2007 because, as noted previously, R&D on SiC/SiC composites will not be funded as a part of the US program in FY-07.

The following NGNP materials related INERI proposals have been submitted to DOE but have not been approved to date:

1. US-Japan proposal related to C/C composite materials for control rod structures for NGNP

2. US France proposal related to high temperature alloy aging and environmental effects

3. US Korea proposal related to irradiation effects, environmental effects and development of the materials handbook for high temperature alloys related to the development of the NGNP

2.6 University NERI Related Collaboration Programs

The NGNP Materials Program is currently providing supplemental funding associated with a Nuclear Engineering Research Initiative (NERI) program initiated by DOE at the University of Michigan. The project objective of this work is to define strategies for the improvement of alloys for structural components, such as the intermediate heat exchanger and primary-to-secondary piping, for service at 1000°C in the He environment of the NGNP. Specifically, the University of Michigan will investigate the oxidation/carburization behavior and microstructure stability and how these processes affect creep. While generating this data, the project will also develop a fundamental understanding of how impurities in the He environment affect these degradation processes and how this understanding can be used to develop more useful life prediction methodologies. The last quarterly progress report for this project for the period from April to June, 2006 is given in Appendix B.

3. Summary of Current VHTR Designs and Materials Issues

This section describes the VHTR designs in summary form which have been developed or are being developed since 1996. The designs noted will undoubtedly have a major impact on recommendations received from trade studies of key NGNP design features. The designs discussed in this section include the following:

1. The Gas Turbine-Modular Helium Reactor (GT-MHR) conceptual design in the design description report released in July 1996 by General Atomics (GA) as report 910720, Revision 1 and INEEL/EXT-03-00870 Rev 1, NGNP Point Design- Results of the Initial Neutronic and Thermal Hydraulic Assessment During FY-03.

2. The Framatome-ANP High Temperature Reactor Concept described most recently at ICAPP 05 in Seoul Korea in May 15-19, 2005 in Paper 5254

3. The Pebble Bed Modular Reactor (PBMR) concept currently being developed in South Africa by PBMR (Pty) Ltd.

All of these designs utilize TRISO fuel, graphite moderation and the utilization of high temperature helium coolant in the primary system in the 800^0C -900^0C temperature range. All of the concepts utilize various passive neutronic design features which result in a core with relatively low power density and a negative temperature coefficient of neutron reactivity. The concepts proposed utilize some type of Brayton Cycle energy conversion system in the secondary system that would efficiently produce electricity from high temperature gas using various combinations of gas turbines, compressors and electrical generators. Some of the concepts also utilize a steam cycle to produce electricity from the lower temperature process gas available. All of the concepts could be used as a basis for the NGNP VHTR concept with relatively minor modifications.

The results of an analysis performed at the INL during 2006 are also included (INL/EXT-06-11057, Reactor Pressure Vessel Temperature Analysis for Prismatic and Pebble-Bed VHTR Designs) in this section. These results were performed because the expected high operating temperatures of the NGNP place significant constraints on the choice of reactor pressure vessel materials. The reactor design determines the maximum operating temperature of the reactor vessel. The maximum operating temperature then determines the vessel materials that will allow an acceptable lifetime. Some designs, such as the PBMR, propose using SA-508/533 LWR steel for the reactor vessel. SA-508/533 has been used for reactor vessels in light water reactors and requires a relatively low operating temperature. The South Africans report a normal operating temperature of 350°C for the PBMR reactor vessel. This low temperature is achieved with a pressure vessel conditioning system that uses an independent coolant stream to keep the reactor pressure vessel at an acceptable temperature. This is not designed as a safety system and thus is assumed to fail during a depressurized conduction cooldown. Other designs, such as the GT-MHR, proposed operating at a higher vessel temperature of 485 °C that requires using SA-336 (9Cr-1Mo-V) steel.

Analyses were performed to determine maximum temperatures in the reactor pressure vessel of two potential VHTR designs during normal operation and during a depressurized conduction cooldown accident. The purpose of the analyses was to aid in the determination of appropriate reactor vessel materials for the VHTR. The prismatic design was based on the 600-MW (thermal) GT-MHR. The pebble-bed design was based on the 400-MW PBMR, but the power was increased to 600 MW, the desired power level of the NGNP. Calculations were performed at vessel outlet fluid temperatures of 900 and 950 °C.

The analyses of the prismatic design were performed using the RELAP5-3D computer code (RELAP5-3D Code Development Team 2005). The analyses of the pebble-bed design were performed with the PEBBED-THERMIX computer code (Gougar, et al. 2005). Because PEBBED-THERMIX has not been extensively validated, confirmatory calculations were also performed with RELAP5-3D for the pebble-bed design. These thermal-hydraulic calculations were performed with best-estimate, rather than conservative assumptions. Thus, the calculations were not intentionally biased to provide higher than expected estimates of the maximum vessel temperatures for licensing purposes.

3.1 GT-MHR VHTR Concept

The key operating parameters for the GT-MHR are listed in Table 2 along with the operating parameters for the Fort St. Vrain high-temperature gas reactor, the largest and most recent gas-cooled reactor to operate in the U.S.[3]

Table 2. Key operating parameters for the GT-MHR and the Fort St. Vrain HTGR.

Condition or Feature	Fort St. Vrain HTGR	GT-MHR
Power Output [MW(t)]	841	600
Average power density (w/cm^3)	6.3	6.5
Coolant @ Pressure	Helium	Helium
(MPa / psia)	@ 4.83 / 700	@ 7.12 / 1032
Moderator	Graphite	Graphite
Core Geometry	Cylindrical	Annular
Safety Design Philosophy	Active Safety Sys	Passive
Plant Design Life (Years)	30	60
Core outlet temperature (°C)	785	850
Core inlet temperature (°C)	406	488
Fuel – Coated Particle	HEU-Th/ ^{235}U (93% enriched)	LEU
Fuel Max Temp – Normal Operation (°C)	1260	1250
Fuel Max Temp – Emergency Conditions (°C)	NA - Active Safety System cools fuel.	1600

Figure 2 shows the GT-MHR reactor system and power conversion system within the reactor building. The plant is designed for a 60-year life with a capacity factor of at least 80%. Passive safety is achieved by designing for a core cool-down during a postulated loss-of-coolant accident that limits the peak fuel temperatures to 1600 °C. This is accomplished by conducting the decay heat radially through the core and pressure vessel and then radiating it to passive air-cooled panels in the reactor cavity building. High temperature concrete is not needed for the reactor building because of the cooling panels. There is also a non-safety shutdown cooling system (SCS) used only to remove decay heat during normal shutdowns, such as during refueling operations.

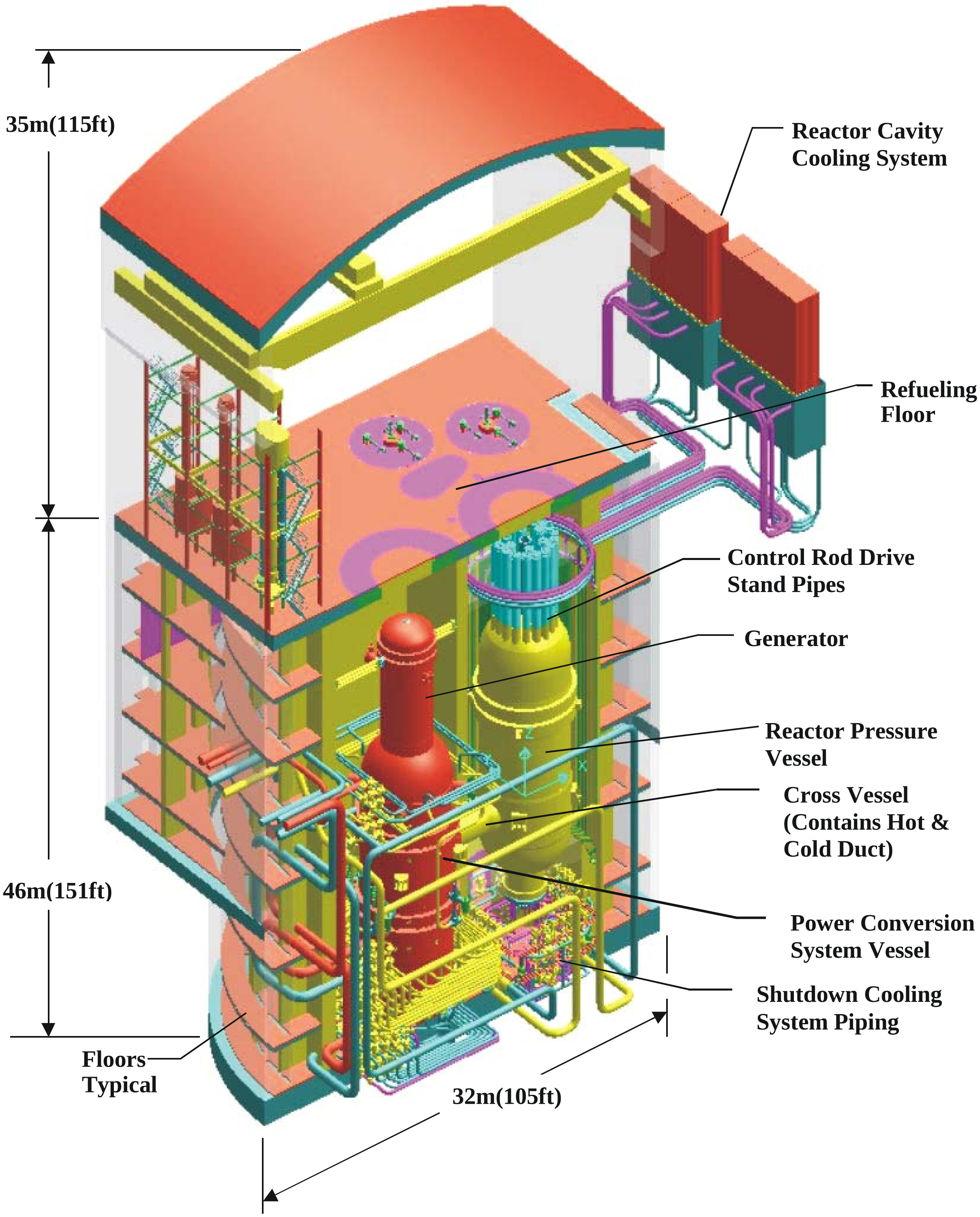


Figure 2. GT-MHR reactor building cutaway showing the arrangement of the reactor and power conversion systems.

The entire reactor confinement structure is underground. The reactor vessel and power conversion vessel are side-by-side and connected by a cross-vessel that is deliberately made as short as possible to minimize thermal expansion differences between the two large vessels. The core exit hot gas flows in the central channel (duct) located along the centerline of the cross-vessel to the turbine inlet. The reactor inlet gas flows in an annular channel between the center hot duct and the cross vessel to the reactor inlet. The power conversion vessel is set somewhat lower than the reactor pressure vessel to prevent natural circulation of hot gases into the power conversion vessel during a loss-of-flow accident (station blackout).

Figure 3 is a cutaway view of the reactor vessel showing more details of the inside of the core. The core consists of graphite blocks with an annular-fueled region surrounded by reflector elements. The fuel is TRISO coated fuel particles embedded in graphite compacts and placed in graphite prismatic blocks. The center of the core is a non-fueled graphite reflector. Normal operating maximum fuel temperatures do not exceed 1250 °C. The reflectors mitigate the high-energy fluxes, and boron pins placed in the outer reaches of the reflectors reduce thermal neutron fluxes on the metallic internals structures and reactor vessel.

From the cross-vessel, the reactor helium inlet coolant (~500 °C and ~7 MPa) flows upward in the annulus between the reactor pressure vessel and the metallic core barrel surrounding the side reflector. Hence it is a major determinant of the vessel operating temperature. The coolant then enters the upper plenum region volume, which contains the lower parts of the control rod housings. The reactor pressure vessel upper head is protected by fibrous “Kaowool” insulation blankets supported by high-temperature metallic plates. The insulation protects the head from hot plumes that could occur during a pressurized loss-of-forced-convection (P-LOFC) accident.

The inlet flow then passes down through the core’s upper support elements, which are made of carbon-carbon composite material that must also withstand the hot gases in a long-term P-LOFC. The coolant then flows primarily into the coolant channel holes in the fuel elements. Some of the flow bypasses these channels, passing through the gaps between the fuel elements and reflector blocks. Thus the temperature rise of the coolant in the various flow paths through the core varies over a wide range. The coolant in the fuel element channels with the highest local power peaking is quite hot whereas the coolant in the relatively unheated gaps adjacent to the cooler reflectors remains near the inlet temperature. Since the average temperature rise through the core is ~350 °C, good mixing of the outlet coolant is needed to avoid excessive thermal stresses in the downstream components resulting from large temperature gradients and fluctuations, and to assure that the gas entering the turbine has a uniform mixed mean temperature of 850 °C. Various design options are available to mitigate the effects of these perturbations.

The reactor vessel operates at a maximum through-wall average temperature of 440°C during normal operation and reaches about 550 °C during a conduction cooldown event. The core’s fuel elements and graphite reflectors, plus the control rods and housings and the shutdown ball channels are all non-metal, capable of withstanding the prescribed maximum core temperatures (~1600°C) or higher in the design-limiting loss-of-coolant accident.

The “Hot Duct” assembly is composed of a structural duct separating the core entrance gas from the core exit gas, and an insulation assembly on the inside surface of the structural duct to protect it from direct contact with the 850 °C core exit gas. The structural duct is subjected to the core pressure drop as an external pressure load on a cylinder. The insulation assemblies are designed to be remotely removed and replaced if needed during the 60-year plant life.

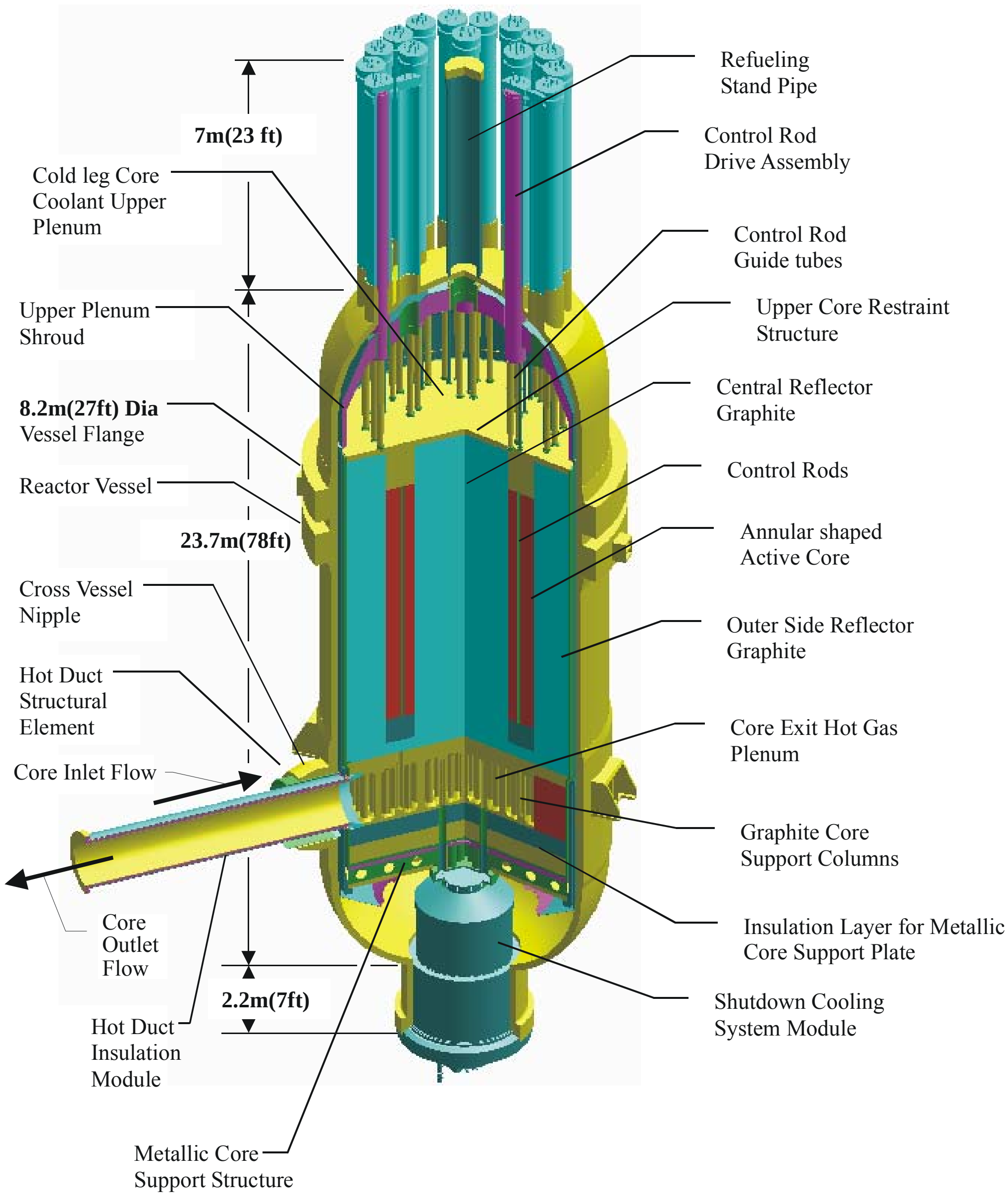


Figure 3. GT-MHR reactor system cutaway showing the metallic internals structures, core, control rod guide tubes, and shutdown cooling system.

Between the core exit plenum and the bottom metallic core support plate is an insulation layer ~1.2 meters thick. It is composed of a meter of nuclear graphite and 200 mm of carbon-carbon composite blocks. This combination of materials and thickness drops the temperature from 850 °C (core outlet temp) to ~510 °C on the top of the core support plate and ~490 °C on the bottom.

Below the bottom metallic core support is the SCS module shown in Figure 4, used to remove decay heat from the core during normal shutdowns. It is not a safety system. It contains a water-cooled heat exchanger and a motor driven circulator. It can be removed and replaced for maintenance. The high-temperature thermal insulation in the upper gas collector plenum will need to be upgraded to withstand the 1000 °C core outlet temperature of the NGNP.

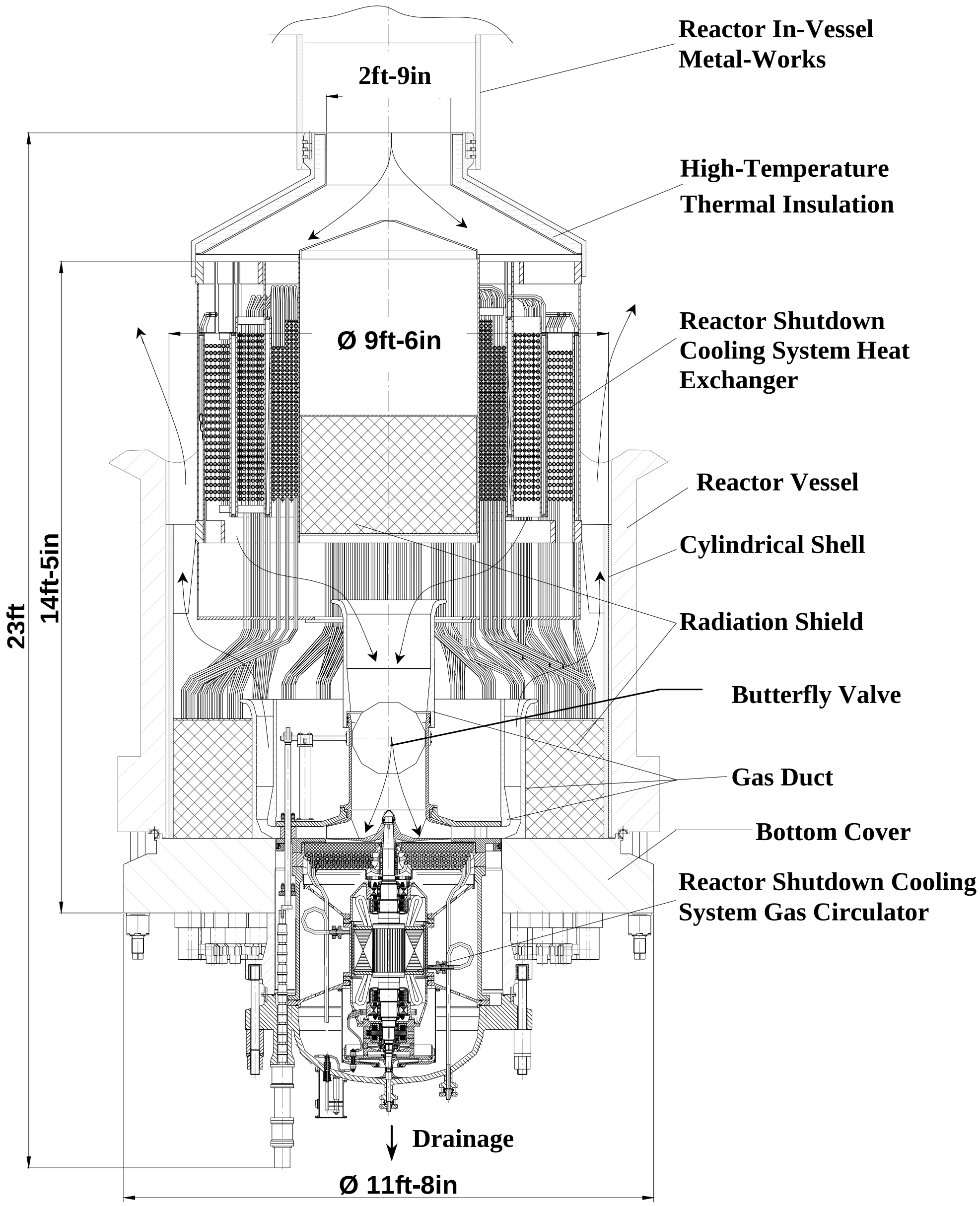


Figure 4. Cross-section of the GT-MHR shutdown cooling system.

The power conversion unit is shown in Figure 5. It is a direct (Brayton) cycle vertical single shaft axial flow gas turbine. The compressor is a two-stage compressor with a pre-cooler and an intercooler. Hot gas from the reactor enters the turbine from the hot duct. The turbine inlet volute is designed as an insulated structure like the hot duct. High temperatures are experienced by the turbine's first few stages, the turbine inlet structure, and the recuperators. All the other power conversion unit structures operate at relatively low temperatures. Gas exiting the turbine is passed through the recuperators to raise the core inlet coolant temperature to ~500°C. The generator is contained within the primary helium coolant.

In the power conversion unit, the turbine blades and disks operate at temperatures that are similar to those of many modern air turbines that have no or only limited blade cooling. Some modern combustion turbines are operated with considerably higher inlet gas temperatures, but provisions are made to provide cooling to the blades and disks to maintain their material temperatures within acceptable ranges. The GT-MHR power conversion unit turbine has the potential benefit of being subjected to relatively clean, pure helium as opposed to the air and combustion products that a combustion turbine has to withstand. Therefore, the GT-MHR turbine lifetime could turn out to be longer than the average lifetime of 6 years for modern combustion turbines. Some components (e.g., the recuperators) must also withstand very rapid and severe temperature transients when the bypass valves operate to prevent generator runaway in a sudden loss of electrical load event.

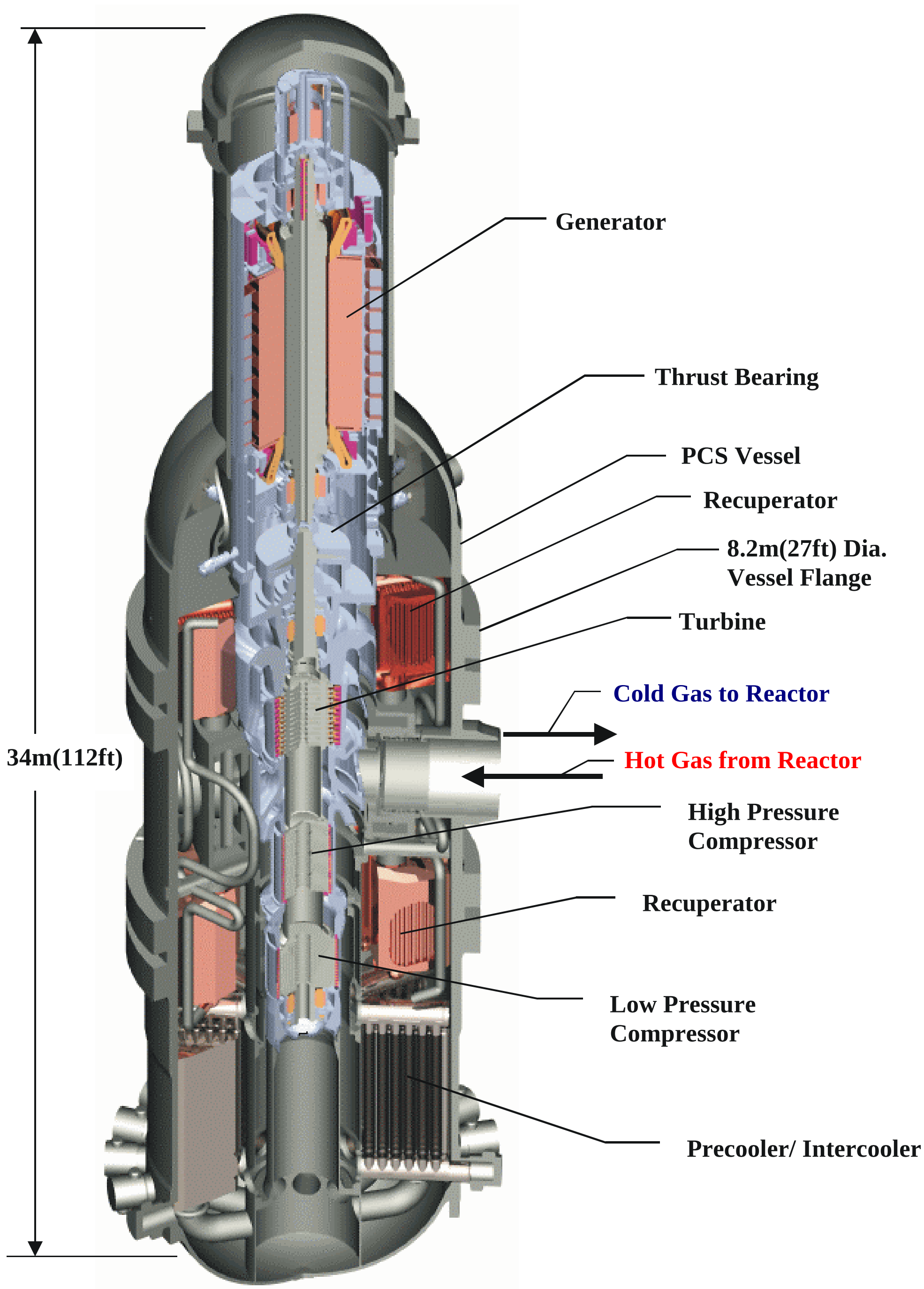


Figure 5. GT-MHR power conversion unit cutaway showing the turbomachinery: turbine, compressors, recuperators, intercooler/precooler, and generator.

3.2 The Framatome–ANP High Temperature Reactor Concept[12,13]

The Framatome-ANP high temperature reactor concept is presently in the preconceptual design phase and several technical options are assessed to choose the optimum technical/economic solutions consistent with low risk. The concept is based in part on the GT-MHR concept. The main plant parameters are listed in Table 3, while Figure 6 presents the plant principle.

Table 3. Approximate Plant Design Parameters

Design Parameters	Design Values
Reactor Power	600 MWt
Reactor Outlet Temperature	850° C
Reactor Inlet Temperature	355° C
Primary Coolant Flow Rate	240 kg/s
Primary Coolant Pressure	5.5 MPa
Reactor Vessel Material	9 Cr – 1 Mo or SA 508
Core Configuration	102 Columns, 10 blocks high.
Fuel Particle Type	SiC Coating UCO or UO2 kernel
Operating Max Fuel Temp. Guideline	1300° C
Accident Peak Fuel Temp. Guideline	1600° C
IHX Design	Compact
IHX Nominal Heat Load	608 MWt
IHX Effectiveness	90 %
IHX Primary Tin	850° C
IHX Tout	350° C
Secondary Fluid	Nitrogen (80%)/Helium (20%) Mixture
IHX Secondary Tout	800° C
IHX Secondary Tin	300° C
Secondary Flow Rate	614 kg/s
Secondary Coolant Pressure	5.5 MPa

The nuclear plant is based on the modular concept whereby the core is contained in a metallic pressure vessel designed to be able to radiate the residual heat in case of a major accident. This passive means by itself is sufficient to limit the fuel kernel temperature to less than 1600°C to ensure fuel particle integrity.

In order to optimize the power level of the core while meeting the previous passive safety requirement, the core has an annular configuration. In order to minimize core pressure drop and to simplify future spent fuel recovery either for storage reasons or to reprocess it if justified, the hexagonal block type core concept has been retained. Each block is loaded with compacts, composed of graphite and fuel microparticles based on the TRISO design. This concept has been implemented and proven in several reactors in the past. Core zoning will be used in order to optimize the fuel cycle.

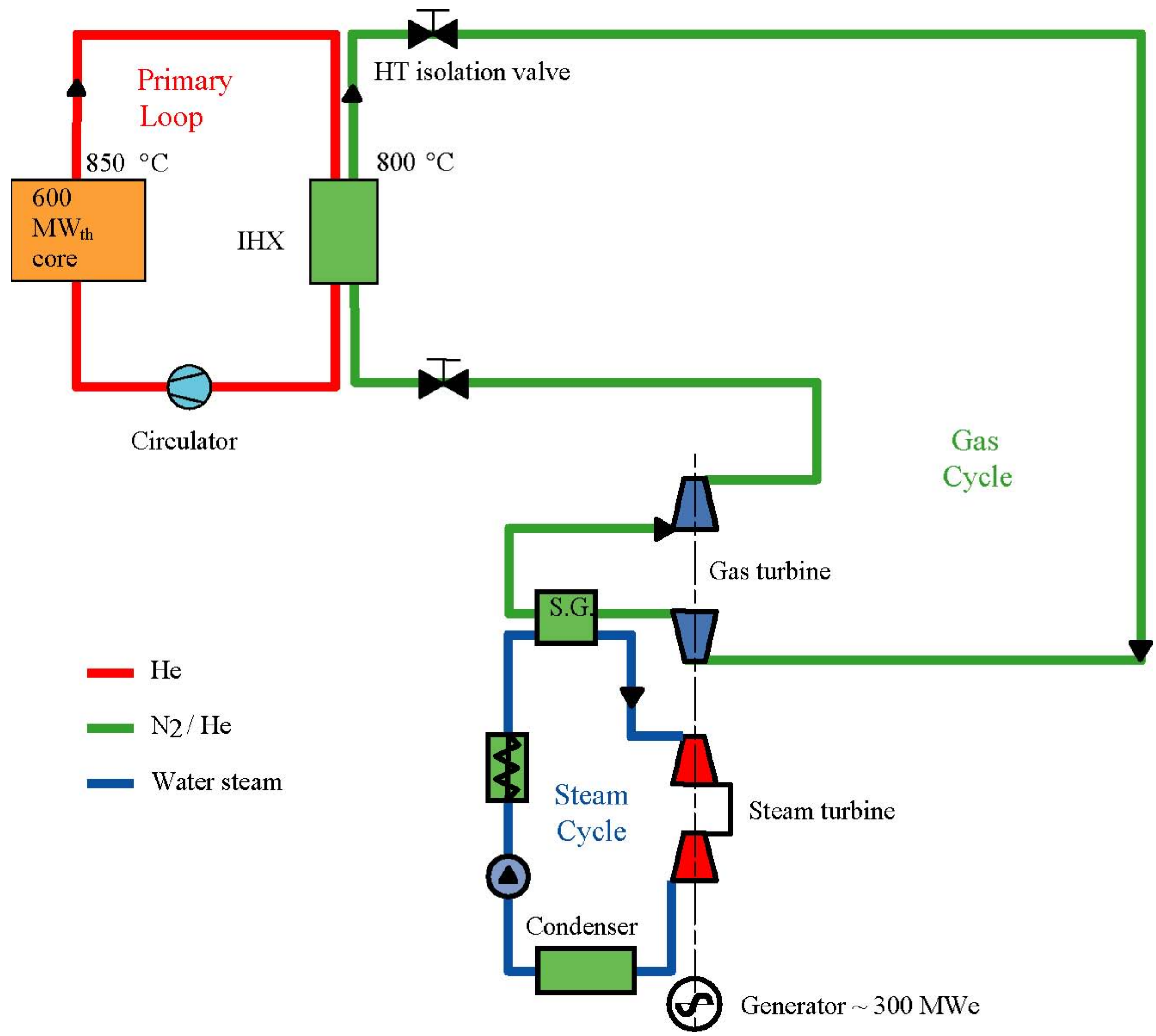


Figure 6. Concept Plant Layout

As can be seen in Figure 6, the primary loop which contains the nuclear components is reduced to a minimum number of components. This has a favorable impact on component costs and also reduces the probability of maintenance in a radiological environment. It should also be pointed out that the primary loop pressure is limited to 5.5 MPa which is substantially less than the pressure required by direct cycle systems ranging from 7 to 9 MPa..

Framatome-ANP has set a goal to achieve a high performance and efficiency while minimizing the cost and delay of new development. An original indirect combined-cycle concept has been developed by Framatome-ANP. It shows a potential for very high efficiency, 47 % or more, while minimizing technological risk.

A detailed assessment was made with a power generating equipment manufacturer, MHI, which confirmed the initial Framatome-ANP evaluation and independently by EDF, the French electric utility.

The salient features of that design can be summarized as follows:

1. A gas Brayton cycle is used to recover energy and convert it into electricity for the temperature range of 800°C to 600°C. Generally speaking, the gas-cycle is efficient at high temperature but since it does not include a condensation step, it is not as efficient in the lower temperature range. For example, a substantial amount of heat at temperatures between 120°C and ambient is lost in the precooler and intercoolers of direct Brayton cycles. In the present combined cycle, all of the gas loop energy is recovered in the steam generator and passed to the steam cycle which converts it efficiently down to the low temperatures of steam condensation at ambient temperature. It must be noted that the gas loop net power production is about 80 MWe while the steam cycle produces about 200 MWe. This results in a steam topping cycle instead of the conventional combined cycle using a bottoming steam cycle.

2. In order to use conventional gas turbine technology and take advantage of the closed gas loop, the gas which best fits our purpose was selected. Therefore, nitrogen, which has similar properties to air, was chosen as the gas vector. In order to improve the heat exchange characteristics of the gas for the IHX design, a small amount of helium can be added to the nitrogen. It does not affect the turbomachine design while decreasing heat surface transfer area for the IHX. It can be shown that the gas loop efficiency is independent of the absolute pressure; therefore the normal operation pressure is chosen so that the IHX has the same pressure on both sides. This limits the load usage factor on this equipment.

3. The fact that the compressor inlet gas is not cold is taken into account in the compressor design, but this is not unusual since, in a conventional combined cycle, air is compressed to about a 20/1 compression ratio and some intermediate compressor stages actually compress air with similar inlet temperature. The gas loop efficiency is affected by this but it is minor in terms of overall efficiency as this is recovered in the generated steam and the steam cycle. The turbomachines are actually a simpler adaptation of conventional air combined cycle designs. They are simpler because our compression-expansion ratio is about 2 compared to about 20. This means our gas turbine has only 3 stages and our compressor 8 or 9 stages. Fewer stages mean less potential for losses and inefficiencies. The maximum inlet temperature in the turbine is 800°C which means conventional materials are used and there is no need for blade or disk cooling. Altogether, the turbomachine design is simpler than conventional designs, which implies at least an equivalent reliability, a lower cost and easier maintenance.

4. Finally, it is important to keep in mind that inefficiencies affecting the gas loop are of a second order importance because this loop contributes to a minor amount of the global production on one hand, and because any inefficiency which results in dissipated heat is actually recovered in the steam cycle on the other hand. From that point of view, the combined cycle is more accommodating than a direct cycle where any operational inefficiency due to design or wear of the turbo machines, or the recuperator can dramatically impact the overall plant efficiency.

5. A modern high efficiency steam cycle operates between 550°C and ambient temperature as a bottoming cycle. It uses conventional superheated steam at 12 to 16 MPa to drive the steam turbine. The equipment is well developed, reliable and economical.

6. The entire power conversion system is outside of the nuclear loop, in a noncontaminated area, and is very close in principle to equipment that most utilities are very familiar with in the conventional natural gas industry. This concept makes maintenance particularly easy without the need for special tooling or processes.

The direct-cycle concept operating parameters are optimized for a direct-cycle Brayton turbine. During cogeneration, operation parameters can change dramatically. The direct-cycle concept has little flexibility to adapt to cogeneration because when operating parameters change, efficiency drops and operation may become difficult. The only noteworthy exception is when low grade heat is extracted from the precooler and intercooler for applications such as desalination or district heating.

Conversely, the indirect combined-cycle is quite flexible to accommodate the simultaneous generation of electricity and industrial heat at any level between 800°C and ambient. This specific characteristic of the indirect-cycle HTR opens new markets to nuclear power.

If the required industrial heat production is below 550°C, it is extracted from the steam loop. Since such systems already exist on conventional plants, there is minimal development. Depending on the industrial heat characteristics, temperature, pressure and steam quality, the extraction is implemented at different locations in the steam plant. Of course such extraction impacts the electricity production of the steam plant as the process heat is removed from the normal steam loop, usually through a heat exchanger. It is important to note that these cases below 550°C cover the majority of the potential needs and that the process heat is separated from the nuclear loop by at least 3 heat exchangers in series, thereby providing an excellent isolation between the nuclear heat source and the process heat. The potential for contamination of the process heat is very low, which may be very important for some of the processes. The cogeneration process implies a modification of the steam loop compared to an electric-only plant, but such modifications are well known and inexpensive. On the other hand, the nuclear heat source is not modified. The goal is to keep the nuclear heat source as standard as possible with the same operating parameters in order to avoid specific lengthy and expensive component qualification which would be dedicated to cogeneration operation.

If the required process heat is between 800°C and 550°C, it must be extracted from the gas loop. These applications are not typical and each will have to be studied specifically.

One such special application would be the testing of a direct Brayton cycle, the heat source of which would be the secondary side of the IHX. This would allow the operating experience and qualification of a direct Brayton cycle in a noncontaminated environment. This would allow easy access for the first-of-a-kind unit, since frequent maintenance and adjustments would be expected. If the Brayton direct cycle proves efficient and operating experience shows very low contamination, then this could justify the direct Brayton cycle for future units.

The Intermediate Heat Exchanger (IHX) is the only truly novel component since no similar equipment exists to meet the concept specification.

There are examples of similar equipments but at lower heat transfer rates. Examples include the HTTR heat exchanger and the 10 MWth IHX developed in Germany for the PNP program. They are both tube heat exchanger units which have operated up to 950°C.

Tube heat exchangers are one option for the application, with the German experience being a good initial reference. Considering the 600 MWth heat transfer requirement, a modern version more compact than the original design, is being studied. It is feasible based on this previous experience.

In parallel with the tube design and taking advantage of the significant improvement in the plate-type heat exchangers during the past 20 years, several plate-type designs which have the potential to be much more compact than the tube design are being investigated. The present program includes investigation of the printed circuit-type design and several different plate fin designs. Mock-ups of each of these will be manufactured and tested in a high temperature loop presently under development with the CEA (French Atomic Energy Commission).

The requirements on the IHX have similar design challenges with the recuperators used by direct-cycle designs. While a higher temperature operation is needed, which limits our choice of IHX materials, the normal operation is pressure balanced whereas a recuperator has a normal operating pressure difference of over 4 MPa. This pressure difference is significant with respect to the recuperator usage factor. In addition, the recuperator is subject to violent transients during normal operation. Overall the IHX and the recuperator share a number of design challenges.

Finally, developing an efficient and reliable IHX is a requirement for any HTR design that intends to deliver high temperature heat to a hydrogen plant; therefore all concepts are equally concerned by this challenge.

The strategy for the reactor pressure vessel is designed to minimize risk and uncertainty in the design process and maximizes operational margin. Grade 91 steel is preferred because of its superior high temperature properties compared to SA 508/533 LWR steel. Qualification of Grade 91 steel for this application is not judged to be substantially more difficult than would be necessary for LWR steel and the resulting margin during a transient situation would be greater. Grade 91 fabrication concerns are actives being addressed and a welding development is underway.

The Commercial Design Concept described previously stresses the reliance on well known technologies and the low technological risk involved, with the exception of the IHX.

An advanced design capable of a higher temperature will be required for efficient hydrogen production. The overall concept does not change, but higher temperatures require new and more expensive materials with significant R&D and testing. This is because when some candidate materials are identified to reach temperatures in the 950 to 1000°C range, they still must be tested industrially over their expected lifetime under reactor conditions. Therefore, in order to prepare for this, a work program has been launched with CEA to generate within the next 10 to 15 years the data allowing the choice of such materials and their forming and assembly processes.

Whatever positive results come out of this development, they will be directly applicable to all commercial electric plant concepts to potentially improve efficiency. It still needs to be seen, however, if the extra cost of the newly developed high temperature materials will justify their use.

For the short term, conventional well-known materials qualified for our environment will be used, which is why the Commercial Design is limited to 850°C.

A few challenges remain, such as developing a life cycle cost model that demonstrates the economic competitiveness of the HTR concept compared to competing technologies.. The HTR brings several positive new features from the safety and efficiency viewpoints, but at the cost of a low core power density up to 15 times lower than light water reactors. As a consequence, it must overcome this inherent design feature with sufficient simplifications.

The licensing of a passive industrial concept has not been demonstrated. In order to meet the economic challenge, the licensing process should recognize the HTR specific features. This is not certain at this time. Hopefully, specific light water licensing legacy will not affect HTR licensing but the actual demonstration is still ahead.

Technical challenges regarding the plant design still exist in terms of industrial fuel qualification with a high degree of reliability since the whole HTR safety is based on its unique fuel characteristics.

Finally, long term material behavior in the helium environment require specific qualification, including helium impurities and nitriding environment of the gas loop at high temperatures.

In conclusion, the commercial Framatome-ANP HTR concept is based on a low risk design using mostly conventional and well-proven technologies with the exception of the IHX for which a strong development program has been launched exploring several technologies.

The original combined cycle power conversion temperature applications such as thermochemical system proposed by Framatome-ANP leads to a hydrogen generation at least on par with the best Brayton Cycle designs but without the requirement for newly developed turbomachines and it is amenable to maintain in a noncontaminated environment.

3.3 PBMR VHTR Concept Design[14,15,16,17,18,19]

The PBMR is a high temperature, helium-gas-cooled, graphite-moderated thermal reactor designed for the safe, highly efficient, production of electricity and/or hydrogen. The reactor is designed to be passively safe during all postulated design-basis accidents, including loss of flow and loss of coolant. The helium coolant is single-phase, non-condensable, chemically inert (except for small amounts of impurity gas content discussed in other areas of this report), and has no reactivity effects. The graphite core provides high heat capacity, slow thermal response, and structural stability at very high temperatures. The reactor has a negative temperature coefficient of reactivity, which inherently shuts down the core when the fuel temperature exceeds normal operating temperatures. The limited total core power allows the reactor to be designed for passive heat conduction from the core, thermal radiation and convection from the vessel and conduction to the confinement structure keeping temperatures low enough to prevent core or fuel damage.

This reactor is being developed in South Africa by PBMR (Pty) Ltd. through a world-wide development effort. The program includes testing of mechanical systems and components, a comprehensive fuel development effort and a testing and verification program to support the licensing process. A full-sized demonstration PBMR reactor will be built at the Koeberg nuclear reactor site (owned by Eskom, the South African national utility) near Capetown, South Africa.

Electrical generation using the PBMR is economically favorable where smaller market conditions prevail. It can also be configured as an industrial process heat source for other activities such as hydrogen production, enhanced oil recovery or coal to liquid fuel conversion. There is currently no other CO_2-free heat source, which in the near- term, can economically provide large amounts of process heat in the 600-900°C temperature range.

The PBMR utilizes carbon-based spherical fuel elements, called pebbles, which are approximately the size of a tennis ball. The design of these pebbles is based on the German High Temperature Reactor (HTR) programs and is characterized as refractory, triple-coated (TRISO) particle fuel design. TRISO fuel consists of a small UO_2 kernel with a porous carbon buffer layer, a dense pyrolytic carbon layer, a very hard silicon carbide layer, and another dense pyrolytic carbon layer. The resulting coated particles are mixed in a graphite matrix, then pressed and sintered into a sphere of 6 centimeters diameter. The heart of the safety case for PBMR is high quality fuel with extremely low particle failure rates. Previous test results from the German HTR programs have shown that as long as the maximum fuel temperature remains below about 1600 °C, fission products will be contained within the fuel and its TRISO coatings. The pebbles are located in an annular cavity in the reactor vessel, between a cylindrical inner graphite reflector and an annular outer graphite reflector. The pebbles are dropped from several points above the core annulus, and a small mound develops below each drop point. Pebbles proceed vertically downward until they are removed at the bottom. On removal they are checked for damage and burnup, and if they are intact and not burned past the burnup limit, they are circulated to the input queue again. Otherwise, they are replaced with fresh pebbles. This on-line refueling feature makes refueling shutdowns unnecessary, and it also allows the reactor to operate with almost no excess reactivity, which confers advantages in safety, economy, and resistance to nuclear weapons proliferation. This fuel design is shown in Figure 7.

For electrical generation, a single loop, direct Brayton cycle is used. This power generation design has the advantage of possible increased cycle efficiencies and elimination of problems due to water induced system component corrosion that could result from the use of a steam cycle. The issues associated with fuel degradation due to interaction with water in-leakage are eliminated. The extensive materials development, ASME Code issues and system reliability issues associated with the use of an intermediate heat exchanger (IHX) are also avoided.

In the Brayton cycle, reactor coolant gas at 9 MPa transfers heat directly to the power conversion system which consists of a gas turbine with an associated generator and gas handling equipment. Power is adjusted by regulating the mass flow rate of gas inside the primary circuit. This is achieved by a combination of compressor bypass and system pressure changes. Increasing the pressure results in an increase in mass flow rate, which results in an increase in the power removed from the core. Power reduction is achieved by removing gas from the circuit. A Helium Inventory Control System (HICS) is used to provide an increase or decrease in system pressure. The main components of the reactor system and the power conversion unit are shown in Figure 8.

The operating parameters of the PBMR are shown in Table 4.

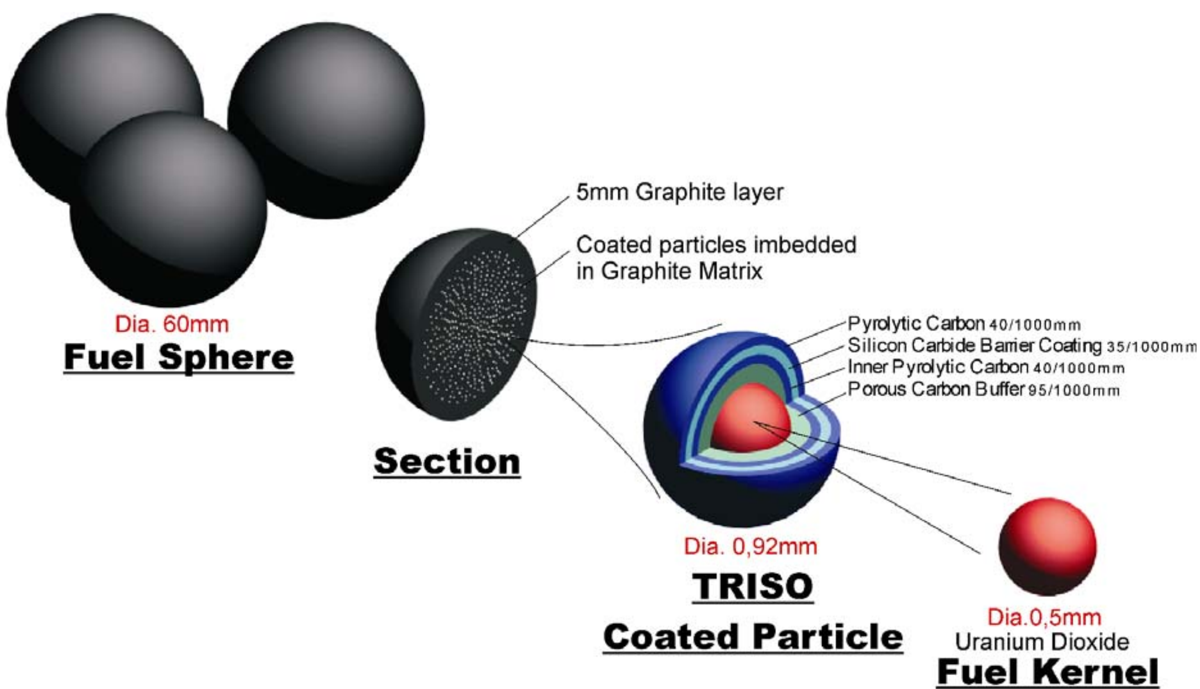
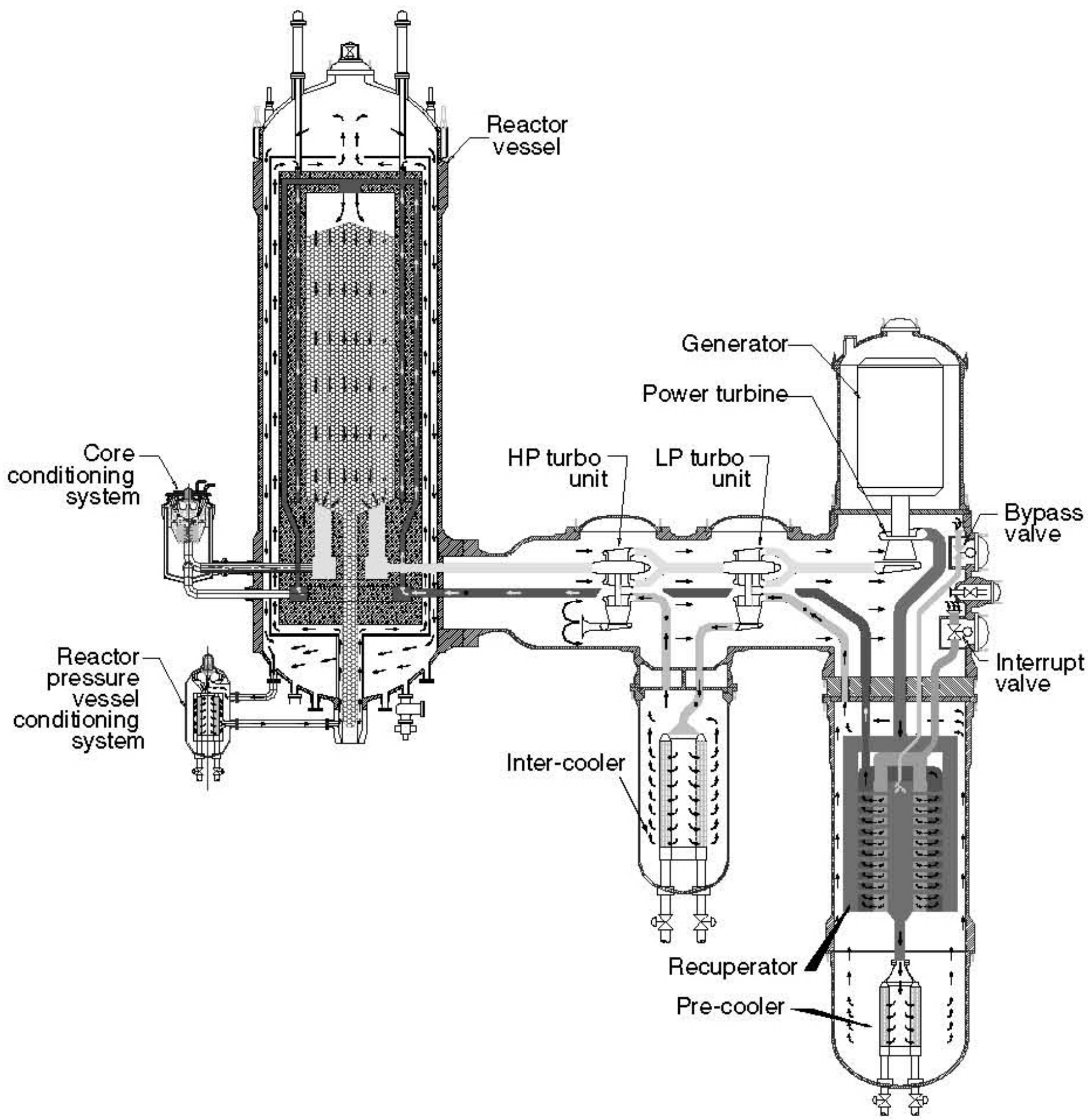


Figure 7. PBMR fuel sphere.



Source: Courtesy of Eskom

Figure 8. PBMR reactor unit and power conversion system.

Table 4. PBMR nominal full power operating parameters.

PBMR Design Parameters	PBMR Design Values
Reactor Power	400 MWt
Core Inlet/Outlet Temperatures	403 °C/900 °C
Core Inlet/Outlet Pressures	8.9/8.6 MPa
Coolant	He
Helium Mass Flow Rate	193 kg/s
Vessel Pressure	9.0 MPa
Reactor Vessel Average Wall Temperature	350 ºC (at 400 MWt)
Reactor Vessel Outside Diameter	6.38 m
Reactor Vessel Wall thickness (nominal)	18 cm
Core Diameter (outer/inner)	3.7 m / 2.0 m
Core Height (nominal)	11 m
Fuel Annulus	85 cm
Mean Core Power Density	4.78 W/cm3
Peak Core Power Density	10.99 W/cm3
Reactivity, Temperature Response	Negative temperature coefficient of reactivity
Mean Fuel Temperature	~800 °C
Peak Fuel Operating Temperature	1057 °C
Refueling	Continuous
Heat Transfer to Power Conversion	Direct Cycle
Net Electrical Output	165 MWe
Net Plant Efficiency	>41%

The PBMR design concept is that a basic reactor unit called a module of 400 MWt thermal capacity (165 MWe) can be configured in groupings of up to eight modules per plant. The building design for a single PBMR module is shown in Figure 9. It consists of a reinforced concrete confinement structure, called the citadel, which houses the power conversion unit, which is located inside a more conventional concrete building that houses all of the auxiliary equipment. The function of the citadel is as a confinement structure to protect the nuclear components of the power conversion unit from external missiles and to retain the vast majority of fission products that might be released in the event of a reactor accident.

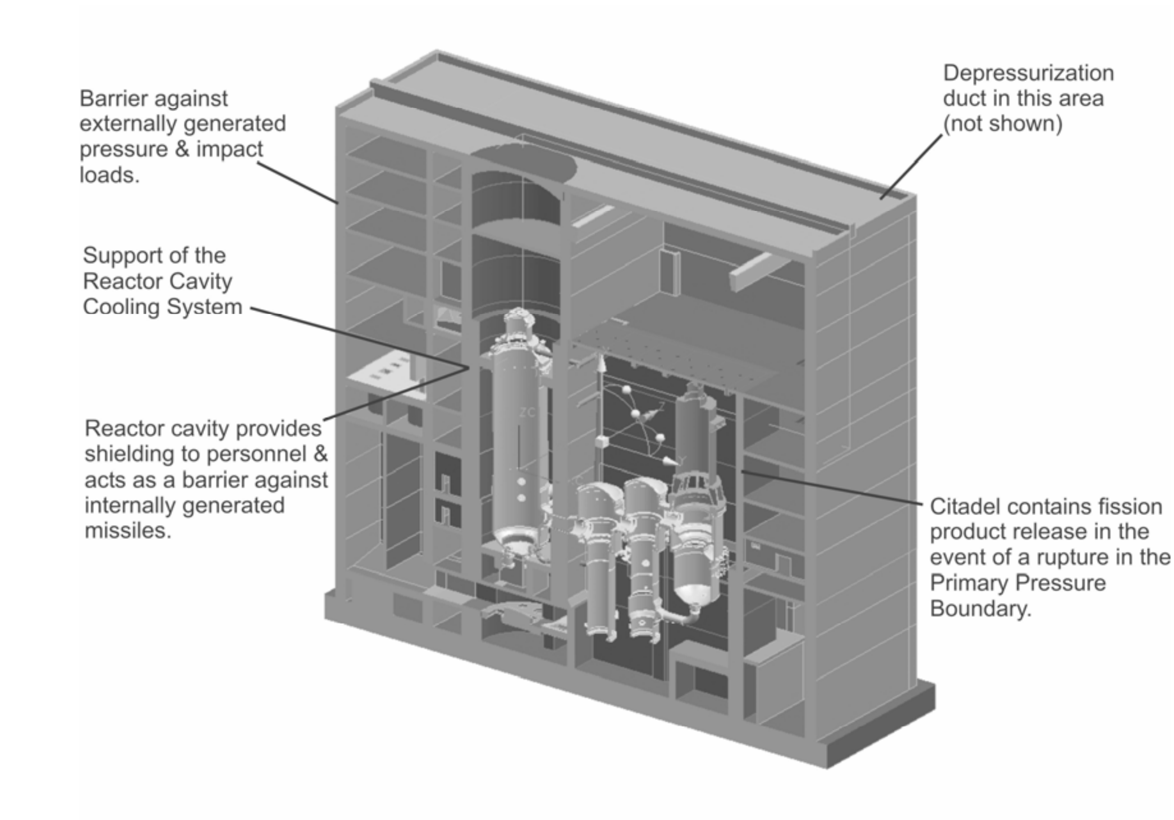


Figure 9. PBMR single module building.

3.3.1 PBMR Primary Materials of Construction

The present design of the PBMR allows the use of readily available materials that have ASME design allowables. These materials will not need any additional development or data base generation for use at the NGNP system design conditions. Table 5 lists the material choices for the major components.

Table 5. PBMR material choices for major components (40 year design life)

Component	Normal Operating Temperature (°C)	Maximum Temperature for Design Basis Events (°C)	Maximum Fast Fluence (n.cm-2) [E > 1 MeV]	Candidate Materials
Reactor Pressure Vessel	350	439	1 x 1018	SA 508, Gr. 3, (forgings) SA 533, Type B (plate)
Core Barrel (Core Support)	400	557	3 x 1018	Type 316 SS
Reactivity Control Rods	700	850	5 x 1021	Incoloy 800H
Core Outlet Pipe Liners	900	-	Not significant	Incoloy 800H

3.3.2 Reactor Pressure Vessel (RPV)

The RPV of the PBMR is a vessel with the nominal dimensions indicated in Figure 10. The vessel design consists of a welded cylindrical shell welded to the bottom head. The top head, containing numerous penetrations for fuel handling and reactor control systems, will be bolted to the cylindrical section. The RPV material will primarily be ring and nozzle forgings of SA508 Grade 3 Class 1 and plate material for the cylindrical conforming to SA533, Type B. These materials have a common chemistry.

The RPV design configuration is such that its normal operating temperature range is from 300-350°C. The selection of SA508 Grade 3 Class 1 and SA533 Type B materials at this temperature provides the following benefits:

1. There is manufacturing experience in forging large diameter, thick ring sections thus ensuring predictable through-thickness material properties.

2. There is welding experience with these materials

3. The SA508 Grade 3 and SA533, Type B materials are ASME qualified material for nuclear pressure vessel design.

4. ASME design rules, in the form of a nuclear code case, for limited use of these materials in the temperature range 371°C to 538°C are available.

5. There is an extensive irradiation response database at the normal operating temperatures incorporated in the NRC licensing guidelines (NRC Regulatory Guide 1.99) and other international standards (ASTM E 900).

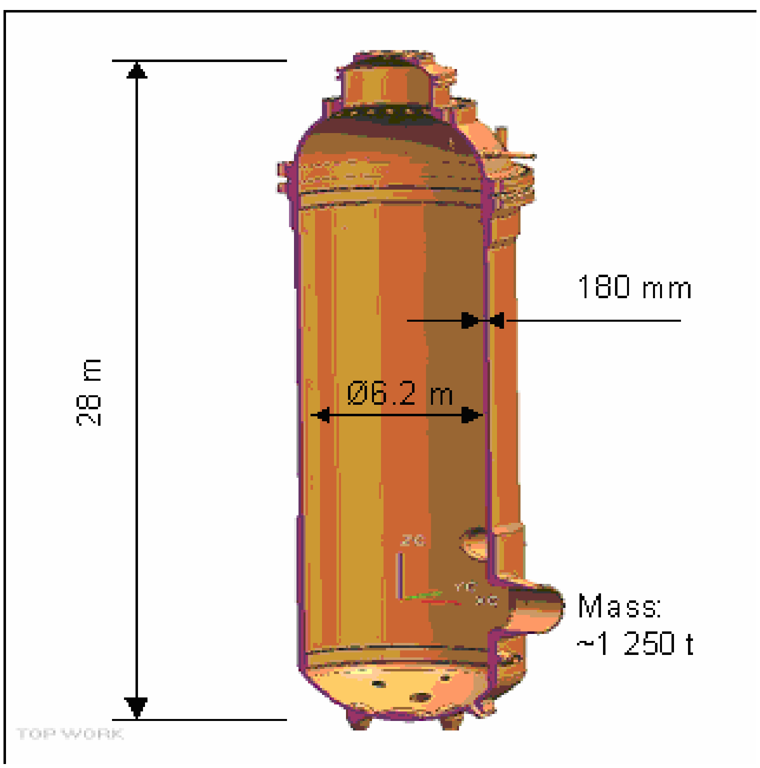


Figure 10. Reactor Pressure Vessel

3.3.3 Core Barrel

The core barrel is a welded internal metallic vessel that supports the graphitic reflector and the fuel spheres. The core barrel is directly supported by the RPV. It further separates the gas flow through the core from the gas flow on the inside of the RPV. It is constructed from SA240 Type 316H plate material. The H-grade is specified to provide enhanced creep resistance when the core barrel is exposed to temperatures above 427°C. There is also extensive industrial experience and inclusion in ASME Code Case N-201 for core support structures for Type 316H material. The core barrel is shown in Figure 11.

The core barrel also supports the core structures, and the design rules within ASME III, Division 1-Subsection NG (Core Support Structures) are appropriate. Due to the temperatures experienced during upset conditions, the design has to utilize the rules provided in ASME Code Case N-201-4, which supplements the design rules of ASME III, Division 1, Subsection NG. This code case provides design rules for operation at temperatures above the limits specified in Subsection NG, and restricts the permissible materials for structural parts to four candidates (i.e., 2¼Cr1Mo, Types 304 and 316 Stainless Steel and Incoloy 800H).

As one of the code allowable materials, Type 316 Stainless Steel provides the following advantages:

1. Extensive qualification and industrial use.
2. Significant resistance to irradiation effects due to fast neutron fluence (E > 1 MeV).
3. Relatively cost-effective in comparison to the other available candidate materials.

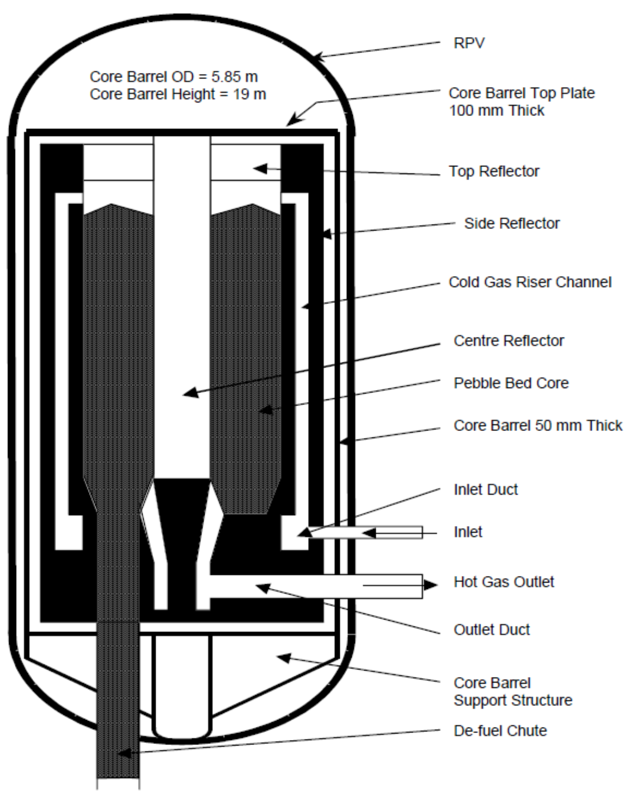


Figure 11 Core Barrel (CB) & Support Structure

3.3.4 Reactivity Control Rods

The control rods see the harshest conditions of all of the PBMR metallic materials with respect to high temperature and neutron irradiation. The control rods are part of the Reactivity Control System (RCS). The RCS consists of control rods and shutdown rods. The design aims to limit the stresses in the RCS cylinders to a minimum and the RCS is designed to be replaceable. The life of the RCS is limited by the creep strength of the material and the embrittlement due to temperature and fast neutron exposure.

Incoloy 800H is seen as the most suitable metallic alloy for the control rods for the following reasons:

1. Adequate high-temperature strength at the normal operating temperature of 700°C

2. Creep resistance sufficiently qualified for long-term operation at 700°C.

3. Limited operation at 850°C under abnormal events is allowed as per available data.

4. Irradiation response has been characterized to high levels of fast fluence.

5. Extensive qualification of Incoloy 800H control rods in previous German HTR programs.

3.3.5 Core Outlet Pipe Liner

The arrangement of the RPV core outlet pipe liner is shown in Figure 12. The liner forms an integral part of the insulation held between the liner and the outer pressure boundary material. The insulation in the core outlet piping is a necessary component of the insulation system required for keeping the outer pressure boundary (ferritic steel) temperatures within operational limits.

The inner liner material of the core outlet pipe is specified as Incoloy 800H. The liner has virtually no load-bearing function and the use of Incolloy 800H is dictated by its oxidation resistance to the impure helium, and adequate high temperature strength. As a liner material, Incoloy 800H has the following advantages:

1. Adequate high-temperature strength and creep resistance

2. Extensive fabrication experience in large diameter pipe sections

3. Extensively tested as liner material for qualification of the insulated “hot pipe” design in the German HTR program up to 950°C.

A limited distribution version of Section 3.3 that provides more details on several aspects of the PBMR design is given in Appendix C in selected NGNP Materials Program Plan reports.

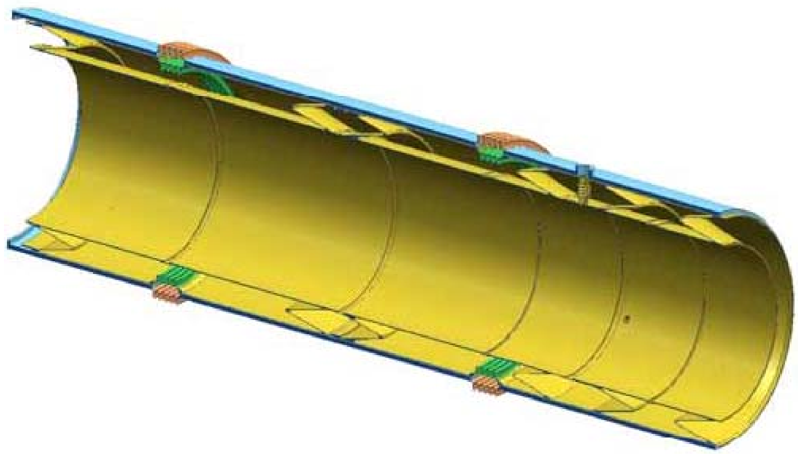


Figure 12 Core Outlet Pipe Liner

3.4 Results of INL Reactor Pressure Vessel Analysis Associated with the Prismatic and Pebble Bed Designs[20]

The analysis of the prismatic design was performed using the RELAP5-3D computer code[21]. The analysis of the pebble-bed design was performed with the PEBBED-THERMIX computer code[22]. Because PEBBED-THERMIX has not been extensively validated, confirmatory calculations were also performed with RELAP5-3D for the pebble-bed design. These thermal-hydraulic calculations were performed with best-estimate, rather than conservative assumptions. Thus, the calculations were not intentionally biased to provide higher than expected estimates of the maximum vessel temperatures for licensing purposes.

Preliminary temperature limits for SA508 and SA336 steels are based on the current boiler and pressure vessel code from the American Society of Mechanical Engineers (ASME)[23,24] are presented in Table 6. These limits are considered preliminary because they do not account for reductions due to welds and operating history. The limits presented in Table 6 are compared with the calculated results obtained from this analysis to determine initial estimates of the suitability of the different materials for the proposed designs. The limits for normal operation are based on the values given in the second column of Table 6 for SA508 and the third column for SA336. The limits for the depressurized conduction cooldown accident are given in the fifth column. Although the code allows operation of SA336 up to 590°C for $3x10^5$ h, it is expected that the vessel temperature will eventually be limited to less than 425°C during normal operation to avoid issues associated with creep.

Table 6. Temperature limits for SA-508 and SA-336 (Grade 91) steels.

Steel	Operation			
	Unlimited	**< $3x10^5$ h**	**< 3000 h**	**< 1000 h**
SA-508	T < 371 °C		371 < T < 427 °C	427 < T < 538 °C
SA-336	T < 371 °C	371 < T < 590 °C		590 < T < 650°C

3.4.1 Prismatic Design

The evaluation of the VHTR containing prismatic blocks is based on the design of the GT-MHR[25]. The GT-MHR was designed to operate at core inlet and outlet temperatures of 491/850°C. The inside wall temperature of the reactor vessel was 485 °C, which was just slightly below the core inlet fluid temperature. The reactor vessel was to be fabricated from 9Cr-1Mo-V (Grade 91) steel. The vessel design limits used by General Atomics were 495 °C for 4.6 x 10^5 h (normal operation) and 538°C for 1.0 x 10^3 h (transient conditions). General Atomics reported a maximum reactor vessel (midwall) temperature of 490°C versus an accident limit of 565°C during a depressurized conduction cooldown accident. Note that the temperature limits used by General Atomics predated the ASME code for 9Cr-1Mo-V steel and were about 100°C lower than the values given in Table 6.

Preliminary discussion regarding the NGNP has related to operation at a higher outlet temperature than the GT-MHR to increase the efficiency of the electrical production cycle and/or to allow for efficient production of hydrogen. Although the NGNP has not yet been designed, current goals call for outlet temperatures in the range of 900 to 950°C. Raising the GT-MHR inlet and outlet temperatures by 50 to 100°C would achieve the desired temperature goal for efficiency, but would also increase the maximum reactor vessel temperature. Maintaining the inlet temperature at the GT-MHR value and reducing the core flow to achieve the desired outlet temperature would keep the vessel temperature relatively low but would cause concerns about flow starvation in the hot channels of the core.

Reza et al.[26] evaluated design modifications to the GT-MHR that routed the inlet flow through holes in the outer reflector rather than through channels between the core barrel and the reactor vessel. This design prevented the inlet flow from contacting the reactor vessel, which enabled the inlet and outlet fluid temperatures to be increased while lowering the operating temperature of the reactor vessel. Detailed simulations of the vessel lower head were not performed, but it was assumed that the lower head could be isolated from the inlet flow, such as by applying insulation or by using a distribution header to supply the flow holes in the outer reflector.

This evaluation utilized the GT-MHR design as modified by Reza et al.[26]. RELAP5-3D[21] calculations were performed to determine the reactor vessel temperature during normal operation and a depressurized conduction cooldown accident. The RELAP5-3D input model was based on the model of Reza et al.[26], which in turn was based on the model developed for the Next Generation Nuclear Plant by MacDonald et al.[27]. The model of Reza et al.[26] was modified to produce the desired core inlet/outlet temperature conditions for this evaluation.

Steady-state calculations were performed with vessel outlet fluid temperatures of 900 and 950°C. Results are presented in Table 7. The table contains two vessel temperatures, one at the inner wall and the other at the radial center of the inner and outer walls. Both maximum temperatures generally occurred at the same axial location in the vessel. The midwall temperature is appropriate for comparison with the temperature limits given in Table 6 because the ASME code[24] specifies the use of the wall-averaged temperature. The higher temperature at the inner wall is an indication of the heat conduction through the wall and the subsequent heat transfer to the RCCS.

Table 7. Calculated thermal-hydraulic conditions during normal operation for the prismatic VHTR.

Parameter	T_{out} = 900°C	T_{out} = 950°C
Power, MW	600	600
Pressure, MPa	7.00	7.00
Differential pressure, MPa	0.0764	0.0802
Inlet temperature, °C	540	590
Outlet temperature, °C	900	950
Flow rate, kg/s	325.0	325.1
Core bypass, %	9.59	9.53
Maximum vessel temperature (inner wall), °C	410	447
Maximum vessel temperature (midwall), °C	388	421
Maximum fuel temperature, °C	1064	1112
RCCS power, MW	1.83	2.13

The steady-state reactor vessel temperature was far less than the limit of 590°C for normal operation with SA336 steel given in Table 6. The calculated results are expected to be reasonably accurate because the original RELAP5-3D model developed by MacDonald et al.[27] was benchmarked against results calculated by General Atomics for the GT-MHR. The benchmarking showed that the maximum reactor vessel temperatures predicted by RELAP5-3D during normal operation and during the depressurized conduction cooldown accident were within 5 and 13°C, respectively, of the values reported by General Atomics.

The calculated response of the maximum midwall temperature of the reactor vessel during a depressurized conduction cooldown accident is shown in Figure 13. During normal operation, heat transfer due to convection to the fluid in the gaps between reflector blocks and the coolant channels in the outer reflector was much greater than the heat transfer due to conduction through the reflector. Conduction became the dominant heat transfer mechanism after the scram and the depressurization. The temperatures of the core barrel and the reactor vessel initially decreased during the accident due to the decrease in convective heat transfer combined with the long thermal response time of the outer reflector. Eventually the temperature gradients due to conduction worked their way through the outer reflector and the vessel temperature increased because of the mismatch between the core decay power and the heat removed by the RCCS (see Figure 14). The reactor vessel temperature peaked near the time when the power removed by the RCCS exceeded the decay heat.

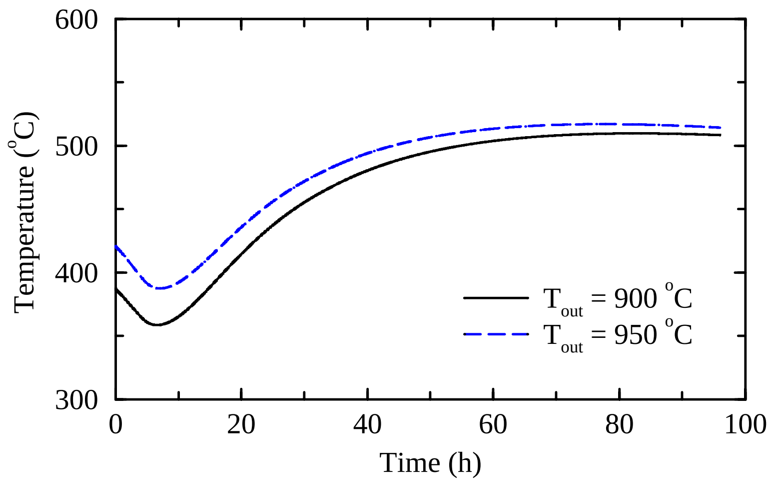


Figure 13. Maximum reactor vessel temperatures during a depressurized conduction cooldown accident with the prismatic design.

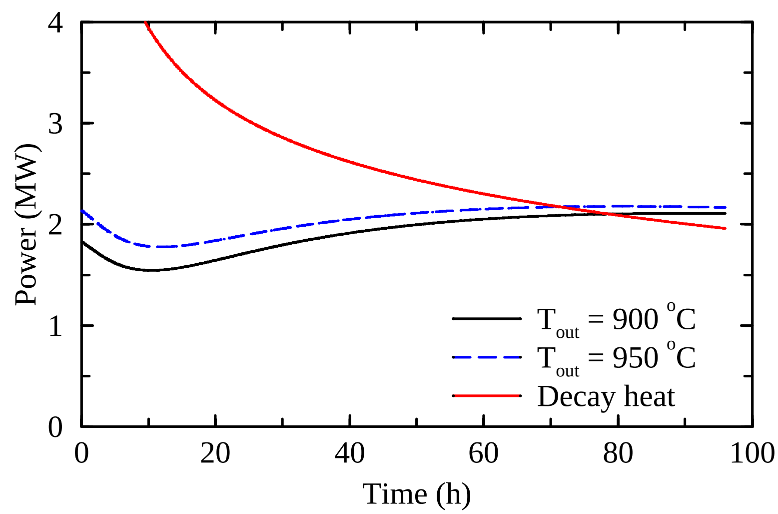


Figure 14. A comparison of RCCS heat removal and core decay power during a depressurized conduction cooldown accident with the prismatic design.

Table 8 presents maximum calculated temperatures during the conduction cooldown event. The maximum predicted midwall temperatures remained well below the accident limit of 650°C given in Table 6. A comparison of Tables 7 and 8 reveals that a 50°C increase in outlet temperature caused the maximum midwall temperature to increase by 33°C at steady state, but by only 7°C during the accident.

Table 8. Maximum temperatures during the depressurized conduction cooldown accident in the prismatic VHTR.

Parameter	T_{out} = 900°C	T_{out} = 950°C
Maximum vessel temperature (inner wall), °C	553	562
Maximum vessel temperature (midwall), °C	510	517
Maximum fuel temperature, °C	1501	1526

Figure 15 shows the axial temperature profiles at the inner surface of the reactor vessel at two times, 0.0 h which corresponds to normal, full-power operation and 83.3 h which is near the time of the peak temperature during the depressurized conduction cooldown accident. The reactor vessel temperature during normal operation is nearly independent of elevation in the region between the bottom of the active fuel (BAF) and the top of the active fuel (TAF). The wall temperature decreased above the active fuel because of the presence of a thermal shield. The calculated temperature below the active core decreased because the model assumed that the inlet and outlet fluids did not contact the vessel and that the only source of heat transfer to the lower head was due to radiation from adjacent heat structures. A more detailed evaluation of this potentially limiting region will be required once the design is completed.

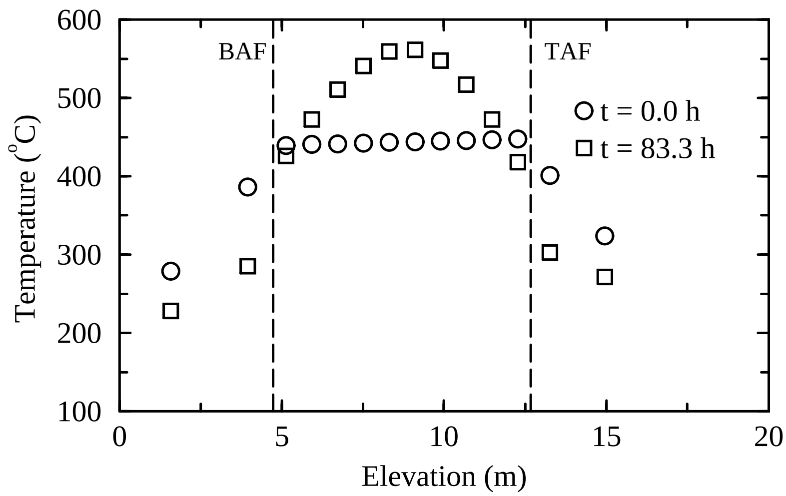


Figure 15. Reactor vessel axial temperature profile for the prismatic design with an outlet fluid temperature of 950°C.

The axial temperature profile shifted significantly during the depressurized conduction cooldown accident. At the time of the peak temperature, the maximum vessel temperature occurred near the axial center of the core, which was the peak power location in this analysis.

3.4.2 Pebble-Bed Design

The evaluation of the pebble-bed VHTR was based on PEBBED-THERMIX and RELAP5-3D models developed specifically for this task.

The PBMR has a rated thermal power of 400 MW and thus its basic design had to be modified to accommodate the higher power of a potential NGNP. A genetic algorithm was employed to automatically search for a core geometry that would yield a vessel comparable in size to other 600 MW high temperature reactor designs and achieve passive safety (i.e., the fuel temperature will remain under 1600°C during a depressurized conduction cooldown accident). Using the PBMR-400 design as a starting point, various perturbations to the core and reflector dimensions were explored and analyzed using PEBBED-THERMIX. Once a promising candidate was identified, final iterations with COMBINE were executed to generate a self-consistent physical model. PEBBED was also used to design a 600 MW pebble-bed VHTR in a previous study[27] but without detailed THERMIX analysis and temperature feedback in the cross sections. The reactor obtained in this current work is thus a more realistic model. The basic reactor dimensions are shown in Table 9.

Table 9. Dimensions of 600 MW pebble-bed VHTR obtained using a genetic algorithm search.

Dimension	Value (m)
Height of active core	10.96
Radius of inner reflector	1.582
Outer radius of fuel annulus	2.464
Outer radius of outer reflector	3.072

The thicknesses of the top and bottom axial reflectors (1.86 m and 4.36 m) were not changed from the values used in the PBMR-400 Coupled-Code Benchmark model[28]. Control rods (B4C) were assumed to be partially inserted into the outer reflector to allow for xenon override.

Steady-state calculations were performed with outlet temperatures of 900 and 950°C. Results are presented in Table 10.

Table 10. RELAP5-3D and THERMIX calculated thermal-hydraulic conditions during normal operation for the pebble-bed VHTR.

Parameter	T_{out} = 900°C		T_{out} = 950°C	
	RELAP5-3D	THERMIX	RELAP5-3D	THERMIX
Power, MW	600	600	600	600
Pressure, MPa	9.00	9.00	9.00	9.00
Differential pressure, MPa	0.290	0.216	0.238	0.181
Inlet temperature, °C	482	482	482	482
Outlet temperature, °C	900	899	950	948
Flow rate, kg/s	275	276	246	247
Core bypass, %	9.8	10.1	10.0	9.9
Maximum vessel temperature (inner wall), °C	354	396	368	410
Maximum vessel temperature (midwall), °C	342	374	356	393
Maximum fuel temperature, °C	1072	1043	1141	1096
RCCS power, MW	1.43		1.46	

The response of the pebble-bed VHTR as calculated by RELAP5-3D and THERMIX during a depressurized conduction cooldown accident is shown in the next several figures. Figure 16 shows the THERMIX-computed axial temperature profile at the inner surface of the reactor vessel at various times during the accident. As was observed with the prismatic core, the maximum temperature initially decreased following the reactor scram and the decrease in convective heat transfer following the depressurization (see Figure 17). The maximum vessel temperature eventually increased because of the mismatch between the core decay power and the heat removed by the RCCS (see Figure 18). The sharp change in slope near ten hours occurred as the location of the maximum temperature switched from near the bottom of the vessel to near the axial center of the core. The maximum temperature occurred about 80 h after the start of the accident and remained far below the accident limit of 538°C for SA508 steel.

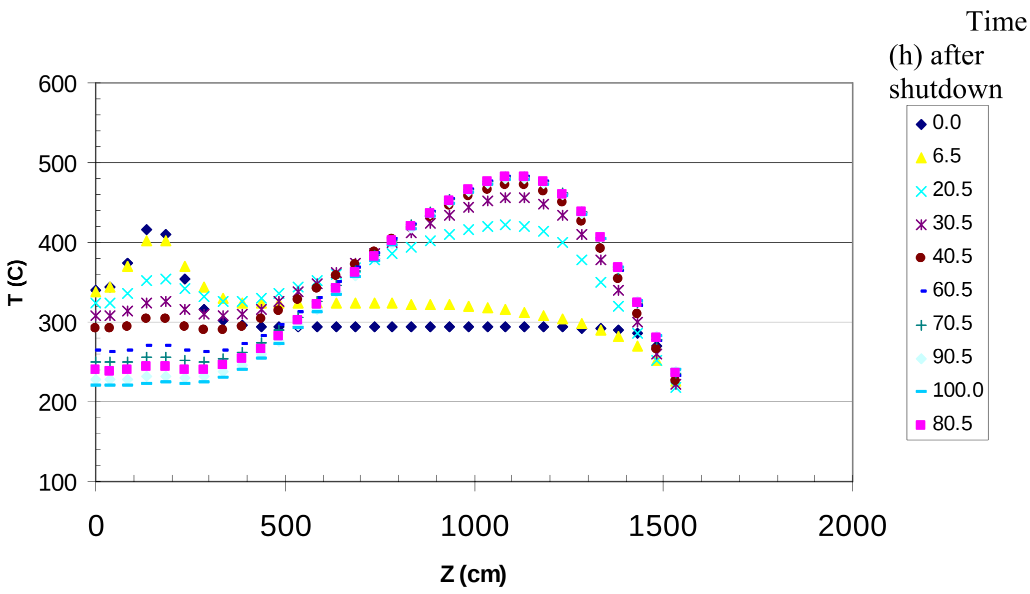


Figure 16. Reactor pressure vessel inner wall temperature profile during a depressurized conduction cooldown accident (950°C outlet fluid temperature).

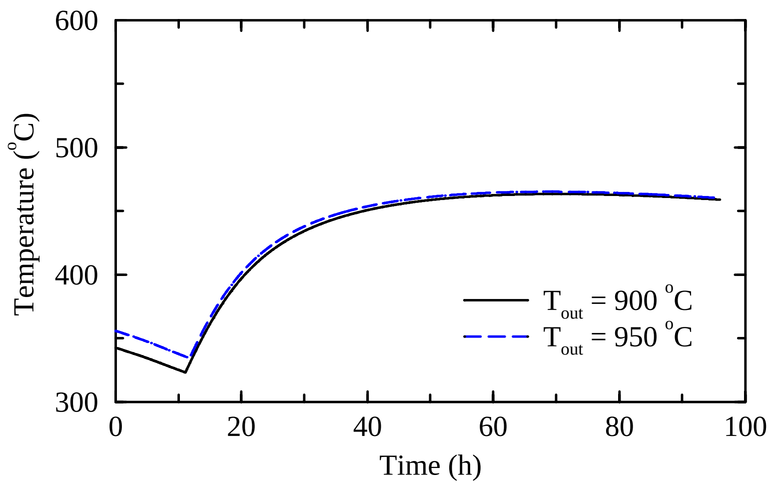


Figure 17. RELAP5-3D calculated maximum reactor vessel midwall temperatures during a depressurized conduction cooldown accident with the pebble-bed design.

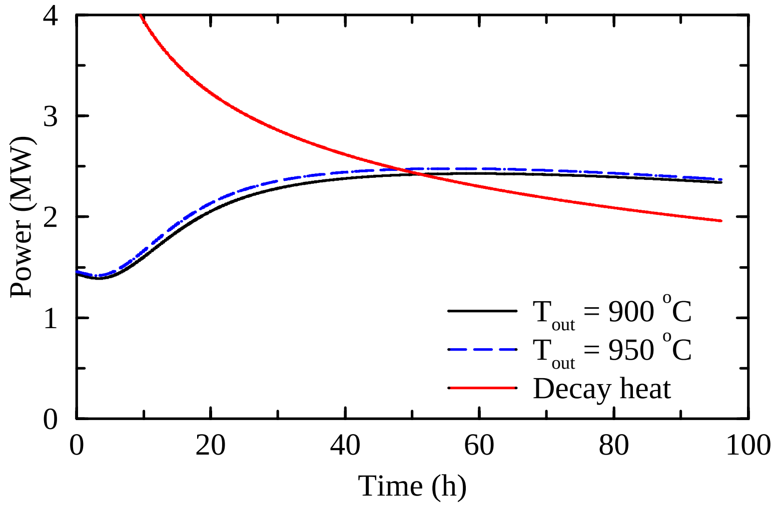


Figure 18. A comparison of RELAP5-3D calculated RCCS heat removal and core decay power during a depressurized conduction cooldown accident with the pebble-bed design.

Figure 19 shows axial temperature profiles at the inner surface of the reactor vessel at 0.0 and 83.3 h after the start of the accident. RELAP5-3D results are shown with symbols while the THERMIX results are shown with solid lines. The calculated temperature profiles were similar with both codes. At steady state, the temperature between the BAF and TAF was nearly independent of the elevation. The maximum temperature occurred below the outlet plenum, which was heated by the outlet flow but not cooled by flow through the riser and bypass flow paths as occurred at higher elevations. The maximum temperature obtained from THERMIX was about 40°C higher than the value obtained from RELAP5-3D, perhaps because of the more detailed nodalization used in the THERMIX model. The calculated trends were similar with both codes during the conduction cooldown accident. In particular, the maximum temperature during the conduction cooldown accident occurred near the elevation of the peak power location in the core, which was slightly above the core centerline for this analysis. The maximum calculated values were within about 10°C during the accident.

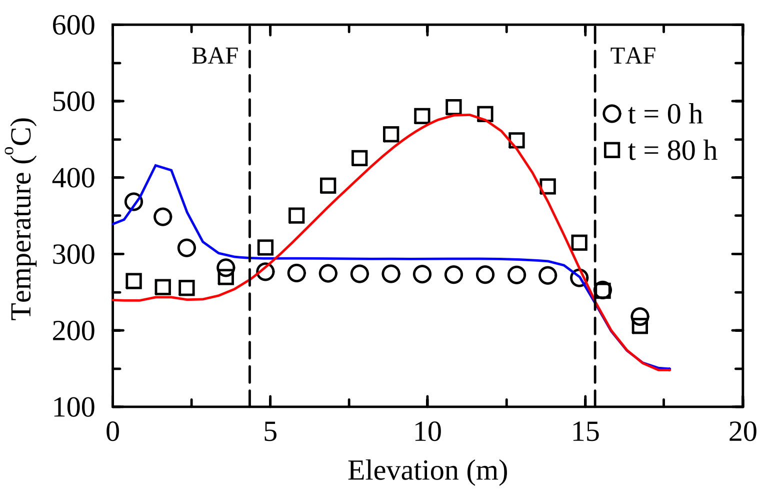


Figure 19. Reactor vessel axial temperature profiles for the pebble-bed design with an outlet fluid temperature of 950°C.

The maximum temperatures calculated during the conduction cooldown event are given in Table 11. The 50°C variation in initial outlet temperature had only a small effect on the maximum temperatures during the accident.

Table 11. Maximum temperatures during the depressurized conduction cooldown accident for the pebble-bed VHTR.

Parameter	T_{out} = 900°C		T_{out} = 950°C	
	RELAP5-3D	THERMIX	RELAP5-3D	THERMIX
Maximum vessel temperature (inner wall), °C	492	480	494	482
Maximum vessel temperature (midwall), °C	463	451	465	452
Maximum fuel temperature, °C	1486	1550	1493	1554

For the 950°C outlet temperature case, the THERMIX calculation was allowed to run until the reactor pressure vessel temperature dropped below 371°C, about 540 hours after shutdown (the control rods were assumed fully inserted to prevent re-criticality). The pressure vessel thus can be expected to remain at an elevated temperature for about 22 days assuming no active measures are taken to cool it. According to Table 11, SA508 steel can withstand elevated temperatures for at least 40 days.

3.4.3 Conclusions

Current designs for an NGNP type VHTR with a prismatic core do not allow the use of SA508 steel for the reactor vessel based on the analyses performed in this study. Best-estimate calculations of the maximum reactor vessel temperature during normal operation exceeded the value allowed for SA508, but were at least 170°C below the 590°C limit for SA336 when the design improvements described by Reza et al.[26] were modeled. The maximum temperature during normal operation was just slightly below the 425°C value where creep is expected to become an issue and further design improvements may be required to provide additional margin. Note that the design of the important lower head region has yet to be completed with the prismatic core.

For the pebble-bed design, the maximum vessel wall temperature during normal operation was 16°C below the allowed value for unlimited operation of SA508 based on the RELAP5-3D calculations, but was 21°C above the allowed value based on the PEBBED-THERMIX calculations. These calculations indicate that some design changes, such as the use of active vessel cooling, are required to limit the vessel temperature during normal operation. Even though the values calculated with RELAP5-3D were lower than the temperature limit, the margin to the limit was probably inadequate based on the uncertainty in the calculation.

The maximum vessel temperatures increased significantly during a depressurized conduction cooldown accident. However, the temperatures remained at least 70°C below the accident limits allowed by the ASME code for 1000 h of operation with both designs.

The PEBBED-THERMIX calculations of the pebble-bed VHTR were generally confirmed by the RELAP5-3D calculations. During normal operation, the predicted axial profiles in reactor vessel temperature were similar with both codes, but the maximum predicted temperature was about 40°C higher with PEBBED-THERMIX. The trends of the calculated vessel temperatures and the maximum values were similar during the depressurized conduction cooldown accident. This agreement is considered reasonable based on the expected uncertainty in either calculation.

4. Discussion of Significant VHTR Materials Issues

The discussion in Section 4 is given primarily from the perspective of the NGNP Materials Program Principal Investigators responsible for the detailed planning and execution of the R&D tasks performed. The materials issues discussed are associated with the following materials categories that have been envisioned to be a key area required to support the development of the NGNP:

1. Nuclear Graphite
2. Reactor Pressure Vessel and Class 1 Boundary Steel
3. Other High Temperature Metallic Alloys
4. High Temperature Design Methodology
5. Nuclear Ceramics and Composites
6. Molten Salt/ Metallic Alloy Interactions at Very High Temperature

Activities regarding various codes and standards implementation and development required to utilize these materials in the construction of the NGNP are integrated into the discussion for each section. Other lower priority materials systems that may be eventually be required for construction of the NGNP are not discussed because funding has not been provided to include these materials areas in the program. A discussion and listing of these areas are given in Section 3.2 of the NGNP Materials Program Plan, INL/EXT-05-00758, Revision 2, September 2005.

The basic issues discussed for each of these areas can be divided into the following categories:

1. Significant technical issues involved in utilizing these material systems in the NGNP
2. Significant ASME Code issues involved in utilizing these material systems in the NGNP
3. Significant ASTM Standards issues involved in utilizing these material systems in the NGNP
4. Significant risk issues involved in utilizing these material systems in the construction of the NGNP that relate primarily to budget limitations and NGNP schedule requirements imposed in the NGNP Preliminary Project Management Plan, INL/EXT-05-00952, Rev.1, March 2006.
5. Significant procurement cost and availability issues involved in utilizing these material systems in the NGNP

These issues were discussed for each material category noted during a facilitated meeting of program PIs and managers, vendor representatives and additional technical experts held in Salt Lake City on June 21 and 22, 2006 at the Sheraton City Center Hotel. A summary of this meeting can be obtained by request by contacting Kevan Weaver at kevan.weaver @inl.gov.

4.1 Nuclear Graphite

The NGNP graphite qualification program will provide irradiation and non-irradiation data to the selected NGNP reactor vendor to assist in the NRC license application for graphite components. The scope of data will be determined by the NGNP in consultation with graphite suppliers, potential reactor vendors, and the NRC. The DOE laboratories have facilities and capabilities which are unique and they can perform irradiation testing of graphite candidates, producing data which the suppliers, vendors, and the NRC need. The NGNP will select one graphite grade for the prismatic and a different grade of graphite for the pebble bed reactor and establish a program to acquire data for the design, accident analysis, and licensing of the NGNP reactor.

The INL has released the Preliminary Program Management Plan for the NGNP and is endorsing Option 2 schedule, Figure 20, which is the most practical balance of overall technology development risk, and project design, licensing and construction risk. This schedule requires a reactor technology selection by the end of preliminary design, CD-2, in FY-10. By the beginning of final design in FY-11 enough data must be available to license the graphite components for the reactor technology chosen. Life time irradiation doses can not be acquired by the beginning of final design, so graphite components will have to be licensed for less than life time dose which could be acquired through irradiation testing by final design. Once the final reactor technology is chosen, the full life time irradiation testing program will begin.

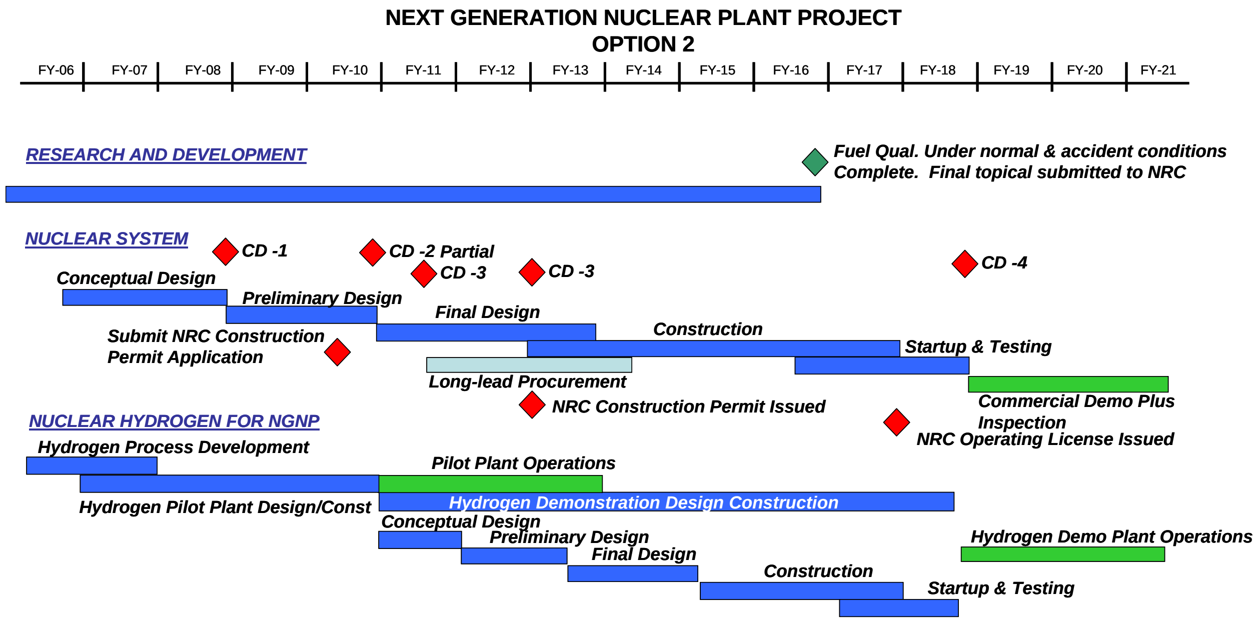


Figure 20. Option 2 Balance Risk Schedule

Graphite from the reactor vendor's technology not selected will be dropped from the full life time irradiation testing program. By Partial CD-3, mid FY-11, a complete NRC license data package must be available for submittal to the NRC for the graphite components. This initial NRC graphite component license will be for a dose less than full life time dose. Once full life time irradiations have been completed, the NGNP will submit a topical report to the NRC extending the graphite component life time to full life time dose. The NGNP will consult with the reactor vendors and graphite suppliers to determine the initial limited graphite component lifetimes during initial operation and the irradiation data that should be obtained to support the initial license. This licensing strategy will have to be developed in collaboration with the NRC.

Discussions with the NRC Office of Nuclear Regulatory Research need to commence in FY-07. These discussions will be to establish and define NRC graphite research needs for non-irradiated and irradiated graphite behavior, and will assist the NRC to establish the basis to license the NGNP graphite components. The scope and cost of the research activities identified need to be incorporated into the graphite qualification program as soon as possible.

The NGNP graphite qualification program will have to accelerate many of its supporting activities to have non-irradiated data available by Partial CD-3 design for the initial graphite component licensing package. First, the MOU between DOE and the NRC must be signed by the beginning of FY-07. After the MOU is in place, the INL will be able to collaborate and negotiate licensing strategies with the NRC and reactor vendors by the beginning of FY-09. Second, development of the ASME graphite probabilistic design code must be accelerated to the extent a strong candidate draft has at least received a round of balloting from the full ASME Section III board, and the full ASME Board on Nuclear Codes and Standards by the first quarter of FY-11 (partial CD-3). After the ASME has completed its balloting, the NGNP will petition the NRC for early adoption of the ASME graphite design code so that the final design phase of the plant can be completed by FY-13. During the ASME code development, new ASTM test procedures must be developed and tested through round robin for NDE of graphite and small component testing. Third, graphite in-core reliability and performance modeling must be completed and fully implemented by the end of FY-10 (CD-2). This effort will interact with ASME code development providing a test bed for the benchmarking exercises and an interface between the non-irradiated and irradiated graphite characterization programs.

There are two graphite suppliers for the NGNP reactor concepts. For the prismatic reactor concept there are two prospective graphites: PCEA from GrafTech and NBG-17 from SGL. For the pebble bed concept there is only NBG-18 from SGL. The prismatic and pebble bed graphites can not be interchanged. The larger grain NBG-18 cannot be used in prismatic design and the economics of the finer grain PCEA and NBG-17 restrict the use of these graphites in a pebble bed design. The NGNP has selected a single graphite grade for the prismatic. SGL Grade NBG-17 is currently in a pre-production development stage and very few full scale-billets have been produced to date. In contrast, GrafTech Grade PCEA is considerably more mature. A small production batch of PCEA has been manufactured and billets purchased by ORNL and AREVA for characterization. Given the need to rationalize the graphite qualification program and the development status of NBG-17, it was determined to eliminate NBG-17 from the program and proceed with the qualification of PCEA graphite only.

After Partial CD-3, long-lead purchase orders, for items such as graphite, will be placed. The NGNP Materials Program must develop a graphite specification for both a prismatic and pebble bed graphite cores by mid FY-08.

The development of these graphite specifications will require full- scale qualification graphite production runs starting in late FY-08 and continuing through FY-10 for both graphite grades. It is imperative that these qualification production runs start in late FY-08 because the duration of graphite production runs are 6 to 9 months. Several full-scale production runs will be required to show competency in production and gather the required post-production properties and statistical data.

Non-irradiated mechanical properties needed for the initial license application must be completed for both graphite grades by Partial CD-3. The characterization work must look at non-irradiated mechanical variances within the billet, from billet to billet, lot to lot, and run to run. Some characterization work is currently being performed on the production billets produced under the generic ASTM nuclear graphite specification. The graphite specimens in the current irradiation test were fabricated from the same production batch. Thus, the characterization from current billets compared to characterization of billets produced under the NGNP graphite specification is needed to establish a comparison between the different graphite batches. If the irradiation data from current irradiation testing is to be used in the licensing application, then the characterization comparison will be crucial to show the graphite produced under the NGNP specification will behave the same as graphite produced under the ASTM specification.

A new activity has been added to the NGNP graphite qualification program to identify (by CD-3) the disposal issues from C-14 burden in the geological repository, or decide if reprocessing graphite using irradiated graphite in the production of new graphite is feasible. Reprocessing graphite would require another irradiation qualification program beyond operational startup of the NGNP reactor in order to gather enough irradiation data to draft a NRC topical to permit the use of reprocessed graphite components if needed.

ASTM D02.F Committee on Manufactured Carbon and Graphites activities in FY-07 include completion and main committee balloting of the second nuclear graphite materials specification, completion of the round-robins currently being performed for standards on graphite x-ray diffraction, graphite air oxidation, and graphite fracture toughness. A new round-robin relating to the determination of Weibull parameters will be initiated. New standards will be drafted for glow discharge mass spectroscopy determination for chemical impurities, non-destructive examination methods, and small (irradiation) specimen test methods for graphite.

A Graphite Working Group has been formed under the Gen-IV International Forum Project Management Board – Components and Materials. The role of the Graphite Working Group (GWG) is to:

1. Coordinate details of sample exchanges and collaborative experiments
2. Review graphite data produced by GIF partners
3. Coordinate efforts to develop physically-based models for the behavior of graphite
4. Liaise with international standards and codes bodies
5. Administer eventual transfer of data to IAEA graphite database
6. Provide input to the Graphite TDA technical coordinator's annual report to the GIF M&C PMB.

Moreover, the GWG shall review and make recommendations at the following R&D stages:

1. Requirements definition
2. Experimental program design
3. Experimental data
4. Models and analysis

Additionally, the GWG shall facilitate the coordination of:

1. Graphite supplier quality requirements and quality audits
2. Data quality from member nations

The GWG shall coordinate research and development activities throughout the graphite lifecycle: including VHTR data acquisition, design methodology and construction, operation and inspection, decommissioning and disposal. The GWG's level of effort will be in proportion to the scope of a particular technology task description. The GWG shall review the Graphite Technology Development Area Collaboration Plan and report progress annually. The GWG shall identify specific program deliverables and sub-tasks for the forthcoming calendar year. Finally, the GWG shall assess the value of experimental data/technical inputs.

In FY-06 two irradiation experiments were being designed: the ATR AGC-1 and HFIR HTV 1 and 2. The ATR AGC-1 experiment will irradiate 90 creep specimen pairs and over 300 piggyback samples at 900°C for up to 4 dpa. It is still planned to start this irradiation in FY-08. The irradiation data will be used to assist in developing graphite core performance computation models. Other AGC irradiations previously planned are being put on hold until NGNP vendor discussions are held. The HTV 1 and 2 experiments were to acquire high temperature data at 900, 1200, and 1500°C (1-2 and 3-6 dpa, respectively) for use in NGNP design and determine parameters needed for design of AGC capsules at 1200°C. These experiments will obtain data on high temperature dimensional changes, thermal conductivity degradation, elastic constants and compressive strength.

The primary objective of irradiation capsule AGC-1 is to provide irradiation creep design data on candidate graphites for the NGNP program. A further objective is to provide design data for the effects of neutron irradiation on the properties of a range of NGNP relevant graphites, such data to include: dimensional changes, strength, elastic modulus, thermal conductivity and coefficient of thermal expansion (CTE). As noted above, this later objective is being down-graded, however, it will remain for the AGC-1 experiment because it is already designed. Moreover, this experiment will provide valuable data on the single-crystal irradiation behavior of graphites to be derived from the inclusion of highly oriented pyrolytic graphite (HOPG) in this experiment.

AGC-1 is one of a series of advanced test reactor (ATR) irradiation creep capsules that were intended to provide graphite irradiation creep data for NGNP relevant graphites. The purpose of the ATR Graphite Creep-1 (AGC-1) capsule is to provide design data on the effects of irradiation on NGNP relevant graphites over the neutron dose range 0.53 x 1021 n/cm^2 - 5.8 x 1021 n/cm^2 [E>0.1 MeV] or 0.39 – 4.2 dpa at an irradiation temperature of 900 °C. Additional advanced graphite reactor capsules may be planned (as noted above) for irradiations at 600 and 1200 °C to provide design data over the anticipated graphite in-reactor operating temperature for a PMR design.

At the time of this report, the FY-07 budget will be restricted due to early year continuing resolution budget restrictions. The restricted funding will only allow the AGC-1 capsule design, fabrication, gas control system installation to continue. Restricted FY-07 funding will continue to support the ASME code committee involvement and will provide funding for discussions with reactor vendors and the NRC. The HTV 1 and 2 capsule design will be resumed once the mid year budget funding has been released. Other activities, such as graphite vendor qualification, graphite billet characterization and multi-axial testing will be funded after mid year budgets are released. A detailed irradiation plan will drafted for the initial NRC graphite component licensing data, after mid year funding is received.

4.2 Reactor Pressure Vessel and Class 1 Steel

Possible primary coolant pressure boundary systems envisioned for the NGNP are illustrated in Section 3 based on the current AREVA and PBMR VHTR designs. The various design options envisioned include a large reactor pressure vessel (RPV) containing the core and internals, a pressure-containing cross vessel (CV) that joins the two vessels, and a second vessel at contains an intermediate heat exchanger (IHX) in the case of the AREVA design. The large thicknesses in the RPV are of particular relevance to the selection, fabrication, and operation of the reactor. The RPV, CV, and IHX containment vessels will be exposed to air on the outside and helium on the inside, with emissivity of the chosen material an important factor regarding radiation of heat from the component to the surrounding air to ensure adequate cooling during accident conditions. Key reactor coolant primary pressure boundary operating conditions that affect candidate material selection are given in Table 12.

Table 12. Nominal Class 1 Boundary Conditions Currently Envisioned for the NGNP Based on Current Proposed Design Concepts

Component	Normal VHTR System Operating Conditions		Abnormal Conditions	Estimated Component Size
	Temp. [ºC] Pressure [MPa]	**Neutron Fluence, E>0.1 MeV (dpa)**		
Reactor Pressure Vessel	350-450 ºC [29] b	$1x10^{18}$ n/cm-2 per 40 years [30]	≈450-562ºC at 1 atm for 300 hours [31]	Diameter: >6m,
Reactor Pressure Vessel	[5.5-9.0 MPa] [30]			Thickness: 100-300mm, Height: >24m
Cross Vessel	unknown	negligible [30]	300 to 560 ºC for 200 h	Diameter: >2.5m,
Cross Vessel	[7.4-8.0 MPa]	(0.077 dpa)	[5.5-9.0 MPa]	Thickness: >100mm Length: 4-5m
Secondary Vessel [[i]]	Unknown	Negligible	Unknown	Unknown
	Unknown			Unknown
Closure	Unknown	Unknown	Unknown	
Bolting		Unknown		

Because of the high temperatures originally envisioned for the RPV in the designs of Generation IV reactors, including the NGNP (up to 650^{o}C), ferritic and martensitic steels were considered for this application, the primary emphasis being on the 9%Cr steels. In particular, the primary candidate is Mod9Cr1Mo steel (Gr91 or P91 steel). However, some design options such as the PBMR design incorporates an active cooling system; as such the vessel will operate at lower temperatures, thus providing an opportunity to use a lower alloy steel. Low alloy ferritic and bainitic steel, such as SA533 Grade B Class 1 plate or SA508 Grade 3 Class 1 forgings (508/533) are primary candidates for the lower operating temperature vessel designs for commercial light water reactors. Other lower temperature candidate alloys are the 2¼Cr1Mo steel and variants thereof, e.g. 2¼Cr1MoV. Hence, the three alloys of interest for NGNP applications are Gr91, 508/533, and 2¼Cr1Mo and these are discussed below.

4.2.1 Variants of 2¼Cr1Mo Steel

A review of the option of using 2 ¼ Cr steels quickly reveals the following:

1. The unmodified version of 2¼Cr1Mo that is code approved for use in ASME Section III (nuclear application) does not possess the mechanical properties required for NGNP vessels.

b A cooling system design and insulation will affect this range.

2. The variants of 2¼Cr1Mo, e.g. the addition of vanadium for strength, are not code approved for ASME Section III, Class 1 applications.

3. There is also a lack of knowledge regarding irradiation effects in the variants of 2¼Cr1Mo.

Hence, the 2¼Cr1Mo steel and its variants are no longer considered candidate alloys for NGNP vessels[32].

4.2.2 Background Information on Gr91 and SA 508/533 Steels

For improved corrosion and oxidation resistance, 9%Cr steels were developed in the 1960s. The microstructures of these materials were designed by balancing austenite and ferrite stabilizers to produce 100% austenite during austenitization and 100% martensite during normalizing or quenching following austenitization. The 2nd generation of 9%Cr steels included the optimization of carbon, niobium, and vanadium, with additions of nitrogen; these changes resulted in an increase in strength and maximum operating temperatures. Among these was Mod9Cr1Mo steel (T91 or Gr91: Fe-9.0Cr-1.0Mo-0.2V-0.08Nb-0.05N-0.40Mn-0.40Si-0.10C). Figure 21 illustrates the typical microstructure achieved after normalizing and tempering.

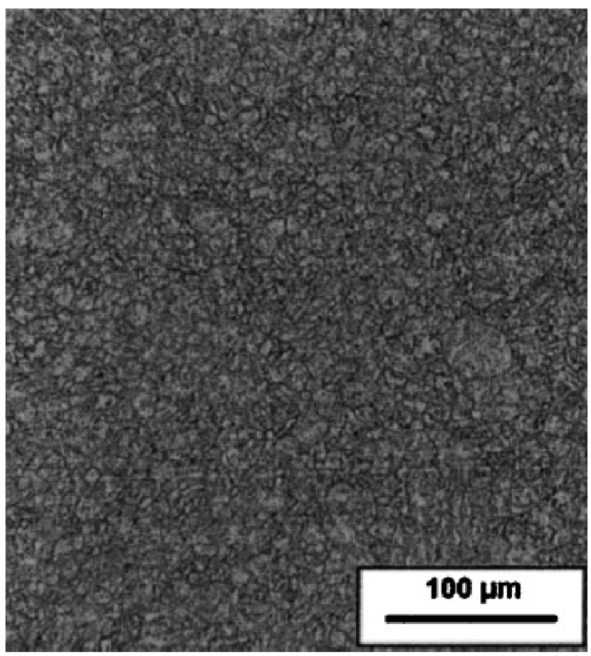


Figure 21. Optical photomicrograph of normalized-and-tempered Grade 91 steel showing tempered martensite microstructure.

Low alloy steels SA508/533 have been used extensively in LWRs (nominally Fe–1.25Mn–0.5Ni–0.5Mo–0.2C). An extensive infrastructure and industrial experience exists with these alloys, including very large and thick sections, as large as 12.4m outer diameter with section thicknesses as large as 467mm. A photograph of a very large ABWR forging of ASME SA508 is shown in Figure 22. The low alloy content of these steels results in a lower hardenability than the 9%Cr steels. Hence, for thick sections, a mixture of bainite and polygonal ferrite occurs, with variations through the thickness due to variations in cooling rates. Depending upon the cooling rates, thicker sections will contain more ferrite[33]. Figure 23 illustrates the typical microstructure of a 508/533 steel.

Figure 22. ABWR RPV beltline forging of ASME SA508 (127 tons, 7.48m OD, 7.12m ID, 3.96m high)[34].

Figure 23. Optical photomicrograph of SA 533B steel, quenched and tempered: upper bainite and granular bainite with grain size of ASMT 8-9[35]

Both Gr91 and SA508/533 are approved for use by the ASME, Section III for Class 1 applications. Gr91 steel was recently ASME code approved in 2004 for use at elevated temperature service (>371^{O}C) in Section III. The approved use of SA508/533 at or above 371^{O}C for nuclear applications is restricted; restrictions are specified in ASME Code Case N-499-2[36]. Typically, the maximum duration of an excursion event is 3,000 hours, with a temperature less than or equal to 427^{O}C (800F); stresses are limited to 184.1 MPa (26.7 ksi). Excursions above 427^{O}C (800F) are restricted to 1,000 hours or less, with various limitations on allowable stress during the excursion. The maximum acceptable excursion temperature is 538^{O}C (1000F); Table 13 summarizes the code approved excursion conditions. The same information is presented graphically in Figure 24.

Table 13. ASME Code Case N-499-2 Time and temperature dependent (Smt) allowable stress intensity values for /SA533 Grade B and SA508 Class 3[36]

	S_{mt} — Allowable Stress Intensity Values, ksi Time at Temperature, hr						
Temperature, °F	1	10	30	100	300	1000	3000
700	26.7	26.7	26.7	26.7	26.7	26.7	26.7
750	26.7	26.7	26.7	26.7	26.7	26.7	26.7
800	26.7	26.7	26.7	26.7	26.7	26.7	26.7
850	25.5	25.5	25.5	25.5	25.5	25.5	. . .
900	24.3	24.3	24.3	24.3	24.3	24.0	. . .
950	22.5	22.5	22.5	22.5	22.0	16.0	. . .
1000	20.7	20.7	20.7	18.0	14.0	9.5	. . .

NOTE:
(1) The S_{mt} values are the lower of the two stress intensity values, S_m (time-dependent) and S_t (time-independent).

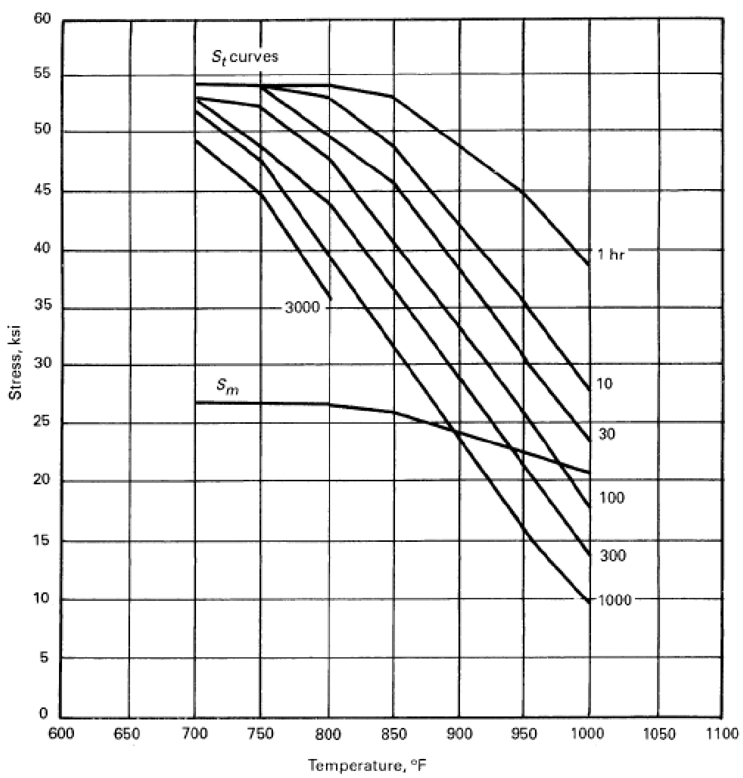


Figure 24. Time and temperature dependent strength allowables for SA508 and SA533B limited elevated temperature service in nuclear applications[36]

Figures 25 - 28 provide a comparison of the isochronous stress-strain curves for Grade 91 steel and SA508/533 at 371°C and 538°C. The enhanced short term strength (tensile curves) for Grade 91 is clearly evident. Also, the enhanced creep strength of Grade 91 steel over SA508/533 steel is also evident. Clearly, SA508/533 requires cooling to maintain tensile and creep strength, while Gr91 steel maintains a significant amount of tensile and creep strength at higher temperatures.

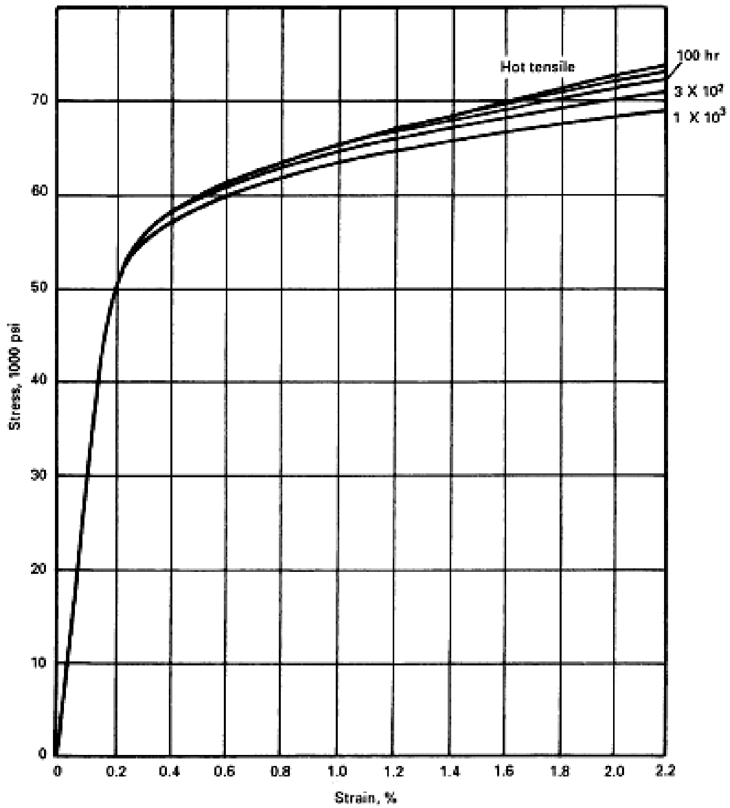


Figure 25. Isochronous stress-strain curves for SA508/SA533B steel at 371OC (700F)[36]

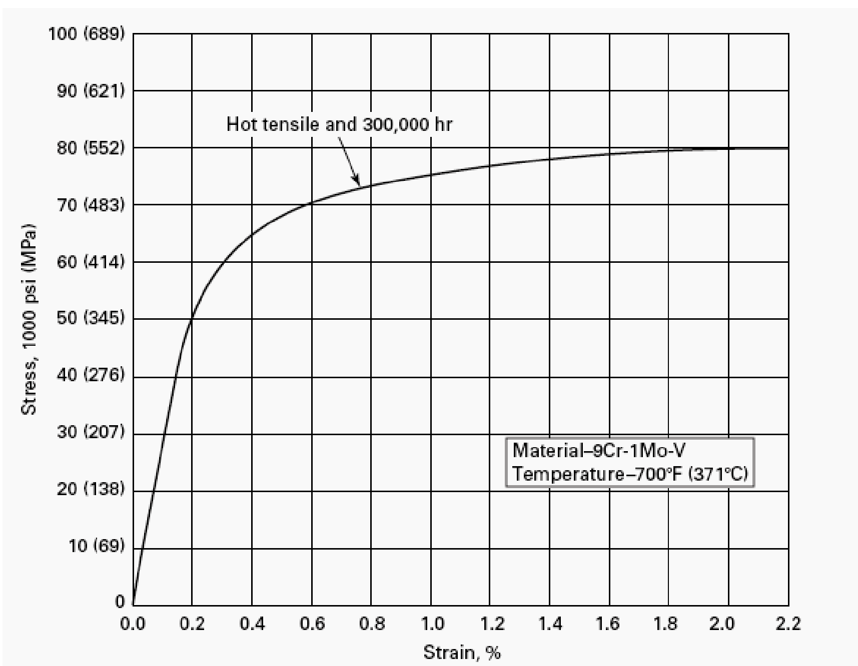


Figure 26. Isochronous stress-strain curves for Gr91 steel at 371OC (700F)[37]

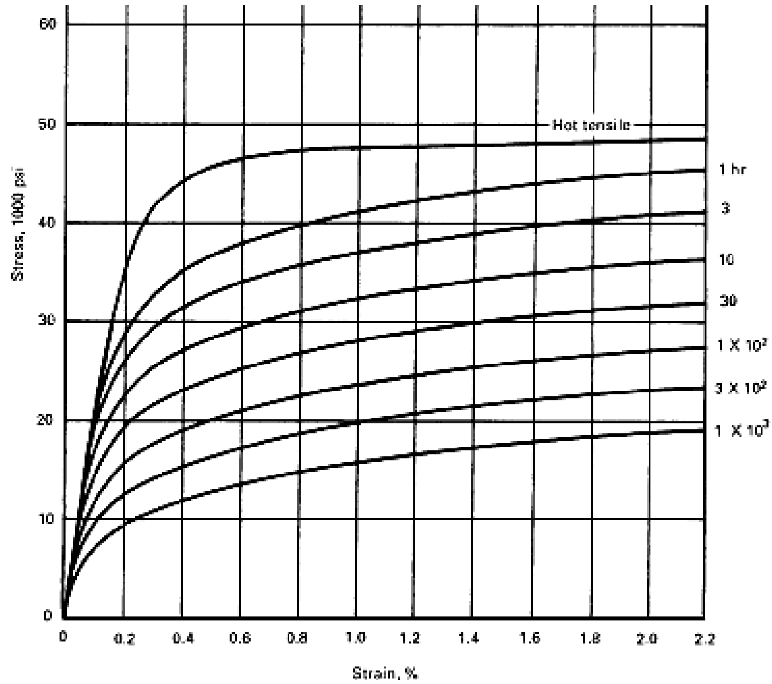


Figure 27. Isochronous stress-strain curves for SA508/SA533B steel at 538°C (1000F)[36]

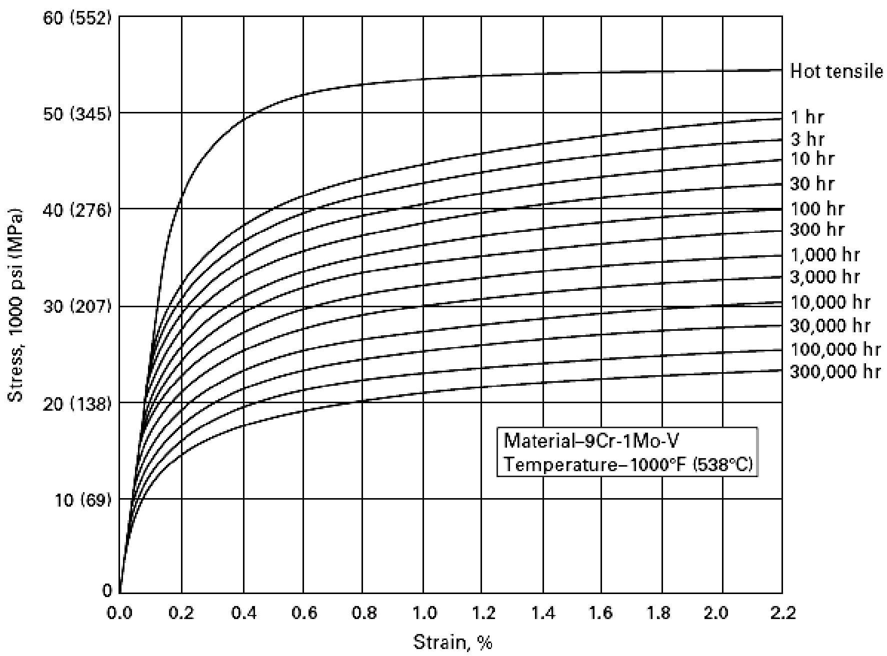


Figure 28. Isochronous stress-strain curves for Gr91 steel at 538°C (1000F)[37]

ORNL has conducted investigations of very long term aging and service-exposure effects of Gr91 steel. Figures 29 - 31 illustrate various tensile and creep properties for Gr91 steel, including virgin material, aged, and service-exposed material. The ASME code does include requirements on penalties for aging effects. Clearly, despite extended elevated temperature aging and exposure, Gr91 maintains a significant amount of strength.

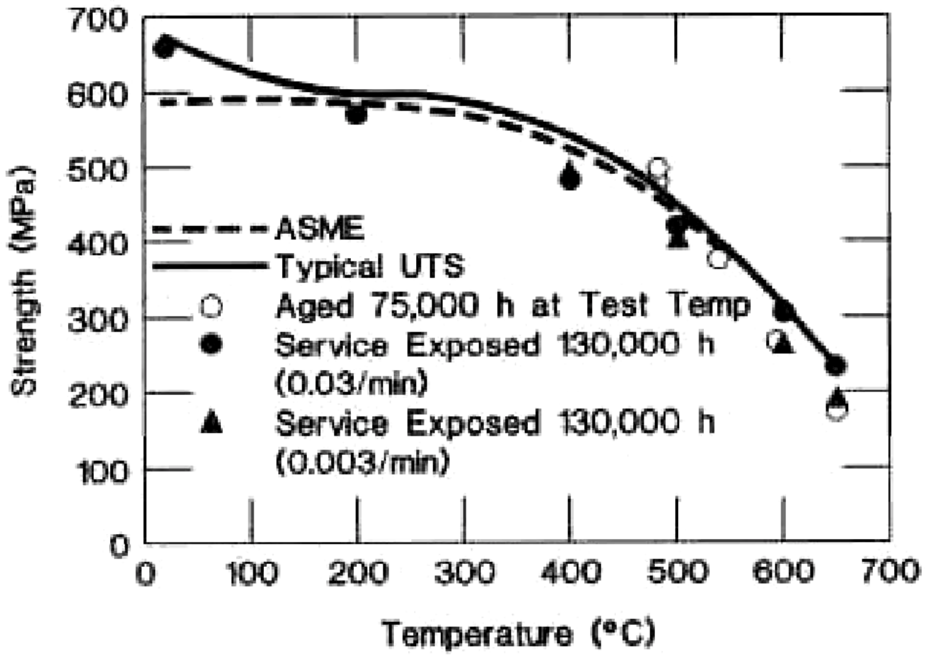


Figure 29. Comparison of ultimate tensile strength of unexposed Gr91 with aged and service-exposed materials[38]

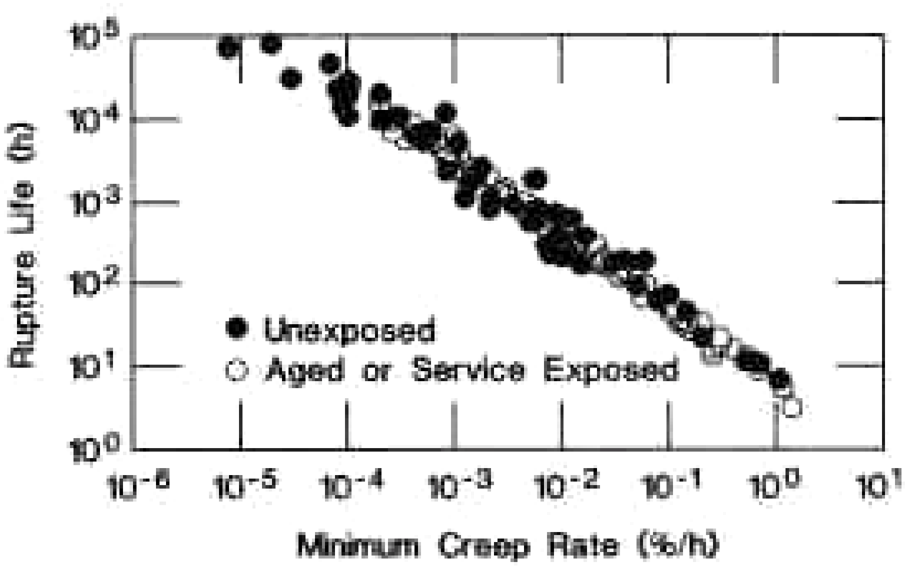


Figure 30. Monkman-Grant plot for unexposed, aged, and service-exposed Gr91 steel[38]

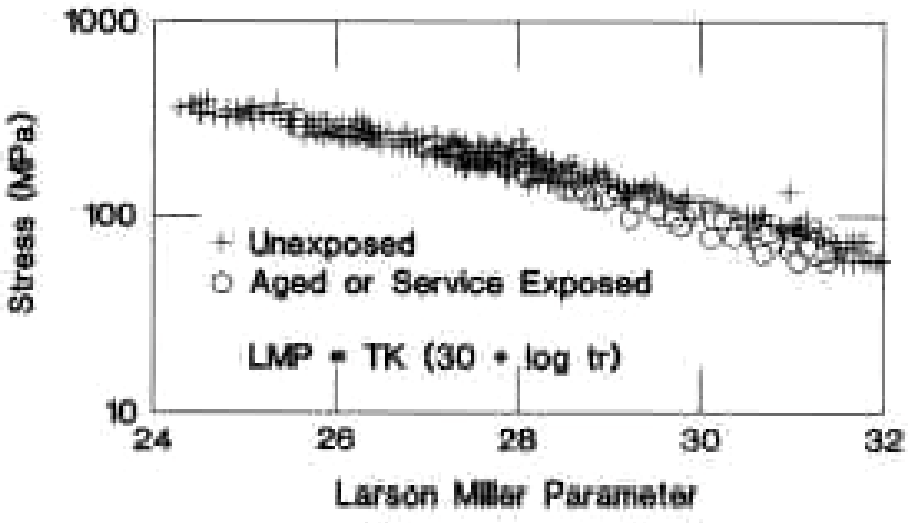


Figure 31. Stress vs. Larson-Miller parameter for unexposed, aged, and service-exposed Gr91 steel[38]

4.2.3 Reactor Pressure Vessel Irradiation Effects Considerations

Irradiation effects, primarily related to embrittlement and shifts in the ductile-to-brittle transition temperature, are currently regulated using NRC Regulatory Guide 1.99. Reg. Guide 1.99 covers pressure vessel steels with chemistries similar to A533 and A508 under a limited temperature range appropriate for light-water-reactor pressure vessel operation. Reactor pressure vessel materials with significantly different chemistries, such as 9Cr-1MoV, or that operate under significantly different temperature, flux, or spectral conditions from those addressed by Reg. Guide 1.99 will need to have their irradiation embrittlement response determined to allow the pressure vessel to be licensed by the NRC.

In order to evaluate the irradiation effects of candidate NGNP RPV alloys under the relatively low-flux test reactor conditions applicable to vessel service, a new low-flux irradiation facility will need be fabricated to replace the similar facility that was shut down recently at the Ford Test Reactor at the University of Michigan, in which such confirmatory experiments were performed. These irradiations cannot be performed in the core of test reactors, such as HFIR and ATR, because the flux is much too high and irradiation volume too low for the sizes of specimens that will be required.

The irradiation facility may be a joint DOE facility with the NRC. Current preliminary design concepts envision two separate and independent operating capsules in one facility, one for the NRC-funded Heavy-Section Steel Irradiation Program and the other for the NGNP Program. The capsules can be readily designed and fabricated to operate from 250 to 650 °C, with a maximum fast neutron flux of about 1 to 2 x 1012 n/cm2·s (>1 MeV)—low enough to be considered relevant to NGNP operating conditions. Prior to initiating the construction and operation of the RPV steel irradiation facility, approval to proceed with the design effort must first be obtained from the DOE and/or NRC and followed by site selection and placement of a contract for facility construction. Any useful hardware from the Ford Test Facility will be retrieved and used in the new facility. However, funding from both agencies must be available or a decision made by DOE to fund the facility and its operation by itself. At the moment, priorities for funding this facility will not include beginning these activities in FY07.

Although the operating temperature of the RPV may change with evolution of the design, it is currently planned to irradiate mechanical test specimens at 400 and 550°C. The choice of these temperatures is based on the assumptions that (1) 550°C is the highest possible operating temperature that can be envisaged for the RPV at this time, (2) 400°C is in the range of the lowest operating temperature that would allow for reasonable achievement of the objectives for the NGNP, and (3) the range between these temperatures would likely provide sufficient information for design and operation of the RPV at any intermediate temperature with respect to irradiation effects.

Irradiations of the preliminary candidate materials, both base metals and weldments, will begin once the facility is completed, with the choice of materials to be based on results of a literature review, as well as the baseline and aging tests completed at the time. For purposes of this plan, specimens to be irradiated will include those for tensile, hardness, creep and stress rupture, Charpy impact, fracture toughness, and fatigue crack growth testing. Based on the currently estimated maximum exposure of about 1x1019 n/cm2 (>0.1 MeV) and 0.075 dpa, the specimens will be irradiated to an exposure about 50% greater to accommodate uncertainties in the exposure estimates. A decision to conduct further test reactor irradiations beyond those noted above will be based on the results of the initial testing.

As currently required by 10 CFR 50, Appendix H, and for reasons of prudence, the NGNP will also need to incorporate a surveillance program. The specific design of the surveillance program, including the specimen complement, will be based on the results obtained from the test program discussed above, but will likely include, as a minimum, tensile, Charpy impact, fracture toughness, and creep specimens. Because the NGNP will be a demonstration reactor, the surveillance program will be more extensive than required by the regulatory authority, such that it could serve as a test bed for irradiation experiments of more advanced materials that may be developed as NGNP operations progress.

4.2.4 Emissivity Considerations

Under normal, and especially, abnormal operating conditions, the RPV would need to radiate heat to its external environment. Should the normal operating temperature of the RPV be such that a 9Cr steel is required, it will be necessary to have a stable, high-emissivity surface on the pressure vessel material such as, 9Cr-1MoVNb and variants, at elevated temperatures (estimated to be 562°C under abnormal conditions). While the emissivities of steel can be increased by the formation of an oxide film,[39] as shown in Figure 32, the conditions under which this film can be created and the stability of this film in air (including the effect of humidity) at operating temperature needs to be established. Work would need to be performed to establish a technology of the application of an emissivity coating and an industrial partner will be needed to provide for scaling of the materials and methods that have proven to be viable.

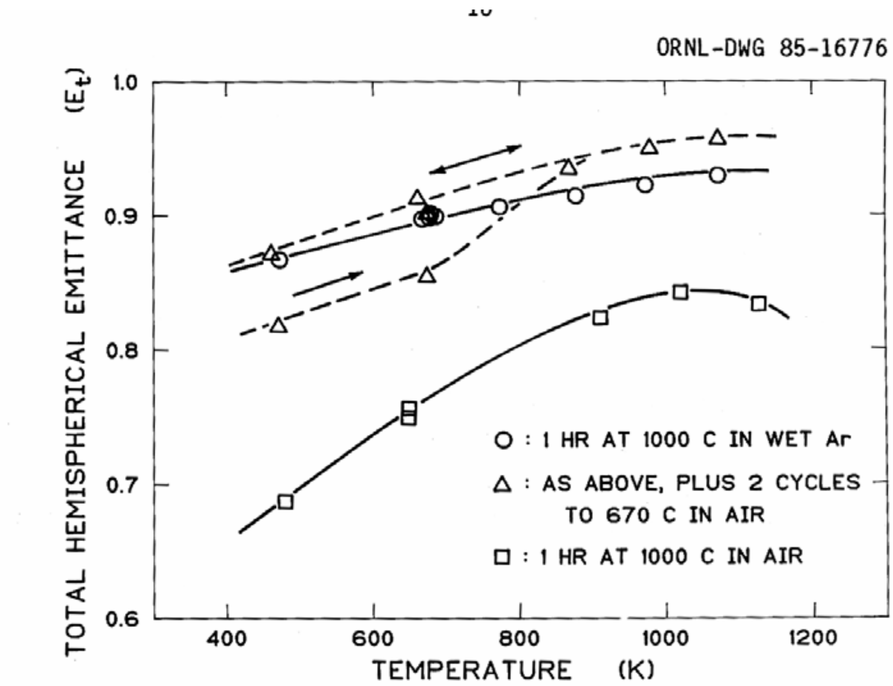


Figure 32. Effect of environment and temperature on the emissivity of type 304 SS.

4.2.5 Key Issues for RPV Materials Selection & Application

Currently, Argonne National Laboratory is conducting an independent assessment of RPV materials selection and requirements; unfortunately, this work was not available in time for inclusion in this report. Once received, the ANL input will be incorporated in the evaluation and decision process with the information provided by industry in response to the INL/DOE request for proposals for pre-conceptual designs for the NGNP.

The key issues to date relevant to selection of reactor design and associated material selections for the primary pressure boundary are discussed below.

4.2.5.1 Low-Temperature RPV Design – PBMR Design. The PBMR design will utilize active cooling, and hence, incorporates the selection of SA508/533. If the active cooling system fails, there is no safety risk associated with the failure of the cooling system; however, there is an economic risk that the RPV may sustain excessive temperatures and stresses for future service. For this reason, PBMR seeks to revisit Code Case N-499-2 to determine if the temperatures and times permitted for service at or above 371OC can be extended.

Additional data required for extending N-499 will include creep testing at relevant times and temperatures. PBMR appears to have developed a test plan to support the extension of time and temperature desired for their design. No DOE activities at the national laboratories exist that address this issue currently. The DOE-ASME Collaboration Agreement did not include this activity since the DOE-ASME Tasks are restricted from conducting testing. This activity should be supported by the national laboratories in support of the PBMR or similar design concepts that will utilize SA508/533 for the RPV and related primary pressure boundary materials.

4.2.5.2 High-Temperature RPV Designs – AREVA Design. The AREVA design does not utilize active cooling, and hence, currently is evaluating the use of Gr91 steel. INL has verified that the AREVA design has no probable economic risk associated with such an event if Grade 91 steel is used for the RPV. ASME code exists to support the AREVA design with several possible restrictions.

1. AREVA seeks to have code approval to raise the temperature at which creep is deemed insignificant. This has no impact on safety, and impacts the economics of the reactor design in terms of the need of a RPV monitoring program. This desire is reflected in the DOE-ASME tasks

2. .Thick sections of Gr91 steel require attention on several levels. Currently, ASME permits thick sections of Gr91 steel provided that no adverse effects exist with respect to mechanical properties and chemistry variations through the thickness. This must be demonstrated to be true, or modifications made to appropriate time, temperature, and stress allowables in ASME code, which must be incorporated in the AREVA design. The ability to fabricate and weld thick sections of Gr91 steel (200-300mm thick), including transportation and post-weld heat-treatment issues related to the same, are a concern.

3. Concerns start with the selection of a welding process, e.g. submerged arc welding (SAW), gas tungsten arc welding (GTA), narrow groove GTA welding. The selection of welding parameters and process control are also important as the temperature history of the base, weld, and HAZ are critical in the evolution of the microstructure and subsequent mechanical properties of the weldment.

4. Generally speaking, the effects of any significant heating of 9Cr steel beyond its normal tempering heat treatment should be established and related to properties and projected service performance. In addition to the welding process/parameter selection another important issue will be the effects of maintaining welding preheat and interpass temperatures on thick-section components. Components of 9Cr steel are typically preheated for welding to at least 200°C, and temperatures are maintained at 200-290°C during welding. After welding, components may also be subjected to a hydrogen bake treatment of extended time at 300-350°C prior to code-required post weld heat treatment. Large, thick-section components could be held at these relatively low temperatures for days. The effects of such treatments on microstructures and properties are unknown but potentially significant. They should be characterized.

5. Another similar issue is the handling of hot components after welding is completed. After welding and possible hydrogen bake cycles, thick-section components will retain a substantial heat content. Metallurgically, it is desirable to cool welded components to room temperature prior to post weld heat treatments. In practice, it is desirable to retain as much of this heat as possible by immediately proceeding to post weld heat treatment operations. This practice, however, will have potentially significant consequences on the phase transformation behavior and properties of the weld deposits. This aspect of the heat treating of welded components should also be investigated.

6. The most important structural performance issue for welded components of 9Cr steels is their susceptibility to developing weakened microstructures in their base metal heat affected zones. This situation directly leads to reduced creep strength of weldments relative to base metal properties. This is termed Type IV failure. Thick-section components will have greater microstructural and chemical through-thickness variability than thin-section metal. The through-thickness variability of Type IV behavior should be characterized and related to welding parameters and the various thermal treatments required for fabrication of thick-section components.

7. The lead time required for acquisition of a large RPV is an issue. Feedback from PMBR/Westinghouse indicates that if an order were placed today for a 508/533 RPV, a minimum of 5 years would be required before delivery. The ability to obtain a forging of the size required for a Gr91 steel RPV is a challenging issue as such as fabrication of a vessel of this size has yet to be demonstrated. Fabrication with thick plates is an option. More information is required to determine the availability of Gr91 forgings or plates required; the timeline will likely be of equal or greater length than for the equivalent LWR RPV steel.

8. It will be necessary to evaluate the irradiation response of Gr91 steel for RPV applications to enable licensing of the NGNP reactor vessel. This will require sufficient irradiations to be conducted at fluxes and temperatures considered to be representative of irradiation conditions that will exist in the NGNP. These irradiations cannot be performed in the core of test reactors, such as HFIR and ATR, because the flux is much too high and irradiation volume too low for the sizes of specimens that will be required. A minimum program of 5-year duration, in a low-flux irradiation facility, will be required to develop initial licensing data for Gr91 steel, with supplementary confirmatory irradiations thereafter for final licensing approval.

   AREVA has recognized the difficulty in welding of thick sections of Gr91 steel; they have invested extensively in a development program to resolve issues in thick section weldments of Gr91 steel. While little information is available in the literature, AREVA claims to be capable of successfully welding and heat-treating these weldments. Other than the limited fossil energy program experience, no other known development efforts have been made in this area. A great deal of experience in the field exists related to Gr91 failures, and should be reviewed to determine to what extent failures are related to improper quality control of processes, improper operating conditions, etc. A confirmatory development program should be undertaken, preferably in conjunction with AREVA, to determine if AREVA's success in this area can be extended to the size, configuration, and possible in-field welding/heat-treating processes for the NGNP. The program should include aspects summarized above regarding welding processes and parameters, as well as a testing program to determine short term (tensile), long term (creep), and microstructure stability (aging and exposure tests) to assess the feasibility of fabricating, welding, and heat treating Gr91 steel for thick section NGNP RPV, CV, and IHX containment vessel applications.

4.2.6 Recommendations and Conclusions[40,41]

There will be a very limited or nonexistent NGNP materials qualification program in FY2007 associated with the RPV. However, the key issues regarding excursion events for SA508/533 in Code Case N-499-2 and thickness effects / fabrication / welding / heat-treatment of Gr91 on RPV designs and selection of designs for the NGNP have been identified. An additional consideration should be confirmation of adequate emissivity of the RPV for passive cooling requirements. These issues should be taken into account and consideration in the evaluation of proposals received from vendors for pre-conceptual designs for the NGNP.

4.3 High Temperature Metallic Alloys

4.3.1 Introduction

The primary candidate materials under consideration for the intermediate heat exchanger (IHX) for the Next Generation Nuclear Plant (NGNP) are Alloy 617 and Alloy 230. The proposed operating conditions for the NGNP present unprecedented challenges for structural materials. In the Very High Temperature Reactor (VHTR), the structural alloys must withstand temperatures up to 950°C in a helium environment for up to 60 years. To qualify candidate materials for the reactor design and construction, sufficient data on mechanical properties must be provided for material selection, model development, and codification under the American Society of Mechanical Engineers Boiler and Pressure Vessel (ASME B&PV) Codes and Standards. Most conventional materials (e.g. ferritic steels) do not possess sufficient strength at such high temperatures. Alloy 617 and Alloy 230 are nickel-base superalloys developed for high temperature service with a good combination of high-temperature strength and oxidation resistance, as well as excellent resistance to a wide range of corrosive environments.

Alloy 617 was introduced in the early 1970's and a significant amount of mechanical property data has been generated over the past several decades[42]. However, reviews of the collected data have revealed that many of the datasets are incomplete, missing important information such as material pedigree and original test data curves that are necessary for studies and modeling required for the Gen IV nuclear reactor development. Data gaps also exist, especially in the very high temperature range required for VHTR applications. Furthermore, significant data scatter has been observed in mechanical properties[43]. The data scatter would require large safety and knockdown factors, leaving very limited allowable strengths for reactor component design. Compared to Alloy 617, Alloy 230 is a relatively new alloy with the possibility to outperform Alloy 617 in several properties (e.g., yield strength, tensile strength) desired for the Gen IV reactor working conditions. However, because Alloy 230 is a relatively new, it lacks an established database, especially relating to performance in a helium environment similar to that expected in the VHTR.

Another viable option may be Hastelloy XR, also a nickel-based superalloy. Hastelloy XR has been accepted by the Japanese nuclear code and used in the Japanese High Temperature Test Reactor (HTTR) at temperatures of 850-950°C for a very short time, but the American database for Hastelloy XR is sparse. If the U.S. can reach an agreement with Japan to acquire their database, it could expedite an ASME code case or possibly acceptance by ASME Subsection NH (elevated temperature design for Class 1 nuclear components). However, until the Hastelloy XR database can be secured, Alloy 617 and Alloy 230 remain the primary candidates.

4.3.2 Material Properties

4.3.2.1 Chemical Composition. The chemical composition of Alloy 617 and Alloy 230 are listed in Table 14. Both materials are austenitic alloys based on nickel and chromium which provide a high resistance to a variety of reducing and oxidizing environments. Alloying additions of cobalt, molybdenum, and tungsten enhance solution strengthening, while oxidation and carburization resistance is further increased by chromium and aluminum additions. Depending upon the thermal history, the small amount of carbon in the alloys can form finely distributed carbide particles throughout the grains and contribute to high temperature strength.

Table 14. Chemical composition limits for Alloy 617 and Alloy 230.[44]

		Ni	Cr	Co	Mo	Fe	Mn	Al	C	Cu	Si	S	Ti	B
Alloy 617	Min	44.5	20.0	10.0	8.0	-	-	0.8	0.05	-	-	-	-	-
	Max	-	24.0	15.0	10.0	3.0	1.0	1.5	0.15	0.5	1.0	0.015	0.6	0.006
Alloy 230	Min	Bal	20.0	-	1.0	-	0.3	0.2	0.05	13.0	0.25	-	0.005	-
	Max		24.0	5.0	3.0	3.0	1.0	0.5	0.15	15.0	0.75	0.015	0.05	0.015

4.3.2.2 Limited Mechanical Properties Database. There is a limited amount of mechanical property data for Alloy 617 and Alloy 230, specifically long term creep testing and creep-fatigue testing. The majority of the data for Alloy 617 was generated in the mid-1980s when a draft Code Case for the design of nuclear components was developed for the High-Temperature Gas-Cooled Reactor (HTGR) Program[45]. The draft case covered temperatures up to 950°C, but was later dropped from consideration because the HTGR program was terminated. Other countries continued to pursue Alloy 617 for the HTGR concept, so the majority of long-term data on Alloy 617 was generated abroad (primarily in Germany[46,47,48] and Japan[49,50,51]). ORNL is currently collecting the data for inclusion in a Gen IV Materials Handbook[52]. A summary of major known sources of creep data and test conditions is shown in Table 15. The German database is the most extensive, including tests that exceed test durations of 20,000 hours, and a limited number in excess of 50,000 hours. However, only a fraction of this data is available through open literature. ORNL is in the process of obtaining a larger portion of the German database for use in the Materials Handbook. Unfortunately, raw creep data from the supplier of Alloy 617, Special Metals, no longer exists. The data only exist in processed and tabulated form, which has significant ramifications in terms of generating test data to support constitutive modeling efforts, apparent errors in tabulating or processing some data, and possibly extrapolation of test data.

Table 15. Major sources for creep data for Inconel 617.

Source	# of Tests	Temp. Range	Max Duration	Min Stress
ORNL-HTGR[53]	51	593 - 871°C	34,231 h	21 MPa
GE-HTGR[54]	36	750 - 1100°C	28,920 h	9.6 MPa
Huntington Alloy[52]	249	593 – 1093°C	40,126 h	6.2 MPa
German HTGR[46]	294	800 – 1000°C	~20,000 h	8.2 MPa

Inspection of Table 15 shows that there are no test data available that approach within an order of magnitude of the 60 year life of the NGNP (or 525,000 hours). Consequently, extrapolation of creep data will be required to estimate operating stresses. Extreme caution should be exercised when extrapolating this database to predict creep properties at 100,000 hours or longer. Although the current database for Alloy 617 can be adequately described by a power law equation, it is possible that the active deformation mechanism changes from dislocation glide-climb to intracranial diffusion, dramatically affect the effective creep rate and high-temperature performance of Alloy 617[55].

Compared to the draft code case developed for Alloy 617, the mechanical property database for Alloy 230 is much less extensive and has similar issues with respect to extrapolation of data for NGNP needs. Haynes has provided a large number of raw creep curves for Alloy 230 to ORNL for use in ASME codification requirements. To date, a total of 299 creep and tensile raw data curves for Alloy 230 have been collected, which include creep times up to 28,391 hours and temperatures up to 1149°C. Although this will significantly reduce testing needs for qualifying Alloy 230, substantial effort will be required to assemble, analyze, and summarize the data. In fact, only a cursory inspection and cataloging of the creep database has been accomplished to date.

4.3.2.3 Tensile, Creep, and Creep-Fatigue Properties. The tensile properties of Alloy 617 and Alloy 230 are shown in Figure 33. The 0.2% yield strength and ultimate tensile strength of Alloy 230 exceeds that of Alloy 617 across all temperatures shown. At the upper operating limit of the NGNP (950°C), the ultimate tensile strengths are nearly identical at ~175 MPa and the yield strength of both alloys are roughly 100 MPa. The 1% strain and rupture creep properties of Alloy 617 are represented by a plot of the Larson Miller parameters in Figure 34. As stated earlier, note that the majority of the data lie at times far below that of interest to the NGNP. Also note that the stress for the Larson Miller parameter for 1% strain in 100,000 hours (25x103) and rupture in 100,000 hours (24x103) are both around 4 MPa, which is a very low design stress. As mentioned earlier, the creep data for Alloy 230 has not yet been processed.

The creep-fatigue properties of Alloy 617 are shown in Figure 35. It is generally recognized that the current linear damage accumulation rule for creep-fatigue has significant shortcomings, particularly at higher temperatures and longer times. Various improvements, such as those based on ductility exhaustion and damage rate concepts, have been proposed, but none have been backed by sufficient research to allow their adoption as a replacement for the linear damage accumulation rule in ASME Section III Subsection NH. Two tasks funded within the DOE-ASME Collaboration Agreement discussed in Section 1.7 are in progress. One of these tasks will investigate the issue of creep-fatigue in an effort to improve upon the current understanding and procedures.

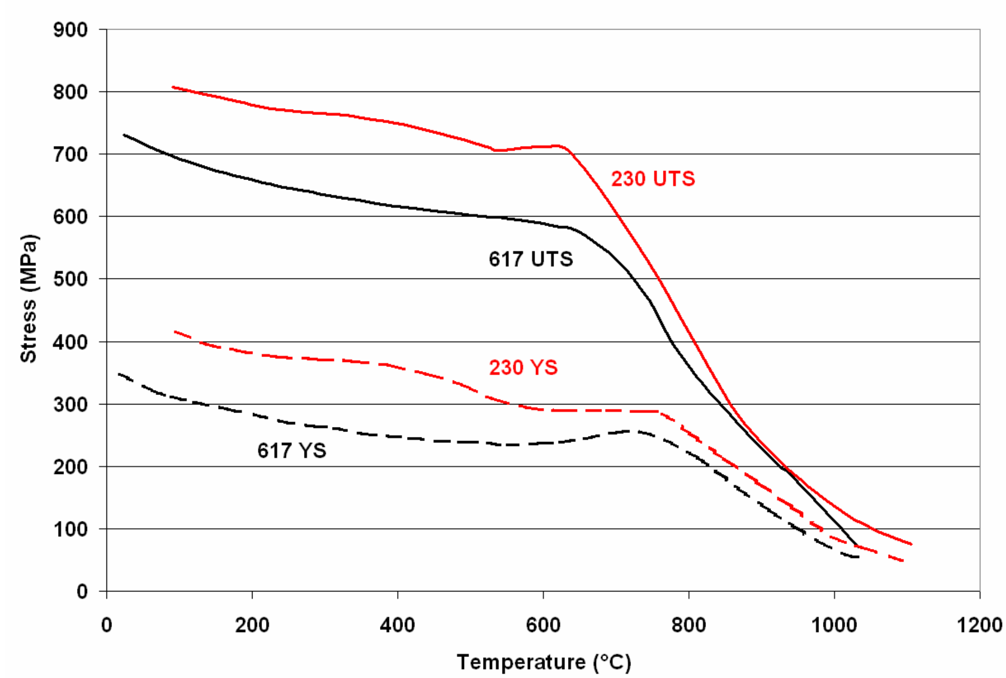


Figure 33. Tensile properties of Alloy 617 and Alloy 230[44].

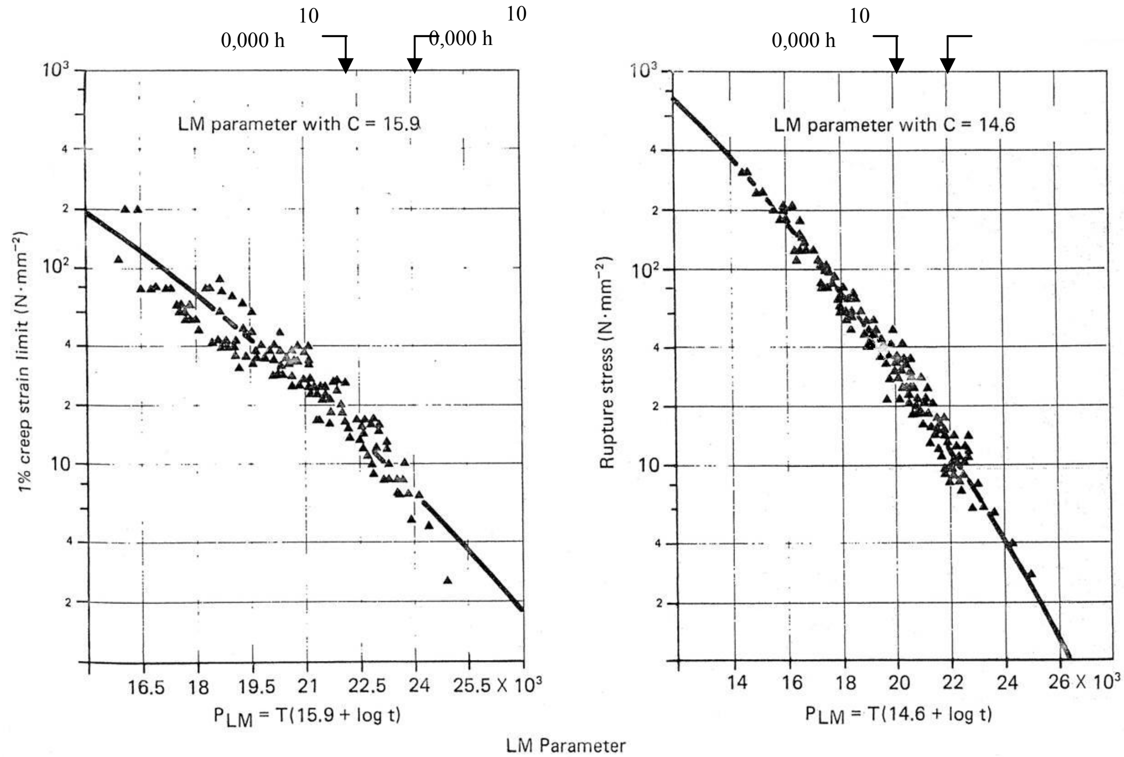


Figure 34. Larson Miller Parameter for Alloy 617[46].

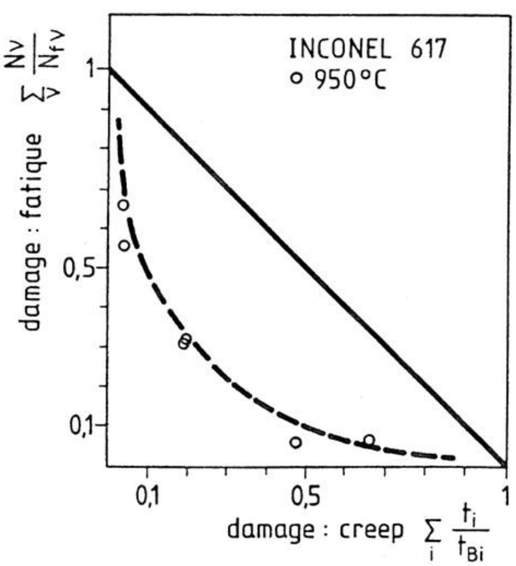


Figure 35 Creep fatigue properties of Alloy 617[56].

4.3.3 Aging Effects

During long-term service at high temperatures, metallic materials inevitably undergo aging processes which result in microstructural evolution and changes in mechanical properties. In order to select structural materials and develop design guidelines for the Gen IV reactors, it is necessary to understand the effect of aging on the mechanical properties of candidate materials.

Tensile data for aged material is important in the development of factors that apply in ASME Subsection NH to account for changes in the short-time properties due to long-time service. Similar to the observations of microstructure and hardness behavior, the effect of aging on tensile properties is variable from one heat to another and both the kinetics and magnitude of the changes reflect this variability. For a single heat of Alloy 617 aged at several temperatures (600 to 750°C) for times up to 12,000 hours, it was found that the yield strength, ultimate tensile strength, and hardness increased with aging time[57]. A rapid loss in ductility occurred with aging at 704 and 760°C. However in a separate study, a different heat of Alloy 617 demonstrated a significant loss in ultimate tensile strength when aged at 871°C[58]. Further research is needed to establish the lower limit for strength and ductility at temperatures of 850°C and above.

The aging effects observed in regard to short-time properties such as hardness, tensile, and impact properties indicated significant changes in microstructure during the first 10 to 100 hours when the exposure temperatures exceeded 650°C. Creep testing, however, easily consumes times beyond 100 hours, so it might be expected that aging effects would become insignificant in long-time creep testing above 650°C. Only the early stages of primary creep would be affected by the simultaneous processes of creep and precipitation associated with aging. Kihara et al.[49] and Schneider et al.[59] studied the influence of aging on the primary creep curve. Both noticed that the primary creep behavior of annealed material was somewhat complicated in the temperature range of 850 to 1,000°C. A very small period of rapidly decreasing creep rate was followed by a period of increasing rate and then a gradual approach to a constant rate which eventually gave way to the tertiary creep stage. Investigators[59,60] found that aging prior to creep eliminated the low creep rate at the start of the test, hence the aged material creeps at a high rate at the start of the test. In the study of Kihara et al., the 1,000 hours aged and virgin sample eventually achieved the same creep rate.

Most of the studies on aging effects in creep have been performed at relatively high stress and testing has generally been limited to 2,000 hours or less. The effect of aging on the low-stress creep and rupture behavior remains an unknown, but it is considered likely that aging effects will become insignificant over the life of the test. However, if aging occurs under a small tensile load, it is possible that the microstructure could be altered to affect the creep properties. Several researchers[49] have observed that carbides re-distributed during the creep process to populate boundaries normal to the tensile stress, creating a local stress concentration and source for micro-cracking. If this re-distribution were to occur during the aging process, the creep properties could be significantly affected.

4.3.4 Product Form / Grain Size Effects

The design of the intermediate heat exchanger (IHX) will have a significant impact on the product form and grain size of the selected material. Depending on whether a conventional or compact IHX design is chosen, the material thickness could vary from 0.2mm (8 mils) to 2mm (80mils). Assuming that a minimum of 10 grains would be desired across the thickness, grain sizes would vary from maximum sizes of about 20 μm (ASTM G.S. 8) to about 200μm (ASTM G.S. of 1.5). Typically, a grain size of ASTM Number 6 (45μm) or coarser is preferred, but it has been shown that creep strength increases with increasing grain size so microstructures with 100 or 200 μm grain size are often produced. A tradeoff exists, however, when fatigue is an issue since finer grain sizes are preferred for fatigue resistance.

It remains to be determined if an IHX made of large grain size Alloy 617 can be designed to meet performance requirements and be economical. The current draft code case for Alloy 617 only supports grain sizes from approximately 127-360 μm (ASTM G.S. 0 to 3). Therefore, selection of smaller grain size product forms for a compact IHX will require additional testing and qualification. At this point, Alloy 617 and Alloy 230 would be on equal footing since thin section (small grain size) alternatives will essentially require codification of a new material with an *extensive* testing program and much longer time period to obtain ASME code approval. Acquisition of the German database for large grain sizes of Alloy 617 could significantly help with qualification of Alloy 617, especially if existing studies of environmental effects for large grain 617 apply directly to smaller grain product forms. Unfortunately, acquisition of the German database remains stalled due to legal issues. Conversely, Alloy 230 in thick section (large grain size) may not take as long to qualify since it has been possible to acquire creep data and raw creep curves from Haynes International.

4.3.5 Environmental Effects

The interactions between structural materials in high purity helium atmospheres associated gas cooled reactors have been the subject of numerous investigations (see, for example Kimball[61]). The results of these studies conducted by various organizations in USA, Germany, England, Norway, Japan, and other places have demonstrated the importance of small changes in impurity levels, high temperatures and high gas flow rates. Metallic materials can be carburized or decarburized, and oxidized internal or at the surface. These corrosion reactions, depending on the rate, can affect long term mechanical properties such as fracture toughness.

The simulated advanced HTGR helium chemistry used in various test programs are shown in Table 16. Because of the low partial pressures of the impurities, the oxidation/carburization potentials at the metallic surface of a gas mixture is established by the kinetics of the individual impurity catalyzed reactions at the surface. As shown, the main impurities are H_2, H_2O, CO and CH_4. The hot graphite core is considered as reacting with all free O_2 and much of the CO_2 to form CO, and with H_2O to form CO and H_2. In addition, in cooler regions of the core, H_2 reacts with the graphite via radiolysis to produce CH_4. Because of the change in surface temperatures around the reactor, and associated changes in reaction mechanisms and rates of reaction on bare metal versus on scaled surfaces, reaction rates and order of reactions are important.

Table 16. Composition helium environments (advanced HTGR) used in past tests

Program	H_2 (µatm)	H_2O (µatm)	CO (µatm)	CO_2 (µatm)	CH_4 (µatm)	N_2 (µatm)	He (atm absolute)
NPH/HHT	500	1.5	40		50	5–10	2
PNP	500	1.5	15		20	<5	2
AGCNR	400	2	40	0.2	20	<20	2

NPH: Nuclear process heat

HHT: High temperature helium turbine systems

PNP: Prototype Nuclear Process Heat

AGCRNR: Advanced Gas Cooled Nuclear Reactor

The overall stability of the proposed helium environment must be evaluated in order to ensure that testing proposed in various sections of the program are performed in environments that have consistent chemical potentials. In addition, the corrosion of metals and nonmetals must be evaluated to establish baseline data where it does not exist. These tests will be performed at temperatures to include at least

50°C above the proposed operating temperature. The corrosion testing is above that proposed in various sections of the program, which is focused .

At high temperature, separation of the effect of temperature from that of the gas reactor helium environment requires an extensive evaluation of materials performance in impure helium versus some standard environment, which has been chosen, in some investigations, as air. In general, high-temperature exposure results in some form of structural changes in a material. Microstructural changes include recrystallization; grain growth; changes in morphology such as loss of coherency or interfacial strength, dissolution of a phase, or precipitate growth; and precipitation of new phases such as carbides[62]. In addition to these microstructural changes, because of the low oxidation potential of the helium environment, any oxide that forms on the surface of the material may not be very protective and result in internal oxidation, and/or carburization or decarburization.

If carburization occurs, the ductility would decrease; if decarburization occurs, the strength would decrease; and if internal oxidation occurs, the potential for crack initiation increases with the associated decrease in ductility and toughness. Also it must be remembered that long exposures will result in thermal aging effects, which are not fully characterized at this time. However, in one study, it has been reported that there were no major differences (differences within the data scatter) in creep strength and fatigue properties for materials tested in impure helium versus air. The rupture ductility of Inconel 617 in impure helium was reported as lower than those for air. Also, as shown in Figure 36, there is a significant effect of sample diameter on rupture properties[63] and it must be noted that the design curves for creep-fatigue of Inconel 617 at 950°C for air are more restrictive than for helium[64]. The sample diameter effect (cross-sectional thickness) results from the greater contribution of the microstructure of surface and near-surface layers to the over performance of the specimen. Consequently, the effect of the environment plays a greater role on the performance of thin cross-section components, such as heat exchangers. Even in situations where the oxygen potential is high enough to form a protective oxide scale on the metal surface, vaporization of the oxide may occur above 800°C. The rate of vaporization is dependent on the composition of the oxide, temperature, and helium flow rate[65]. For the materials under consideration, Alloy 617 and Alloy 230, chromia is the most volatile oxide and alumina the most stable oxide. Because the level of aluminum in Alloy 617 is much higher than that in Alloy 230, Alloy 617 is more likely than Alloy 230 to form an external alumina layer rather than internal alumina oxides under appropriate oxygen/temperature environments (formation of an external alumina layer is extremely difficult for these alloys). However, if the external layer is not continuous, it is unlikely to be protective.

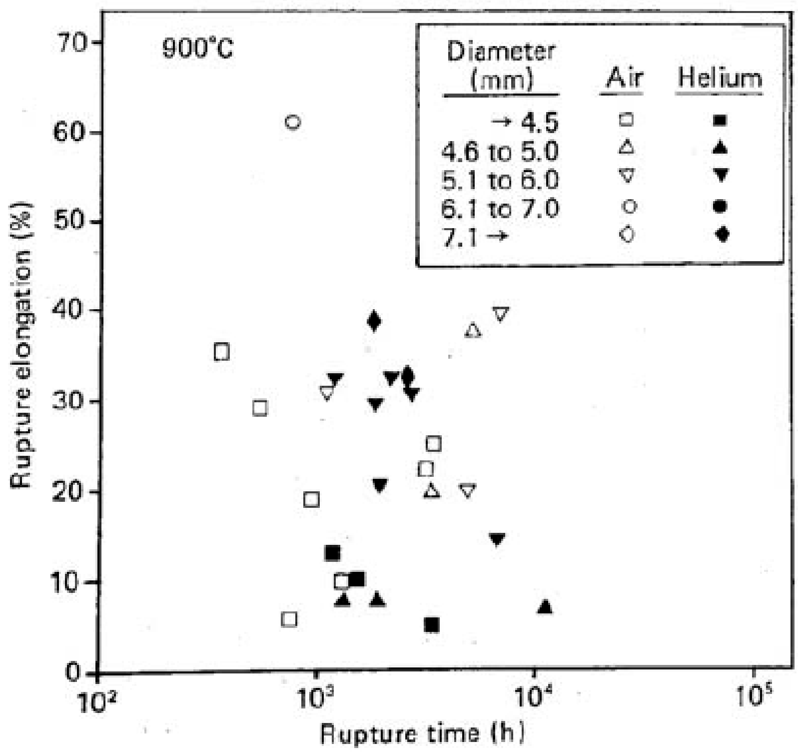


Figure 36. Effect of specimen size on rupture behavior. Data from Schubert[66]

Overall, there is a significant amount of gas/metal interaction data for candidate alloys but there is a lack of consistency between studies. This inconsistency, is also apparent in the mechanical properties data. These differences are most likely due to differences in the targeted impurity levels and in the actually obtained impurity levels (due to depletion). To address these differences, experimental systems using controlled additions of impurities must be operated such that the output gas chemistry is within 10% of the input gas chemistry and overall gas chemistry is within 10% of the targeted gas chemistry.

In order to ensure operation in a protective regime, the temperature/helium with controlled impurity levels space must be fully explored especially for materials that have seen little exposure in helium. Data must be generated in the temperature range of 500 to 1000^0C in helium environments [(μatm) H_2 400 ± 100; H_2O 2 ± 1; CO 40 ± 10; CO_2 0.2 ± 0.15; CH_4 20 ± 5; N_2 < 10] for times to 10,000 h and longer. Tests, such as creep-fatigue, that allow for repeated exposure of specimens of candidate material to the environment are especially applicable.

4.3.6 Irradiation Effects

Both Alloy 617 and Alloy 230 are negatively affected by exposure to neutron irradiation. The strength of the materials increase, but they become embrittled and swell over time. Fortunately, these materials are being considered for an application that is located outside of the primary core, away from the majority of neutron irradiation. One possible issue, however, is that elements on the surface of the superalloys could erode and pass into the coolant stream at elevated temperatures. For example, Alloy 617 contains a significant amount of cobalt (up to 15% by weight) which is highly active in a neutron environment. This could pollute the coolant stream with radioactive particles and possibly pose a safety hazard in the event of a leak.

4.3.7 Relevance to IHX Design

Several versions of the VHTR include an intermediate heat exchanger (IHX) that serves to transfer energy to a secondary plant dedicated to the production of hydrogen. The specific type of IHX has yet to be selected, (e.g. tube/shell, plate/fin, or a micro-channel type IHX). Argonne National Laboratory is working to provide another independent assessment of viable options that will be detailed in a recent submission of vendor proposals. All aspects of possible IHX concepts must be identified, including materials, design, fabrication, examination, testing, overpressure protection and in-service inspections that are used in the construction and operation of the IHX. Furthermore, joining and welding practices need to be defined for each material combination. In particular, the reliability of diffusion bonding is questionable when exposed to thermal cycling.

Although materials for the IHX need to be qualified by the ASME Boiler and Pressure Vessel code, the heat exchanger designs do not lend themselves to ASME design procedures because the structure has little or no resemblance to classical pressure vessels. Furthermore, the extrapolation of creep and creep-fatigue properties to 100,000's of hours does not seem appropriate without further testing at extremely low stress levels. Given the further uncertainties of aging, grain size, and environmental effects, Alloy 617 and Alloy 230 do not have the requirements for 60 years of service; rather the IHX will probably be replaced on a rotating basis of 5 to 15 years.

Currently, the PBMR design does not include an IHX; instead, the direct cycle path was selected, eliminating R&D and qualification efforts needed for an IHX. However, the trade-off is the need to deal with possible contamination issue of balance of plant due to the direct cycle.

AREVA has selected to include the IHX in their NGNP design and has made significant investments in R&D in the area of materials and design for the IHX. There are no expectations that the IHX will be the primary pressure boundary; the IHX will be housed inside a containment vessel. The containment vessel will be at a much lower temperature, permitting use of Gr91 steel. However, a safety valve will be required between the secondary side of the IHX that provides process heat to the hydrogen plant (to be determined). AREVA believes that the combination of the containment vessel and the safety valve will permit the core of the IHX to be classified as a Class 2 or 3 component. The implications of both of these approaches, as well as supporting ASME code for materials, fabrication, joining, inspection, etc. require substantial efforts. A task within the DOE-ASME Collaboration Agreement is planned to initiate these activities. One major obstacle is the interest and willingness for the vendors of the various IHX designs to participate at a level that provides substantial value to the development of ASME code; at this time, vendors have not been forthcoming with information due to the competitive stage that the IHX and NGNP program exist.

4.3.8 Material Comparison

Alloy 617 and Alloy 230 are the two primary candidate materials for construction of the IHX in the NGNP. The following lists pros and cons of each material selection with regard to material properties, available data, and qualification through the ASME B&PV Code.

4.3.8.1 Alloy 617.

Pros:

An ASME draft code case exists for Alloy 617 (large grain material).

A large amount of tensile and creep data exist for Alloy 617 as a result of the HTGR efforts in the 1970s and 1980s. The majority of this data belongs to Germany and Japan.

Certain conclusions from existing database for large grained material may be extended to small grained material (environmental effects, aging, etc).

A large amount of environmental effects data in helium, representative of high temperature reactor coolants, has been developed for 617.

Cons:

Huntington Alloys original creep curves for Alloy 617 are missing. Only tabulated data remain.

Legal issues are impeding procurement of German the database.

Very little data available at elevated temperatures for extended time periods.

Very little data available for small grained material.

Higher cobalt content may pollute coolant stream.

4.3.8.2 Alloy 230.

Pros:

Slightly higher tensile properties at lower temperatures, almost equal at 950°C.

Almost 300 tensile and creep curves are readily available from manufacturer at elevated temperature; they require in depth analyses.

Low cobalt content.

Cons:

Relatively new material – no history of Section III elevated temperature design code qualification.

Very little data available at elevated temperatures for extended time periods.

Very little data available for small grained material.

Very little environmental effects data in helium, representative of high temperature reactor coolants, has been developed for 230.

4.4 High Temperature Design Methodology

4.4.1 Introduction

This section describes those task elements and activities required to establish the high-temperature structural design technology necessary to support the codification and utilization of structural materials in the NGNP. Subsection NH of Section III of the ASME Boiler and Pressure Vessel Code governs the design and construction of elevated-temperature, Class 1 nuclear components. Similarly, Subsection NB of Section III governs the design of Class 1 nuclear components that are not at elevated temperatures. Subsections NC, ND, and NG govern Class 2 and Class 3 nuclear components and core support structures, respectively. Various code cases also exist that have been developed to address specific material or component needs, e.g. N-499 and N-201. The tasks and activities are directly linked to and justified by the need to address historical regulatory and licensing issues raised by the NRC during licensing discussions associated with the Clinch River Breeder Reactor (CRBR), the Power Reactor Innovative Small Module (PRISM) Liquid-Metal Reactor, and several evaluations conducted for the NRC to determine the adequacy of ASME Subsection NH for elevated temperature design. Performance and technical requirements for the NGNP to deliver both electricity and hydrogen using a gas-gas intermediate heat exchanger (IHX) necessitate the need for research and development of materials and design methods to support selection of the type of IHX, material selection, as well as supporting material and design criteria code needs. Additional justification is provided by ongoing interactions and feedback received from industrial stakeholders who have shown significant interest and activity in obtaining license approval for the design, construction, and operation of the NGNP in Idaho. As such, this section will discuss the data and models required by ASME Code groups to formulate time-dependent and independent failure criteria and rules that will assure adequate life for components fabricated from materials identified for the NGNP. The generation and development of design data needed to quantify the failure criteria and design rules (e.g., uniaxial creep-rupture data, are not included in this task); this data is specific to the High Temperature Materials and Pressure Vessel sections – although the activities are integrally linked. Experimentally based constitutive models that are the foundation of inelastic design analyses specifically required by the ASME code are discussed in this section, and appropriate simplified design procedures provided as necessary for use in conceptual and preliminary design phases, as well as possible final design applications. All of the key parts of a high-temperature design methodology - failure criteria, inelastic design analysis methods, and alternate & simplified design procedures require experimental verified by an assortment of representative structural tests. Efforts will be focused (based on budget limitations) on materials for the reactor pressure vessel (508/533 steel and Gr91 steel) and the intermediate heat exchanger (Alloy 617, 230, X/XR). The initial results and design methodology framework, while providing for use of two viable materials for the NGNP, will also serve as a springboard for adding additional materials to the methodology in later years.

Time-dependent failure modes and time- and rate-dependent deformation response to time-varying thermal and mechanical loadings characterize the design of NGNP metallic components operating at high-temperatures. The threshold defining high-temperature operation is 371°C (700°F) for the ferritic steels currently permitted for construction of Class 1 nuclear components by the ASME Boiler and Pressure Vessel Code, Section III, Subsection NH, and 427°C (800°F) for the permitted austenitic alloys. The primary roles of the High-Temperature Design Methodology (HTDM) task, which is an integral and inseparable part of the overall NGNP materials program, are as follows:

1. The HTDM task will provide the data[c] and models required by ASME Code groups to formulate and verify time-dependent failure criteria that will assure adequate life for components fabricated from the selected NGNP materials.

2. The task will provide the experimentally-based constitutive models that are the foundation of the inelastic design analyses specifically required by Subsection NH of Section III of the ASME Boiler and Pressure Vessel Code; specifically, the draft code case for Alloy 617 is the foundation for which components made with any of the candidate nickel based alloys, e.g. Alloy 617, 230, or Hastelloy X/XR, must be designed for elevated-temperature nuclear components.

3. The HTDM task will review and provide appropriate alternate and simplified design procedures required for the conceptual and preliminary phases of design and will be useful for many less-critical regions in final design.

4. The HTDM task will assist in the development of a "Guideline for Inelastic Analysis and Life Prediction"; this will replace or update the NE guideline F9-5T that was developed specifically for high temperature design of LMFBR components. The guideline will be effective in obtaining acceptance from the NRC for licensing of the NGNP.

5. The HTDM task will assist in identifying existing NRC concerns, and any remaining or unresolved issues following a request from DOE to obtain a license from the NRC to construct and operate the NGNP; subsequently, this task will include the development and execution of R&D plans to resolve such concerns.

6. Overall this task is a key part of the codification, utilization, and NRC licensing of the selected NGNP nuclear structural materials and components that will operate at elevate temperature. The task can be divided into 7 subtasks:

    a. alternative primary load design methods

    b. simplified methods and criteria,

    c. failure models for design criteria,

    d. inelastic design analysis methods,

    e. confirmatory structural tests and analyses,

    f. safety / reliability assessments, and

    g. resolution of identified shortcomings, issues, and regulatory concerns.

An overview of the technical issues by subtask follows, with required time estimates. The impact and risk of various activities and associated scope on the NGNP schedule is then discussed. Finally, the

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c A clear distinction should be made between the development of criteria (e.g., the damage accumulation rule and multiaxial strength criterion needed to guard against creep rupture) and the design data needed to quantify the criteria (e.g., uniaxial creep-rupture data). The former are largely the purview of the High-Temperature Structural Design Technology task; the latter are the responsibility of the design data generation tasks.

information is summarized with respect to the timeline requirements for ASME codification and NRC licensing. The discussion pertains to how the HTDM subtask relate to associated risks in meeting the Option 2 Schedule for the NGNP as presented in the NGNP Plant – Preliminary Project management Plan [INL/EXT-05-00952, Rev. 1, March 2006].

4.4.2 Alternative Primary Load Design Methods

Currently, the analysis techniques permitted in ASME-NH for "Load Controlled Stress Criteria", primary stress design criteria, only permit the use of elastic analysis. The methodology in the ASME code requires that the stresses resulting from analytical or computer aided elastic analysis to be classified into primary, secondary, and peak stresses. Many of the tools available to engineers today are not capable of classifying stresses as required by ASME code, or the effects of geometry, such as a stepped cylinder, make classification of stresses very difficult or impossible. Proper classification of primary stresses is an essential step in the design process; these stresses are required and applied to design rules and compared to allowable stress limits to ensure against various failure modes, particularly time-independent failure from short term loading events or time-dependent failure resulting from creep damage or creep rupture. Furthermore, a compact IHX does not lend itself to implementation of current ASME design procedures as the structure has little or no resemblance to classical pressure vessels. An alternative method, other than full inelastic analysis, is desired.

Recent efforts within the HTDM task at ORNL have focused on adoption and adaptation of the reference stress approach as a permissible inelastic analysis method to satisfy load controlled stress criteria in ASME-NH [2005 & 2006 reports on Reference Stress, McGreevy et al]. Note, the load controlled stress criteria address both time-independent and time-dependent load controlled failure of structures, i.e. plastic collapse and creep/creep-rupture respectively. While the reference stress approach has solid foundations in Europe and is extensively used in nuclear and non-nuclear industries in Europe, adoption of the approach by subcommittee members in NH has not occurred. The foundation of stress classification in ASME is based upon limit analysis; the reference stress approach is an extension of limit analysis to time dependent failure. Progress has been made in establishing reference stress design methods and criteria equivalent to the intent of the current ASME-NH code. Efforts will need to continue in FY07 to establish and obtain approval for this approach, including modifications as deemed appropriate by Subsection NH, as an alternative methodology used in designing for load controlled stresses within Subsection NH.

The NH subcommittee requires any proposed modifications of the rules to be agreed upon by consensus; consequently, any proposed reference stress design criteria will likely require confirmatory structural testing. Verification testing may include testing of simple coupon samples or structures such as plates or beams with or without structural features such as notches; additional testing can include actual structures such as the intersection of a nozzle-to-sphere, cylinder-to-cylinder, or similar. ORNL experience in HTDM during the period of the LMFBR program includes significant testing of coupons, simple structures, and complex structures. A review of available test conditions, materials, and test types will likely be required to support any proposed design criteria modification in NH. Research conducted in Europe to support the approach will also be used. Depending upon the consensus of the subcommittee, additional testing may be required; with sufficient funding, this will take 1-2 years. Additional testing specifically for the undetermined IHX design and material selection will need to be conducted for design verification; such testing will need to occur in direct collaboration with the stakeholders. Currently, AREVA is the stakeholder known to have made significant progress and consideration of various IHX design concepts.

The current methodology in ASME-NH requires either Elastic Analysis (NH-3214.1) or Inelastic Analysis (NH-3214.2). Typical inelastic analysis requires sophisticated constitutive modeling to address plastic strain hardening including cyclic loading effects and the hardening and softening that can occur with high temperature exposure, primary creep, creep hardening and softening, tertiary creep, etc. Full inelastic analysis is a timely and expensive effort that is unlikely to be used in conceptual and preliminary design phases, especially since such constitutive models do not and will not exist for materials that are not already approved by NH for use. Elastic analysis or simplified design methods are typically used instead in the early design stages.

NH is incapable of addressing primary load limits for structures with inherent complexities such as in a compact IHX because the structure does not resemble a typical pressure boundary; specifically, the great difficulty and/or inability to classify stresses properly and to implement stress intensities with the design criteria of NH-3222. Similarly, the current NH load controlled stress criteria do not address discontinuity effects such as notches or intersections of nozzles with cylinders or spheres – clearly, a compact IHX contains numerous discontinuities. On the other hand, the reference stress approach is capable and useful in conducting conceptual, preliminary, and in many instances final analysis for selection and design of structures with discontinuities. Furthermore, the reference stress approach does not require a sophisticated material model, nor does it require a sophisticated creep database and a creep exponent – only the geometry and loading need to be defined along with an arbitrary yield strength. Comparison of reference stress approaches with coupon samples, structural feature-like test samples, and small scale versions of a compact IHX will permit extensive direct comparisons of the materials and design of an IHX; correlation and verification will be necessary for regulatory acceptance. This activity will directly address some concerns of “unverified analytical tools” and “limited supporting technology and research” documented in the 1994 NRC review of DOE’s submittal of GE’s PSID for PRISM [NUREG-1368], as well as concerns brought out in the NRC/ACRS review of the CRBR project [Griffin, 1985, pp.299-306].

4.4.3 Simplified Methods and Criteria:

The effects of cyclic loading on the creep and fatigue of structures at elevated temperatures is currently addressed in the non-mandatory Appendix T of Subsection NH; certainly, adequate design against cyclic effects will be required by the NRC for licensing of the NGNP. Appendix T will likely become mandatory in Subsection NH, and will require merging with the load controlled stress criteria that are currently mandatory in Subsection NH; if not, both the NRC and industrial stakeholders will certainly conduct cyclic analysis to address failure modes associated with cyclic operation, e.g. strain limits and creep-fatigue. The effects of cyclic loading, cyclic thermal stresses due to temperature gradients through the section of a pressure vessel wall for example, are known to result in enhanced creep strains. Effectively, creep strains in a structure can and typically will accumulate at a faster rate due to cyclic loading as opposed to sustained monotonic loading. Limitations are placed on permissible strains both in the mandatory load controlled stress criteria as well as in non-mandatory Appendix T. Consequently, cyclic loading typically tends to reduce the useful life of components.

Simplified design analysis methods were developed during the LMFBR programs in an attempt to conservatively predict the effects of cyclic loading on creep strain accumulation, and to prevent ratchetingd. The very high temperatures of the NGNP reactor raised concerns regarding the validity of the simplified methods for VHTR applications. In fact, the 617 draft code case does not permit the use of simplified methods above 649^{o}C [Corum & Blass]. Verification of the validity of the simplified methods

d Ratcheting differs from enhanced creep strain, but can be thought of as an undesirable accumulation of strain due to excessive loads. Typically, ratcheting is limited to infrequent severe loading events such that both ratcheting strains and enhanced creep strain accumulation during normal operating conditions do not exceed permissible strain limits.

for VHTR applications has been the focus of ongoing work at ORNL [2005, 2006 reports McGreevy]. Unpublished results of various load histories reveal that one of the simplified methods, the B-1 Test, is not conservative when applied to a thin tube of Alloy 617 under constant pressure subjected to cyclic linear thermal gradients; a conservative solution is being developed[e]. The results have significant implications with respect to the current methodology in Appendix T, and will likely impact other simplified methods in Appendix T as well. Furthermore, the results will be of particular significance to addressing NRC concerns with analysis techniques as discussed shortly. No component should pass any simplified analysis and design criteria that would otherwise be rejected by a more detailed criteria and inelastic design analysis route. While the simplified methods are not intended to be rigorous solutions with great accuracy, the intent is to provide a conservative, simple, and efficient screening tool for conceptual and preliminary designs, and in many instances for final design stages of components that will not experience demanding or challenging loading. This approach leaves more time for designers to focus efforts on more demanding component design issues in final design stages.

The applicability of the simplified methods to general structures and loading, other than the simplified geometry and loading for which the methods were developed, is a concern – especially in light of recent unpublished results. The NRC has raised concerns regarding the use of "unverified analytical tools" and "limited supporting technology and research", as stated earlier. Their concerns also apply to cyclic loading effects. An extension of the research conducted on cyclic effects in 2005 was initiated in 2006; the extension includes the development of and evaluation of simplified analysis methods to general structures and loading where general structures and loading refers to non-axisymmetric structures and non-axisymmetric loading. This effort will need to continue for another 2-3 years. An alternative is to develop unified constitutive models for inelastic design analysis; however, without a material selection made and the lack of raw data for 617 that was used to generate the draft code case, efforts have been focused on simplified methods development. Furthermore, none of the current simplified methods address cyclic primary loads or effects of the history of variable loading; additional simplified methods or full inelastic analysis may be required to address such issues.

4.4.4 Failure Models for Design Criteria- Creep-fatigue

Cyclic loading introduces a failure mechanism that must be considered in structural components at elevated temperatures, creep-fatigue. This failure mechanism is addressed in Appendix T. Code developers and researchers worldwide generally recognize that the current linear damage accumulation rule for creep-fatigue used in Appendix T and many other design codes has significant shortcomings, particularly at higher temperatures and longer times. Various improvements, such as those based on ductility exhaustion and damage rate concepts, have been proposed, but none have been backed by sufficient R&D to allow their adoption as a replacement for linear damage in Appendix T. These several shortcomings should be remedied for the NGNP, particularly for use of Gr91 steel at elevated temperatures and the IHX material (Alloy 617 or 230). Creep-fatigue has been a technical and safety issue that the NRC has raised concerns about consistently; this concern remains today [Swindeman, CRBR, PRISM, Stuart Rubin].

Efforts have been initiated with creep-fatigue testing in air of Alloy 617 and 230 at ORNL and INL; furthermore, two tasks within the DOE-ASME collaboration for GenIV address creep-fatigue of Gr91 steel and Hastelloy XR[f]. The DOE-ASME tasks are efforts to make modifications to the existing

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e Unpublished results were generated in 2006, but likely will be included in the August 2006 report on simplified methods.

f Hastelloy XR is a nickel based super alloy; success by the Japanese in predicting creep-fatigue of Hastelloy XR is being pursued as a possible procedure for addressing creep-fatigue of Alloy 617 or 230.

methodology for predicting creep-fatigue damage for very short term implementation into existing ASME code; this timeframe will likely be 3-5 years. The effect of the environment, i.e. impure helium, may have significant effects on the creep-fatigue performance of materials; while not addressed by the ASME code, such effects must be demonstrated as insignificant or appropriate means taken to account for differences. Hence, establishing testing capabilities at very high temperature in impure He environments is essential and remains to be accomplished on the large scale required.

The University of Illinois in Urbana-Champaign (UIUC) has resubmitted a NERI proposal to conduct research that addresses major materials performance and design methodology issues for the design and construction of very high temperature components, e.g. an IHX. ORNL will be collaborating with UIUC if the grant is awarded. The work provides a synergy between the development of simplified, but robust, design rules for high temperature systems and materials testing, performance and improvement for these systems. The program will address time-dependent materials properties (creep, creep-fatigue, high temperature corrosion) so that these issues can be properly handled in the design of components with complex stress states, long intended service lives and aggressive operating environments. This includes refurbishment of an impure He loop that will be used in creep and creep-fatigue testing of various grain sizes of IHX materials at high temperature in anticipated NGNP environments. Welds will also be addressed.

The creep and fatigue performance of materials will certainly change significantly with variations in product form and accompanying grain size, e.g. plate vs. sheet vs. foil with large or small grains. Data and understanding of such variations for both Alloy 617 and 230 for the IHX must be generated and obtained, for both air and impure He environments. Similar test programs will be required if Gr91 steel is to be used for reactor or core internals, and possibly the reactor pressure vessel if the operating temperature of the RPV is considered high enough that creep is significant. Air testing provides a baseline that is required and used for the ASME code; environmental effects are required by the NRC for licensing in addition to the baseline. Conducting tests of this nature is not trivial; appropriate test facilities are rare if they exist at all, especially for controlled impure He facilities. Refurbishing or modification of existing equipment requires careful planning, time, and resources. Such activities will take at a minimum 1 year to establish capabilities and work out significant issues; additional time and resources will be required to expand upon such capabilities and will be driven by available resources. If awarded a NERI grant, efforts at the UIUC will be very beneficial.

Acceleration of creep-fatigue experiments is typically accomplished by using shorter durations during a test for creep to occur. Literature data for Alloy 617 indicate that longer and longer creep durations in the creep-fatigue tests continue to decrease the number of fatigue cycles before failure, i.e. accelerated tests do not agree with less accelerated tests and do so non-conservatively. An understanding of the mechanisms of damage is desired, especially for extrapolating and applying accelerated testing to applications with very long service lives coupled with environmental effects. This issue may well be amplified if smaller grain sizes are used for a compact IHX. The smaller grain size will certainly lead to less creep resistance and more fatigue resistance, but the stress and strain levels in a compact IHX will need to be low in order to meet functional requirements for operating at 850-950^{o}C for 100,000's of hours. The creep mechanisms that are active at such low levels are likely driven by diffusion processes as opposed to dislocation movement in power law creep; consequently, to observe any interaction of fatigue and diffusion mechanisms will likely require longer test durations. Hence, the dominate deformation mechanisms may impose further restrictions on acceleration of creep-fatigue tests. On the other hand, low levels of strain and stress may indicate that fatigue damage will be negligible, but the impact of fatigue on creep (rather than creep on fatigue) may be significant. The effects of higher stress and strain levels for infrequent anticipated and unanticipated loading events will also be required.

The creep-fatigue interaction diagram for Gr91 steel in Appendix T of Subsection NH is much more conservative than that of other alloys such as stainless steel and 2 ¼ Cr-1Mo steel. The reason for the conservative diagram is a lack of understanding of creep-fatigue interaction for Gr91 steel; creep-fatigue results have been observed to vary significantly for Gr91 steel when the design methodology for predicting creep-fatigue damage in Appendix T is implemented. The cyclic softening of Gr91 steel, as opposed to the stable or cyclic hardening behavior of normalized 2 ¼ Cr-1Mo and stainless steels, may be partially responsible for the variation. Research conducted by the U.S. and Japan to address creep-fatigue of Gr91 steel ended prior to reaching an understanding of the material and the implications of the design methodology to predict interaction of creep and fatigue. Consequently, Subsection NH chose to implement a conservative creep-fatigue design envelop until a better understanding to warrant modifications was reached. Furthermore, the methodology for predicting creep-fatigue interaction may require modifications for a given material; currently, the methodology is applied identically to all materials. Weldment fatigue and creep-fatigue data is lacking as well.

The discussion of creep-fatigue clearly indicates a weak point in failure criteria in Subsection NH, and many other codes for that matter. A fundamental understanding of the creep-fatigue interaction process does not exist. Significant progress has been made in the field of thermo-mechanical fatigue by simply observing active mechanisms, modeling such mechanisms individually, and then integrating individual models in such a way to capture the interaction between mechanisms. Different mechanisms likely exist in different classes of materials, resulting in different interactions as well; a more robust and perhaps material or material class specific life prediction methodology should address this fact. A similar process to understand and predict creep-fatigue behavior is essential if the existing methodology is to be improved or replaced. This will require the integration of an understanding of material behavior at the microstructural level with mechanics. As opposed to the DOE-ASME tasks, this approach will require more time and is a more fundamental approach. The timeframe is uncertain, but will take at least 5 years. If successful, it will provide a solid foundation for establishing more robust life prediction methods and extrapolation of such methods for NGNP components that will be in service for very long times. Such an understanding would certainly resolve creep-fatigue concerns expressed by the NRC.

4.4.5 Extrapolation of Creep Data for Very Long Service Applications

The 60-year plant life for the NGNP is unprecedented. No creep database contains test durations of 525,000 hours. Consequently, extrapolation of creep data is required. Typically the longest creep test durations in the U.S. are limited to several tens of thousands of hours, with limited data approaching 100,000 hours. Extensive databases in terms of data and test durations on the order of 100,000 hours or more exist for materials such as Alloy 800H that have decades of research and development and industrial experience. However, the U.S. and ASME database for Alloy 617 is not as large nor does it include as extensive test durations as say Alloy 800H. Furthermore, various compact IHX designs entertain a wide range of product thicknesses from 0.2mm (8 mils) to 2mm (80mils). Assuming that a minimum of 10 grains would be desired across the thickness, grain sizes would vary from maximum sizes of about 20 μm (ASTM G.S. 8) to about 200um (ASTM G.S. of 1.5). The current draft code case considers various product forms including hot rolled (H.R.) sheet, H.R. round, H.R. flat, cold rolled (C.R.) sheet, forged square, cold drawn tube, and extruded tube. Within these products, grain sizes varied from approximately 127-360 μm (ASTM G.S. 0 to 3). As such, the draft code case for Alloy 617 would only support these grain sizes; selection of smaller grain size product forms (thin section or thin product forms) for a compact IHX will require additional testing and R&D. Extrapolation of creep data used for the 617 draft code case is a significant issue for the periods required for NGNP applications and will be a major NRC concern; similar concerns will arise for material with smaller grains as the database is less extensive. A nickel based super alloy IHX will not operate for 60 years, but must be replaced – likely several times. Assuming an IHX will operate for 20 years, testing will need to be conducted at various stress levels and times to verify what deformation mechanisms will be operative during service to determine if extrapolation of data is justified, to what extent data can be extrapolated, and how to extrapolate.

Compared to the current draft code case, Alloy 230 has a less extensive database compared to Alloy 617, and has similar issues with respect to various grain sizes and extrapolation of data for NGNP needs and compact IHX designs. Haynes has provided creep curves for Alloy 230 to ORNL for use in ASME code needs; this likely will significantly reduce testing needs for codifying this material although efforts are needed to assemble, analyze, and summarize the data. Unfortunately, Special Metals reports that the raw creep data for 617 no longer exists – the data only exist in processed and tabulated form. This has significant ramification in terms of generating test data to support constitutive modeling efforts, apparent errors in tabulating or processing some data, and possibly extrapolation of test data.

Several issues arise in terms of obtaining ASME code approval for these materials.

1. In order to meet the schedule proposed for the NGNP, extrapolation of creep data will be required; an understanding of dominate deformation mechanisms that apply to the extent of extrapolation is required to help ensure that extrapolation is appropriate and safe.

2. Generation of mechanical property data on various grain sizes is necessary to obtain interim or screening material property data. The interim data will be used to generate interim constitutive models to assist in predicting longer term material behavior in the form of isochronous curves – this subtask is discussed in detail later. Preliminary design efforts will need to incorporate the alternate and simplified design methods with the interim mechanical property data and constitutive equations, aging, and environmental data & results to reach a decision on what type of IHX to design, construct, and operate for the first NGNP reactor.

3. A material must be selected for fabricating an IHX – Alloy 617 or 230. The selection of an IHX material will clearly depend upon limited interim mechanical property data, aging, and environmental studies. The option of using Hastelloy XR may still be considered a viable option if the U.S. can reach an agreement with Japan to acquire their database of Hastelloy XR to expedite an ASME code case or acceptance by NH; Hastelloy XR has been accepted by the Japanese nuclear code and used in the Japanese High Temperature Test Reactor (HTTR) at temperatures of 850oC-950oC. The limitations of this alloy will be similar to the current draft code case for 617 in terms of grain size and product form and limitations on service time.

4. An extensive testing program will be required to generate supporting tensile, creep, fatigue, and creep-fatigue data to be used in determining stress and strain allowables, fatigue and creep-fatigue design criteria for acceptance of the selected material by the ASME code[g].

4.4.6 Data Required for ASME Code Case Acceptance

A brief summary of the extent and type of data required by ASME for acceptance of a material for a code case or adoption follows [Ren & Swindeman 2004 High-Temp Mtls Test Plan].

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g The task of generating mechanical property data for use in establishing limits for design criteria is summarized in the High Temperature Metallic Alloys section and the Vessel sections of this report. The task of establishing design criteria such as rupture, fatigue, and creep-fatigue criteria is closely linked to the task of generating data, but is a separate task.

1. Tensile test data every 25oC, including testing 50□C higher than the maximum expected use temperature; duplicate testing will be required.

2. If cold forming is permitted, unlikely in the application of 617 and 230 at very high temperatures, cold forming limits must be determined, otherwise, if certain temperature and cold forming limits occur, the cold work can impair material properties such as fatigue, creep rupture, impact toughness etc. Subsection NH requires a post fabrication heat treatment depending on the amount of the cold work.

3. The aging deterioration effects on the ultimate tensile strength and the yield strength are treated in Subsection NH with strength reduction factors. This will require testing and modeling programs.

4. To support the NGNP design and construction, the values of stress-to-rupture should be acquired at intervals of 25ºC. A code case for a material may or may not permit relaxation of the temperature increment within certain temperature ranges if the stability of the material is well understood and demonstrated.

5. The values of the stress rupture factor should be generated from derivation and modeling based on data provided by stress rupture tests conducted on the weld metal, the base metals, and weldments of the candidate materials. This is discussed further in the following subsection.

6. The curves for strain range versus number of allowable cycles should be generated for each candidate material at 400ºC (752ºF) and above, increasing at intervals of 50ºC, to 50ºC higher than the maximum design service temperature.

7. Because of the high operating temperature for the NGNP reactor system, the requirements for a material to become qualified are considerably more extensive than those in current Subsection NH since there will be little or no application experience to draw upon, and the required life of 60 years for the NGNP reactor system is nearly double that currently permitted by Subsection NH rulesh.

8. The criteria for setting the allowable stresses for ASME Section III Class 2 and Class 3 components are identical to those for ASME Section I and Section VIII, Division 1. Values are provided in ASME Section II, Part D, Tables 1A and 1B. However, the usage of these tables is not permitted for temperatures above 371ºC (700ºF) for ferritic steels and 427ºC (800ºF) for austenitic stainless steels and nickel base alloys. For Class 2 and Class 3 component construction for higher temperatures, it is necessary to revert to code cases, such as N-253-11, in which stress intensity, rather than maximum stress is used and many of the rules set forth in Section III, Subsection NH must be followed. Data requirements for use of N-253-11, N-254, and the like are similar to those identified for Subsection NH.

9. Alloy 617 must also be added to the low-temperature rules of Section III.

10. Modification of all the former material allowables due to aging and environmental effects will also be required for either the ASME code or an NRC license for the IHX.

h Service approval for 300,000 hours is not typical for all materials in NH; the duration denotes the longest permissible service time currently approved in NH.

The effort in terms of time, man-hours, and testing facilities should not be overlooked for Task 10. If Alloy 617 and the current grain size range supported by the draft code case for Alloy 617 are selected, the timeline and resources for acceptance into the ASME code may be much shorter than any other alternative that includes a nickel base super alloy, with exception to Alloy 230 as discussed below. Any and all thin section (small grain size) alternatives will essentially require codification of a new material with an *extensive* testing program and much longer time period to obtain ASME code approval. Acquisition of German 617 data for large grain sizes could greatly expedite addressing the extrapolation issue; the database is available, but acquisition remains stalled due to legal issues. Alloy 230 in thick section (large grain size) may or may not take as long due to recent acquisition of creep data and curves from Haynes International.

Time is required to assess and review this data to determine its impact on the program. Best estimates are 2-3 years of testing and data analysis of existing international 617 and 230 databases to justify extrapolation. For thinner product forms (smaller grain sizes), a minimum of 3 years to a more likely 5 year testing program will be required. Note, five year creep data would require extrapolation of rupture data by a factor of 4; three year data by a factor of about 7. This does not include time for obtaining acceptance from ASME for a code case or addition of material into the code. Furthermore, long term creep testing should continue during and after code acceptance to support resolving any NRC concerns that arise in terms of verification of extrapolation or revision of extrapolation with longer test data.

4.4.7 Weldments and Discontinuities

Most high-temperature structural failures occur at weldments. Welded pipe, for example, has failed in high-temperature fossil plants after many years of operation. Reliably guarding against weldment failures is particularly challenging at high-temperatures, where variations in the inelastic response of the constituent parts of the weldment (i.e., weld metal, heat-affected zone, and base metal) can result in a strong metallurgical discontinuity. In the hearings for a construction permit for CRBRP, early weldment cracking was identified by the NRC as the foremost structural integrity concern. The NRC and ACRS felt that designers should have a better understanding of the metallurgical interactions that take place in weldments and their effects on weldment life. The CRBRP project committed to a five-year development program to address these issues prior to issuance of a plant operating license. The program was never carried out because of the subsequent demise of the project. This issue will certainly resurface with NGNP and other GenIV reactors.

A review of the existing weld reduction factors for materials currently in NH will be required to begin to address the NRC concerns [Corum & Blass]. Additional testing may be required, especially demonstration of the welding process of thick sections of Gr91 steel, filler metal, thick section properties, and heat treatment. Collaboration with AREVA will be crucial in order to complete this activity successfully and in the timeframe for the NGNP. Extensive research and development will be required to demonstrate the joining process, filler metal, and any necessary heat treatment for a to be determined material, grain size and product form, and joining process for the to determined IHX. Even if a grain size covered by the Alloy 617 draft code case is used, weld metal creep rupture data is sparse. Overall, collaboration with AREVA and/or GIF nations would assist greatly in the costs, time, and scope of this activity.

Like metallurgical discontinuities, geometric discontinuities (i.e., notches and other local structural discontinuities) are sources of component failure initiation. The adequacy of the design methodology to handle such discontinuities is a reliability and licensing issue, particularly when heat-to-heat variability, strain hardening/softening, triaxiality, aging and environmental effects on ductility and creep ductility, and cyclic loadings are considered. This was the second unresolved issue (after weldments) in the CRBRP licensing hearings, and again a multi-year development program was required by the NRC. Reviewers felt that the effects of stress gradients were not reflected in creep-fatigue design limits and that general notch weakening and loss of ductility under long-term cyclic loadings were not well understood. Notches will need to receive particular attention in the development of the required high-temperature structural design technology for the NGNP and other Gen IV reactors. Selection of materials for the NGNP will permit focused efforts as material behavior can play a significant role in adequately resolving this issue.

4.4.8 Inelastic Design Analysis Methods

Constitutive models, or equations, are the key ingredients of the inelastic design analyses that are required by Subsection NH. These equations describe the inelastic, multiaxial flow response of a material to complex time-varying, multiaxial, loadings. Their development must be based on results of a body of exploratory experimental uniaxial and multiaxial tests in which specimens are subjected to a variety of relevant thermal and mechanical loading histories. These exploratory tests reveal key material behavioral features (e.g., flow, hardening/softening, recovery, path dependency, etc.) that must be adequately reflected in the resulting constitutive theory. Ultimately the adequacy of the constitutive equations must be demonstrated by incorporating them into inelastic structural analysis computer programs and benchmarking the resulting predictions against the results of pertinent high-temperature structural tests. The ultimate goal and responsibility of Subsection NH is to provide the combination of constitutive models, structural analysis procedures, and design criterion to reliably provide a suitable margin against various structural failure mechanisms.

Experimentally based constitutive equations must be developed for NGNP materials when distinguishing between plastic and creep behavior is not possible. This applies specifically to nickel based super alloys, as the current draft code case for Alloy 617 requires such constitutive equations. Furthermore, AREVA has expressed interest in constitutive equations, and is generating data to support such development for Alloy 230. Additional input from AREVA indicates that the RPV will experience much lower temperatures than originally anticipated by the U.S. earlier in the NGNP program. As such, the issue of indistinguishable creep and plasticity no longer applies for Gr91 steel. However, AREVA is pursuing approval for ASME to raise the insignificant creep temperature for Gr91 steel from 371^{o}C to approximately 400-450^{o}C. Depending upon if and how Subsection NH modifies the definition of insignificant creep, and available data to support the criterion, a constitutive equation will likely be required to capture the effects of any microstructural changes over long periods of time and at temperature on subsequent material behavior in terms of strain accumulation. Clearly the constitutive equation development effort must be carried out in close coordination with the materials data tasks, since they provide the design data for the final quantification of the models.

Recent modeling efforts on Alloy 617 includes rough first order predictions based on Ashby deformation mechanism equations for various grain sizes, since the U.S. has no experimental data to support modeling efforts for different grain sizes at this point. AREVA has access and/or is generating relevant data that would be useful in developing constitutive models; establishing a formal collaboration mechanism would be of great value to this subtask. The results indicate that smaller grains sizes can result in the change of the deformation controlling mechanism from power law creep to diffusion dominated mechanisms for times, temperatures, and stress levels applicable to the NGNP IHX. Consequently, the maximum constant stress permitted over a 100,000 hour period can be significantly lower than that in the larger grain material currently in the draft code case for Alloy 617; the difference obviously increases with temperature. Experimental data are required to first validate the model, and then make improvements. For conceptual and preliminary design needs, the model, preferably the experimentally based model, should be incorporated into a suitable structural analysis program and coupled with thermal analysis to investigate a variety of thermal and mechanical load histories, design configurations, and so forth. Full inelastic simulation of an IHX would be very challenging given the complexity of the structure, requirements of mesh size, and so forth. Incorporation of simplified design methods that utilize elastic-plastic finite element methods and lend themselves to complex structures will be of great value. Isochronous stress strain curves generated by even a rough model permit designers to investigate design issues and gain experience in ascertaining difficulties and issues associated with various design configurations, material proper limitations, and so on.

Once initial and sufficient mechanical property data, e.g. tensile, creep, fatigue, and relaxation testing, have been compiled, the experimental data can be used to develop an experimentally based constitutive model.

When a selection of material is made, more extensive testing and modeling development will be required. While every effort will be made to utilize information from the data generation tasks, experience shows that an extensive test program of uniaxial and biaxial exploratory tests will be required to establish key response features resulting from various mechanical and thermal load histories, and biaxial loading paths. Establishing test matrices for such tests a priori is not easy; tests must be planned and carried out in concert with model development efforts, and subsequent tests depend on the findings from the previous tests.

The exploratory deformation tests being discussed are very demanding in terms of test equipment, since they involve precise control of load history and conditions. Often potential surfaces, (akin to yield surfaces or loading surfaces in classical plasticity) must be established in biaxial stress space. One example of this in past unified equation development efforts was the determination of surfaces of constant inelastic strain rate. In addition to uniaxial specimens, thin-walled tubular specimens subjected to combined internal pressure, axial loading, and torsional loading will be employed. This will require that suitable test facilities, controllers, and extensometers be designed, procured, and built. Also, structural tests were recommended to be performed against which inelastic analysis predictions could be benchmarked. None of this testing was ever carried out because of the demise of the LMFBR program. A final iteration on the equations for the final NGNP materials will likely be required. To be of value, this must be completed early in the final design phase. This final iteration will correct shortcomings identified from the results of structural tests and from design analysis experience.

An integral part of constitutive equation development, particularly in the case of unified theories, is that stiff equations and computational expensive algorithms can result. Advances in development of such equations have been made since the LMFBR period, and incorporation of the models into finite element programs for parameter studies, behavioral predictions, and analyses of structural tests is much more readily feasible. ORNL has purchased commercially available software to assist in this area; the software is flexible in that it permits users to define the variety of equations that form the constitutive equations, and includes an optimization package that assists in fitting material parameters to actual test data. Use of the equations outside of the temperatures and times for which test data exist to develop the model must be made with care; however, experimentally and mechanistic based constitutive equations help ensure that extrapolation is prudent and possible.

The timeline to obtain constitutive models is directly related to the time required to generate experimental data. As such, interim models can be obtained after 1-2 years of testing of a given material and product, with modified models obtained after testing for another year. Structural features testing may reveal shortcomings and require a final iteration for final design use. Funding has been limited, and has been utilized for the most part on refurbishing equipment and focusing on Ashby deformation models to predict behavior. Selection of a material and product form will permit focused efforts of testing and modeling to complete this subtask.

4.4.9 Confirmatory Structural Tests and Analyses

Confirmatory time-dependent structural tests have, in the case of most of the current Subsection NH materials and design methodology, provided data that either (1) validated the high-temperature design methodology (inelastic design analysis methods, simplified design methods, alternate design methods, and design criteria) or (2) led to changes in inelastic design analysis guidelines or ASME Code rules. The role of structural tests will be even more important for the NGNP because of the lack of long-term service experience. This need was recognized by the developers of the draft Alloy 617 Code case. The need for very-high-temperature, time-dependent tests of Alloy 617 structural models was identified to (1) provide a better understanding of structural behavior and failure modes, (2) validate inelastic analysis methods, and (3) provide some applications feedback to the Code. Even in the case of Gr91 steel, applicable experience within ASME is limited as the material was recently approved for use in Section III and Subsection NH, and while structural tests were planned under the LMFBR program, they were never carried out.

One should note that the structural tests to be performed under this subtask are not tests of NGNP component structures. Rather, they are tests of carefully chosen, simple, but representative, geometrical and metallurgical features subjected to time-varying, and cyclic thermal and mechanical loadings. The tests will be contrived to explore key features or problem areas of the methodology and to validate inelastic design analysis (constitutive equations), failure criteria (e.g. creep-fatigue and limit loads), and, where applicable, alternate design methods and simplified design methods and criteria. Key aspects to be examined include notches and other discontinuities, weldments, and elastic follow-up. Verification of reference stress techniques for variable primary loading (as opposed to constant load testing) may also be required by the NH Subcommittee and the NRC.

Three types of structural tests, and associated analyses, were successfully employed in developing and validating methods and criteria for the liquid-metal reactor, and tests, usually to failure, of these same types are envisioned for the NGNP reactor. The three types of tests and likely examples of each are tabulated below.

1. Basic structures
    a. Axially-loaded notched bars and bars with a step change in diameter (e.g., to generate data for the SMT method)
    b. axially-loaded flat plates with holes, axial/transverse welds
    c. Two-bar shakedown and ratcheting tests
    d. pressurized tubes with step changes in wall thickness, built-in ends
2. Simple structures
    a. beams in bending
    b. pressurized cylindrical shells with heads
    c. thick cylinders subjected to cyclic radial thermal gradients and axial loads on internal pressure

3. Component configurations

    a. nozzle to spherical shell with internal pressure

    b. nozzle to spherical shell with internal pressure and non-axisymmetric loading of the nozzle

    c. representative pieces of a compact IHX under:

        - internal pressure loading
        - cyclic internal pressure
        - internal pressure with thermal gradients
        - other relevant loadings as deemed significant during various design stages

Experience has shown that with these tests most of the key behavioral features exhibited by actual systems and components can be modeled and evaluated. The resulting data will provide a check of the most important features of the methods and criteria.

Planning for this subtask could have started early in the program, but limited funding scenarios necessitate postponing this significant activity until material selections are made. A DOE-ASME task is anticipated to be funded in late 2006 to initiate a review of past structural testing and additional testing needs. A current DOE-ASME task of NRC concerns may reveal additional needs. Testing requirements directly related to the NGNP (IHX) will certainly arise; additional planning will be required, design of tests, and conducting tests. Limited testing, 1-2 years, in support of early verification of design criteria would be useful to verify the reference stress method. Additional testing later in the program will take at least 5 years; additional testing to resolve NRC concerns may also be required.

4.4.10 Impact & Risk of Scope of Activities and Schedule

The objective of the HTDM task is to provide technology and R&D in support of ASME code needs for a vendor to obtain a construction and operating license from the NRC for the NGNP. In terms of major components for the NGNP, the HTDM encompasses materials and design methods & criteria for the RPV and the IHX. As such, success of this task is dependent at least upon adequate funding and activities in support of the RPV and IHX. Draft schedules for RPV and IHX were developed by ANL, INL, ORNL etc. in the NGNP meeting in Salt Lake City in June 2006. The following discussion relates directly to these schedules.

4.4.10.1 Very High Temperature Alloys. An IHX is required for the NGNP to produce hydrogen; however, an IHX only for this purpose would not be required to handle the total heat load of the plant. The definition of design and material performance requirements is a critical starting point. In order for a decision to be made with respect to a design and material selection, material data is required. Presently, the selection of 617 or 230 has yet to be made. Furthermore, selection of a product form with a small grain size or large grain size has yet to be made as data is lacking to support such a decision. Selection of a material inherently includes a feasible joining process with acceptable weldment properties. The SLC meeting (noted previously) schedule reflects these needs with most activities starting before or in FY-07. The scope of these activities is immense, particularly if options of 617 and 230 along with grain size options are all considered. Funding requirements follow suit. Many of the subtasks discussed above depend upon selection of a material and product form (grain size), e.g. creep-fatigue, inelastic design analysis methods, failure models, confirmatory structural tests, and to some extent even simplified methods. Selection of 617 in the product form and grain size supported by the draft code case will relieve some requirements for R&D; however, not by an order of magnitude. Furthermore, it remains to be determined if an IHX made of large grain size 617 can be designed to meet performance requirements and be economical. If a thinner product form is required which contains smaller grains, 617 and 230 would be on nearly equal footing as each would require significant data generation, 617 perhaps having a slight advantage if existing studies of environmental effects for large grain 617 apply directly to smaller grain product forms.

Several scenarios are available. These scenarios along with significant pros and cons are listed below:

1. Select 617 in the grain size supported by the draft code case.

   Pros:

   - This will reduce data generation requirements, assisting in with schedule constraint.

   Cons:

   - Ramifications with respect to IHX design and performance requirements unknown.
   - Vendor material selection may differ, as may grain size to support a particular IHX design; joining process may differ significantly as well. Focus of DOE supported R&D would need to change – no gain in early selection of 617 and grain size.

   Risk:

   - A different material and grain size selection by a vendor would effectively push the schedule out the length of additional time until this occurs. Obtaining relevant R&D to support ASME code and licensing would to a very large extent refocus, with little progress made in the meantime.

2. Select 617, but pursue smaller grain sizes.

   Pros:

   - Provides preliminary design data in support of selection of grain size and impact on assessment of design and performance of IHX concepts based upon engineering analysis and material property data.
   - Equipment and infrastructure investments would apply to testing Alloy 230 if vendor does not select 617, for both small and large grain sizes.

   Cons:

   - Requires some additional preliminary material property data.
   - Requires joining process for thin section product.

- Vendor material selection may differ; joining process may differ significantly as well. Focus of DOE supported R&D would need to change – less gain in early selection of 617.

Risk:

- A different material selection by a vendor would effectively push the schedule out the length of additional time until this occurs. Obtaining relevant R&D to support ASME code and licensing would to a very large extent refocus this portion of the program. If smaller grain size is selected, experience and any investment in infrastructure will assist in transition from 617 to 230. Small schedule advantage over scenario #1.

3. Pursue 617 and 230 in product form and grain sizes for which data exist today.

Pros:

- Recent acquisition of Alloy 230 creep data from Haynes permits this option.

- Grain size selection reduces data generation requirements, assisting with schedule constraint.

Cons:

- Requires some additional preliminary material property data, e.g. creep-fatigue of Alloy 230.

- Requires joining process for thin section product, both Alloy 617 and 230.

- Vendor grain size selection may differ to support different IHX design concept; joining process may differ significantly as well. Focus of DOE supported R&D would need to change – no gain in early selection of grain size.

Risk:

- A different grain size selection by a vendor would effectively push the schedule out the length of additional time until this occurs. Obtaining relevant R&D to support ASME code and licensing would to a very large extent refocus the program. Vendor selection of Alloy 230 in grain size supported by Haynes data would have only slight disadvantage over 617, mainly in the area of environmental compatibility. Small schedule advantage over scenario #1, about equivalent to scenario #2.

4. Pursue 617 and 230 in all product form and grain sizes.

Pros:

- Recent acquisition of Alloy 230 creep data from Haynes.

- Vendor material and grain size selection encompassed.

Cons:

- Requires additional preliminary material property data for small grain size Alloy 617.
- Requires additional preliminary material property data for 230 in small and large grain sizes.
- Requires joining process for thin section product, both Alloy 617 and 230.

Risk:

- Requires unrestricted funding and resources. Vendor selection of material and grain size easily Alloy 230 in grain size supported by Haynes data would have only slight disadvantage over 617, mainly in the area of environmental compatibility. Small schedule advantage over scenario #1, about equivalent to scenario #2.

5. Pursue 617 and 230 in thin product form (small grain sizes).

   Pros:

   - Data generation provides needed material properties for selection of material and section size.

   Cons:

   - Requires additional expenses for acquiring, machining, and testing material.
   - Should include joining process R&D for thin section products.
   - Thick section 617 and 230 material acquired in 2006, samples machined, test machines configured for testing thick section product.
   - Does not address the thicker sections needed for headers, manifolds, etc.

   Risk:

   - Vendor will not select small grain size product form.

6. In addition to scenarios1-5, the option of using Hastelloy XR exists.

   Pros:

   - Requires no action or commitment on the part of DOE, until and if a vendor selects Hastelloy XR.
   - Japanese currently using Hastelloy XR in heat exchanger in the HTTR – experience in VHTR.
   - Material is codified in Japan for nuclear use; accelerates code acceptance in ASME.
   - Extensive environmental database and experience exists.

Cons:

- Requires agreement between U.S. and Japan to obtain database.
- Japanese database may be limited to large grain size, similar to Alloy 617 draft code case.

Risk:

- May have similar issues that need to be addressed for 617 draft code case, e.g. weldments.

These scenarios can be weighed with respect to different funding scenarios. Unrestricted funding would support scenario #4; establishing collaborative agreements with a vendor would clearly facilitate the qualification of alloys that could be used for construction of the IHX.

4.4.10.2 Reactor Pressure Alloys (Grade 91 Steel SA 508/533 Steel). The RPV materials being considered are Grade 91 steel (AREVA) and SA508/533 steel (S. Africa). One must be aware that the use of Gr91 steel is not limited to the RPV in the AREVA design, but will also be used in internals; as such, creep-fatigue of Gr91 steel is an issue in the AREVA design – regardless of the RPV.

The use of 508/533 has little impact in terms of code needs other than one need. Code Case N-499 specifies permissible time and temperatures excursions for 508/533. The PBMR design seeks extension of these times and temperatures. PBMR appears to be planning to pursue this if DOE does not. Otherwise, evaluation of data used for the existing code case and the associated basis for the code case required a review. Generation of a desired test plan is needed with subsequent testing and analysis to support the extension of the code case. An estimate is not available, but a guesstimate is 3 years. However, it should be noted that this extension not required for the PBMR-400 plant based on current analysis performed by PBMR.

There are several key HTDM related activities required for the usage of Grade91 steel and these include:

1. Creep-fatigue
2. Insignificant creep
3. Thick section effects on mechanical properties (base metal and weldments)

Creep-fatigue was discussed earlier at length, with a best case timeline of 3-5 years. Insignificant creep was discussed briefly; the on-going DOE-ASME task has yet to be completed with a recommended test matrix. As such, a guesstimate of 3-5 years to complete this is proposed. Thick section effects on mechanical properties, base metal and weldments is also estimated to take 3-5 years. Since Gr91 steel is already a code approved material, modifications may take less time, although this is an assumption. As such, modifications could take 2-3 years after R&D to support of such changes. As these are currently being discussed in Subsection NH, and 2 of the 3 are being addressed by DOE-ASME tasks, it is quite possible that modifications may be implemented into ASME code within 1 year of completion of required R&D.

AREVA has developed a process for thick section welding of Gr91 steel. Demonstration of successful large scale post-weld heat treatment of thick section Gr91 steel required by NGNP may still need to be demonstrated. Without adequate collaboration with AREVA, even with sufficient funding, there is a potential risk that even an independent DOE supported development and verification of welding process and heat treatment would not meet the long-lead procurement deadline.

4.4.11 ASME Codification & NRC Licensing

The current NGNP program schedule includes application for an NRC construction permit in 2010 with the permit issued by the end of calendar year 2013. It is assumed or interpreted that this will be a Preliminary Safety Assessment Report (PSAR) that addresses overall safety issues related to the general design and operation of the NGNP, or a more extensive and detailed application.

No dates are listed for submission of a *Final Safety Analysis Report (FSAR)* required to obtain an operating license. This FSAR shall include information that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components and of the facility as a whole. This includes but is not limited to a final analysis and evaluation of the design and performance of structures and components taking into account any pertinent information developed since the submittal of the PSAR. It will also include a description and evaluation of the results of the applicant's programs, including research and development, if any, to demonstrate that any safety questions identified at the construction permit stage have been resolved. Certainly this will require completion of R&D to support the application. The assumption is made that the review of the operating license will take at least 3 years. The current NGNP schedule calls for the NRC operating license to be completed by end of calendar year 2017; hence, the application must be submitted by the end of calendar year 2014. This leaves only 8 years for all R&D efforts to be completed to support the application. The following program scenarios are being considered:

4.4.11.1 Selection of materials to reduce the time and effort required for R&D. If the S. African design or equivalent is selected, the research on Gr91 steel need not be pursued. However, a very high temperature alloy, namely Alloy 617 in large grain size, would need to be pursued. If DOE can convince stakeholders such as AREVA and S. Africa to combine design concepts with one lead stakeholder, then this best scenario would require full funding immediately and immediate agreement between stakeholders. In this case, the 2013 deadline for submission of an operating license could be met with a high likelihood that ASME code support and approval would be in place. Risk will be consistent with feedback from the NRC prior to application for operating license. This would require that any unresolved issues that the NRC/ACRS may have as a result of the operating license application be resolved in the last 1-2 years prior to CD-4, the “Start-up and Testing”, in 2018. In terms of HTDM requirements, this may be considered as the lowest risk approach for an NGNP.

4.4.11.2 Pursue issues related to Grade91 and R&D to support thin section 617 or 230 for use in the IHX. This will be the highest risk approach. Essentially, a new material would need to be tested and approved by ASME code. With full funding, the timeline could extend another 2-3 years. The probability that ASME code approval for 617 or 230 thin section material may not be obtained prior to submitting the FSAR exists; as such, the FSAR would need to be submitted prior to ASME approval. This is a licensing risk that would need to be considered.

If unrestricted funding is not provided, a realistic timeline for construction and operation of the NGNP will extend as well.

4.5 Nuclear Ceramics and Composites

The first area of discussion in this section deals with the issue of metallic control rods for the prismatic VHTR design. This information is presented to put the case for composites, particularly for control rod applications, into the proper context. The remainder of this section, as the title of the section implies, discusses the use of composites in the VHTR.

4.5.1 Metallic Control Rods

The use of carbon-carbon (C/C) and/or silicon carbide-silicon carbide (SiC/SiC) composites may be desired for several applications within the NGNP primarily because of their strength and toughness retention at high temperatures and in radiation environments. Particularly, C/C and/or SiC/SiC may be needed for the control rod sheath for a prismatic VHTR because metallic heat-resistant materials cannot withstand neutron irradiation and high temperatures found in the core. Based on information obtained from PBMR (see Section 3.3.4) this may not be an issue for a pebble bed designed NGNP; however, for a prismatic designed NGNP it is assumed that a basic GT-MHR core cross-sectional lay out would be used noted in Figure 37. This design uses 36 operating control rods and 12 start-up control rods. The start-up control rods are located in a more inner core position with respect to the operating control rods and would consequentially be subjected to higher temperatures during insertion. However, the start-up control rods are only inserted during start-up/shutdown and refueling operating modes. They are fully withdrawn whenever the core is critical and fully inserted when subcritical. The operating control rods are inserted to varying heights for control during operation for control and fully inserted for protection (scram). Therefore, the operating control rods could be subjected to temperatures greater than 1000^0C depending on the degree of rod insertion. It is assumed that the start-up control rods would never be inserted until the core had cooled well below this temperature. Therefore, the discussion regarding composite applications for control rods is directed at applications for the operating control rods for the prismatic NGNP only.

Orifices are located in the control rod flow channels to minimize bypass flow while maintaining adequate cooling for the control rods. The 12 startup control rods are located in the innermost row of fuel columns and are fully withdrawn when the core is critical. The startup control rods are used to keep the core subcritical when cooling the plant down to cold shutdown. The startup control rods are not needed for criticality control during a depressurized conduction cooldown accident because the operating control rods are inserted and the reactivity temperature coefficient is negative. Thus, there is no need to insert the startup control rods early during a depressurized conduction cooldown accident. In fact, inserting the startup control rods during the accident could damage them because of the high temperatures expected in the innermost fueled ring.

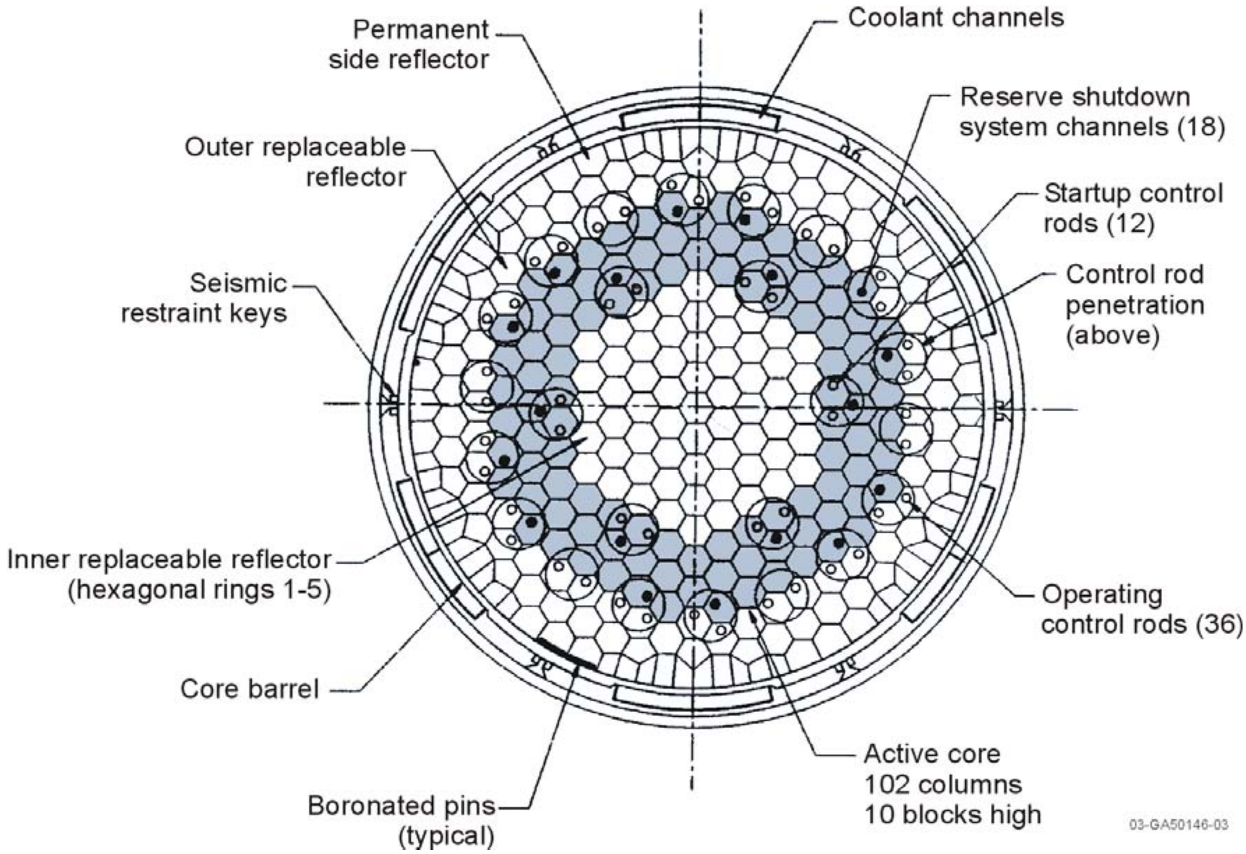


Figure 37. GT-MHR core cross-section from GA-A25401, H2-MHR Pre Conceptual Design Report[67]

Previous analyses of the prismatic VHTR were performed to determine the thermal response of the reactor vessel during a depressurized conduction cooldown accident and this analysis is discussed in Section 3.4[68]. A recent unpublished analysis given below provides further clarification regarding the temperatures of the control rods during normal and off-normal operation and hence provides further insight regarding required materials.

The model used by Gougar and Davis[68] did not explicitly represent either the startup or the operating control rods. However, nearby heat structures were modeled and their temperatures are judged to be reasonably representative of the control rods. The coolant channel surface temperature of the innermost ring of fuel was judged to be representative of the startup control rods assuming that they were fully inserted into the core. As mentioned previously, the startup control rods are not inserted during normal operation and are not required for reactivity control during a depressurized conduction cooldown accident.

The flow channels for the operating control rods are located in the outer reflector. Hence, the calculated temperature of the outer reflector would be expected to be representative of that of the operating control rods when they are inserted in the core. However, a review of the model indicated that the calculated temperature of the outer reflector would be less than expected during normal operation because of the modeling of the core bypass. The core bypass flow was modeled as two streams, one (Component 142) flowing through the inner reflector and the other (Component 145) flowing through the outer reflector. These bypass flow paths simulated the gaps between adjacent blocks in the core and reflectors. These gaps are physically distributed throughout the inner reflector, core, and outer reflector, but were modeled with only two paths for convenience. The model assumed that 9.5% of the flow bypassed the core, with about 30% of the bypass flowing through the inner path and the remainder flowing through the outer path. The outer bypass path thus represented many more gaps than those actually separating the outer fuel ring from the outer reflector. Consequently, the calculated flow through the outer bypass path was much greater than that which would cool the inner surface of the outer reflector. Furthermore, the outer bypass flow path was not thermally connected to the outermost fuel ring to prevent an artificial reduction in the fuel temperature. As a result, the fluid temperature rise in the outer bypass path was only about 10% of that across the fluid channels in the outer fueled ring. The combination of a higher flow rate and a lower fluid temperature resulted in a calculated reflector temperature that was significantly lower than that expected during normal operation.

The RELAP5-3D model of Gougar and Davis[68] was revised to provide a better estimate of the temperature of the outer reflector during normal operation. Specifically, the flow area of the outer bypass path was reduced to reflect the physical area of the gaps between the outer fuel ring and the outer reflector, and the flow area of the inner bypass path was increased to compensate. The hydraulic resistance of the bypass paths was then adjusted to obtain the same total bypass flow as in the original model. The outer surface of the heat structure representing the outermost fuel ring was also thermally connected to the outer bypass path. This thermal connection reduced the fuel temperature in the outer ring by about 10°C, but had almost no impact on the fuel temperatures in the inner two rings. As shown in Table 17, the revision to the model had almost no impact on any of the parameters except for temperature of the outer reflector at the radial centerline of the coolant channel for the operating control rods. The maximum temperature of the reflector was obtained near the bottom of the core and would be representative of an operating control rod only if it were fully inserted into the core.

Table 17. Calculated thermal-hydraulic conditions during normal operation for the prismatic VHTR.

Parameter	Original	Revised
Power, MW	600	600
Pressure, MPa	7.00	7.00
Differential pressure, MPa	0.0802	0.0802
Inlet temperature, °C	590	590
Outlet temperature, °C	950	950
Flow rate, kg/s	325	325
Core bypass, %	9.53	9.53
Maximum fuel temperature, °C	1106	1107
Maximum graphite temperature at the radial centerline of an operating control rod, °C	632	927
Maximum vessel temperature (midwall), °C	421	422
RCCS power, MW	2.13	2.15

Although the revised RELAP5-3D model better represents the temperature of the graphite at the radial location of the operating control rods, the model still does not explicitly represent the control rod geometry or operating conditions. The absorbing material in the operating control rods consists of B_4C granules that are formed into annular compacts enclosed within Incoloy 800H or carbon composite canisters[67]. The inside and outside surfaces of the canisters are cooled by the downwards flow of helium during normal operation. The radial centerline of the control rod is 8.2 cm from the interface with the outer fueled ring. The minimum distance between the control rod absorber material and the interface with the outer fueled ring is 3.2 cm. An operating control rod could generate significant power due to gamma heating and the scattering/absorption of neutrons when it is partially inserted into the core during normal operation. However, the orifice controlling the flow into the control rod coolant channel could be adjusted to obtain adequate cooling during normal operation. Thus, the details of the heat generation and the flow are assumed to be unimportant for this analysis.

The prismatic VHTR calculations described by Gougar and Davis[68] were repeated with the revised model. Figure 38 presents the calculated axial temperature profile in the outer reflector at the radial centerline of the operating control rods at four different times varying between 0 and 800 h after the start of the depressurized conduction cooldown accident. The maximum temperature during normal operation (0 h) was 927 °C. The temperature profile was almost monotonic, with the minimum temperature occurring near the top of the active fuel (TAF) and the maximum temperature occurring near the bottom of the active fuel (BAF). The temperature profile shifted during the accident so that the maximum value occurred near the peak power location which was near the axial center of the core. The maximum temperature during the accident was 1153 °C. The transient results obtained with the revised model were similar to those obtained with the original model.

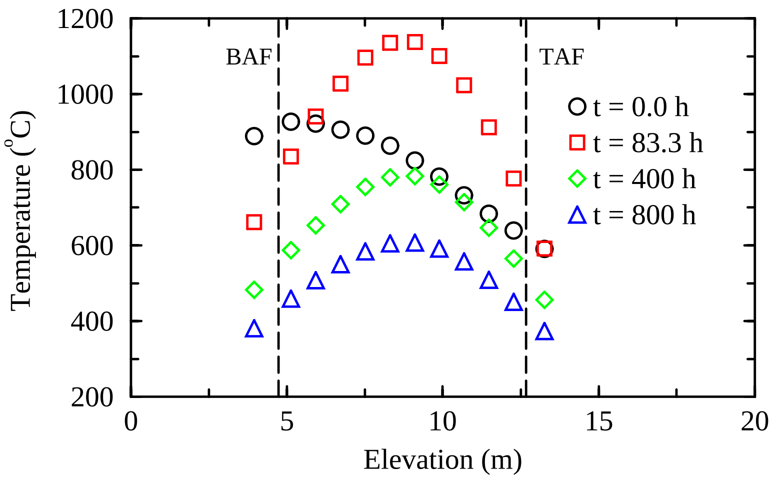


Figure 38. Axial temperature profiles in the outer reflector of a prismatic VHTR during a depressurized conduction cooldown accident.

The calculated results shown in Figure 38 are representative of the values expected for the operating control rods when they are inserted into the core. The maximum calculated temperature at 0 h exceeds the maximum allowable value in the ASME code for Incoloy 800H. However, an operating control rod is not expected to be fully inserted into the core during normal operation. Inserting the operating control rod 30% into the core yields a maximum temperature of 732 °C, which should be within the capability of metallic control rods. Furthermore, the orifice in the control rod flow channel can be sized to provide the required cooling. Thus, the use of metallic operating control rods appears feasible during normal operation. This conclusion is consistent with General Atomics' most recent design[67].

The operating control rods will be inserted into the core during a depressurized conduction cooldown accident[67]. Figure 38 shows that the temperature of the operating control rods will increase significantly during the accident. The maximum temperature of 1153 °C obtained here is reasonably consistent with the 1223 °C reported previously by General Atomics[69]. These relatively high temperatures during the accident will pose a challenge to metallic control rods. However, damage to the operating control rods during this accident might be acceptable considering the low probability of the event.

Figure 39 presents the calculated axial temperature profile for the coolant channel surface in the inner fuel ring. The maximum value during normal operation occurred near the BAF and was 1081 °C. The maximum value during the accident was 1506 °C. These temperatures are representative of the environment that a startup control rod would see if it were fully inserted into the core. However, the startup control rods are not inserted into the core during normal operation or during a depressurized conduction cooldown to preclude their damage[67]. The startup control rods are required to achieve cold shutdown and thus would probably be inserted after the core temperatures had been reduced significantly.

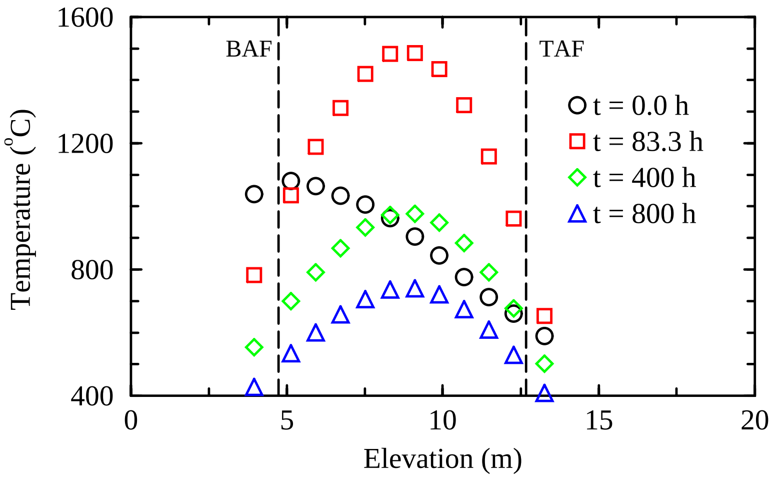


Figure 39. Axial profiles for the coolant channel surface temperature in the inner fuel ring of a prismatic VHTR during a depressurized conduction cooldown accident.

Figures 40 and 41 present the maximum temperature in the outer reflector and the coolant channel surface temperature in the inner fuel ring as a function of time. The peak calculated values occurred about 60 h after the start of the accident. The temperatures then decreased slowly for the remainder of the calculation. The outer reflector temperature decreased to 760 °C at 430 h and would reach 600 °C near 820 h. The coolant channel surface temperature in the inner fuel ring decreased to 760 °C at 748 h and would reach 600 °C around 1140 h. These results indicate that the control rods could be at elevated temperatures for more than a month if only the passive decay heat removal system is available. However, the operators could probably restore active cooling in much less time than a month. Thus, the temperatures could probably be reduced to less than 600 °C much faster than the values given above.

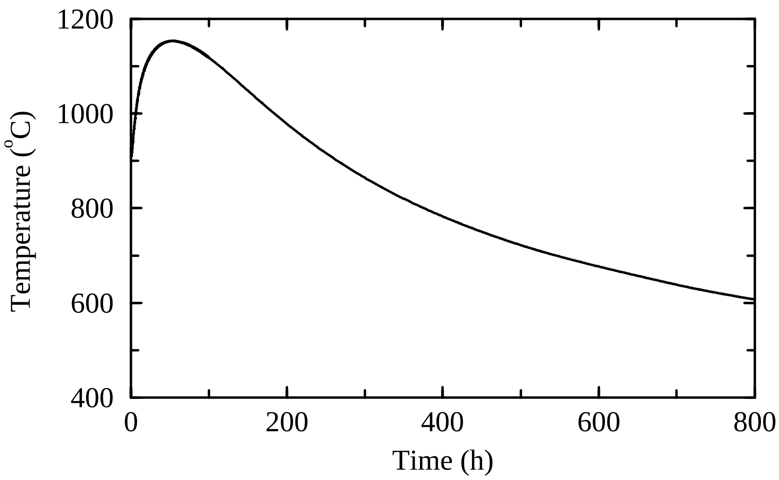


Figure 40. Maximum outer reflector temperature at the radial centerline of an operating control rod channel during a depressurized conduction cooldown accident.

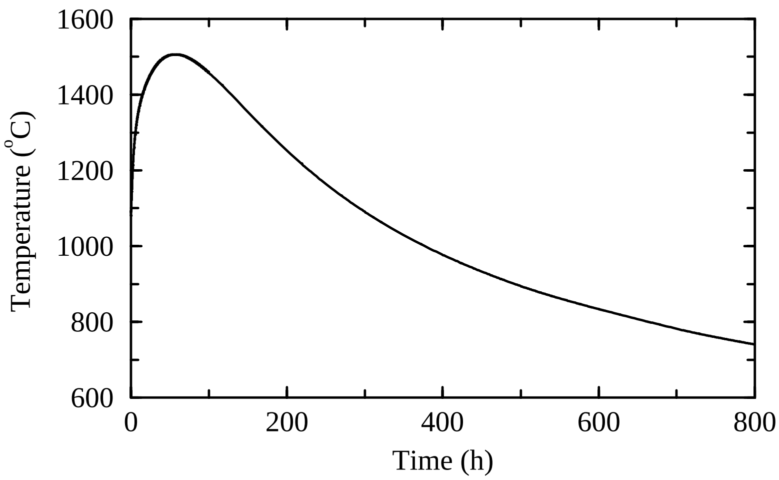


Figure 41. Maximum coolant channel surface temperature in the inner fuel ring during a depressurized conduction cooldown accident.

Figure 42 shows the calculated gas temperature in the inlet plenum of the reactor during the accident. Portions of the startup control rods would be located in the inlet plenum since the rods are not inserted into the core during the accident. Although not visible in the figure because of the scale, the gas temperature initially decreased because of the depressurization. The temperature then increased until 85 h. The maximum calculated temperature was 590 °C and occurred during normal operation. This relatively low value should prevent concerns with metallic startup control rods. However, the maximum temperature could be higher than that calculated because the model used a single control volume to represent the upper plenum and thus neglects axial and radial temperature variations. Furthermore, the temperatures in the inlet plenum would probably be higher during a pressurized conduction cooldown accident than in a depressurized accident because natural convection at higher pressure would transfer more energy from the center of the core to the inlet plenum. Nonetheless, issues relative to the temperature of the startup control rods in the inlet plenum are expected to be much less severe than those described previously for the operating control rods. Richards et al. [67] state that there would be operational benefits to using a carbon-carbon composite cladding that would allow partial insertion of the startup control rods during normal operation.

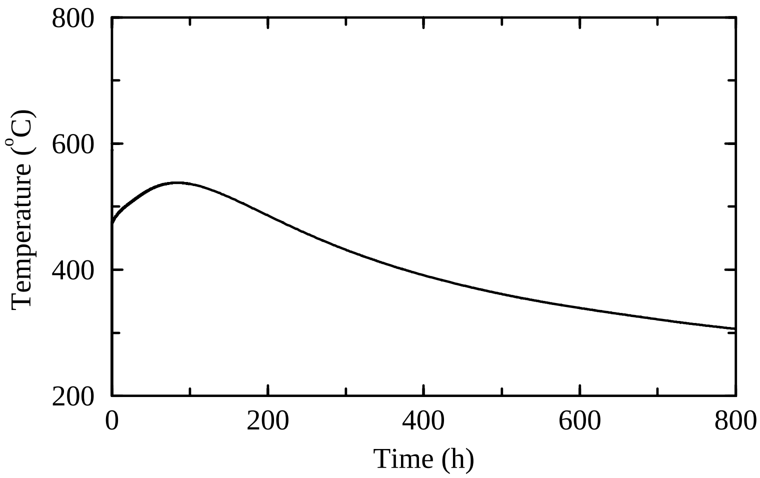


Figure 42. Average gas temperature in the inlet plenum during a depressurized conduction cooldown accident.

4.5.2 Potential Composite NGNP Composites

The anticipated composite material components of the NGNP are listed in Table 18.

Table 18. Conditions affecting materials selection for structural composites and potential candidate NGNP materials. Operating conditions given have not been verified based on a specific design

Component	Sub-components	Normal NGNP operating conditions			Abnormal operating conditions	Potential Candidate Materials
		Nominal Temp (C)	Neutron fluence E≥0.1 MeV	Medium		
CPS drive	Control rod guide tube	600 at CRD to UPS Interface.	$3 \cdot 10^{16}$ cm^{-2} per year	Helium	Working fluid temperature in cooldown mode through RCCS can increase to 1200°C within 100 h	C/C SiC/SiC
RSS drive	RSS balls guide tube					
SCS unit Structures	Conical shell at SCS HX	1000	$3 \cdot 10^{16}$ cm^{-2} per 60 years	Helium	~1200°C at start of cool down. Then ~1000°C	C/C
Hot gas duct	Outer shell of thermal insulation element unit	1000	$2 \cdot 10^{17}$ cm^{-2} per 60 years	Helium	1000°C at start of cooldown	C/C
	Inner shell of thermal insulation element unit	650			1000°C at start of cooldown	C/C

Component	Sub-components	Normal NGNP operating conditions			Abnormal operating conditions	Potential Candidate Materials
		Nominal Temp (C)	Neutron fluence E≥0.1 MeV	Medium		
In-vessel Internals	SCS entrance tubes & chamber insulation assembly	600	$2.0 \cdot 10^{17}$ cm^{-2} per year	Helium	~1200°C at start of cool down. Then ~1000°C	C/C
	Upper core restraint				~1200°C	C/C

C/C and SiC/SiC composite materials are typically designed and manufactured for specific applications and are not available off the shelf. The composite architecture (i.e., fiber type, fraction, orientation, lay-up) and processing conditions are selected to tailor the composite properties for a specific application. Thus, prototype components must be produced from which material test specimens are cut and subjected to the appropriate thermal and irradiation conditions in the materials test program.

A fortunate aspect of C/C is that existing data has shown that the C/C components that are not in the core of the reactor are not expected to experience neutron exposures high enough to cause any problems with strength, swelling, thermal conductivity, etc. Existing data has shown that C/C can easily withstand the neutron doses in all the components with the exception of the control rods that are partially inserted in the core for extended periods. The same data also shows that C/C composite, when used for high-exposure segments of the VHTR control rods, may need to be changed at least twice during the reactor lifetime. The maximum neutron dose that C/C composites can withstand is determined by irradiation experiment in a relevant condition for specific material. Exposure to very high temperatures should not be an issue in helium for the composites discussed.

There is recent evidence that control rods fabricated from SiC/SiC composites have the potential to survive for the full reactor lifetime in the high radiation environment within the core. Recent advancement in characterization and evaluation of these radiation-resistant SiC/SiC composites provides promise for the applicability of SiC/SiC to the control rod and many other in-vessel components in VHTR. Therefore, it has been considered to be highly desirable to evaluate both C/C and SiC/SiC for the control rod material. The combined use of C/C for low-exposure control rod segments and SiC/SiC for high-exposure segments will provide the permanent control rod structures which can reliably be operated during both normal and off-normal conditions. However, due to limited program funding, the development of SiC/SiC for NGNP applications has been discontinued.

A preliminary list of selection factors for primary ceramic composites candidate materials is provided in Table 19. Major technical and regulatory issues associated with the deployment of these emerging composite materials have been previously identified:

1. Verifying fabricability of reliable components including relatively large control rod sleeves and articulating joint parts

2. Effect on neutron irradiation at very high temperatures on the mechanical integrity

3. Environmental effect and lifetime-limiting issues such as creep and/or fatigue

4. Development of material / component test standards and design codes for specific applications.

Table 19. Relative strengths of candidate materials for NGNP control rod applications.

Pros	Cons
SiC_f/SiC Composites	
Good oxidation resistance	Higher cost than C/C
Higher cracking stress than C/C	Less industrial experience than C/C
Greater radiation damage resistance than C/C	Qualification – different weaves require a new qualification. ASME specification issue.
Less change-out, lasts longer	Lack of design criteria.
C_f/C Composites (Note: Replacement for super alloys. Could be used for guide tubes [~10 feet long, telescope feature] and the Upper Core Restraint structure.)	
Good material for accident situation.	More Radiation damage/shrinkage than SiC/SiC.
Eliminates metal from the core.	Qualification – different weaves require a new qualification. ASME specification issue.
Good residual properties (e.g., strength). Strength and fracture resistance is greater than graphite.	Lack of design criteria.
Metallic Superalloys	
Extended experience in industry and nuclear systems.	Severe radiation-induced degradation (embrittlement and loss of high temperature strength).
ASME design codes exist for limited, non-/low-radiation exposure use of 800H.	Limited heat resistance.
Refractory Metals, Monolithic Ceramics, Short-fiber/particulate Composites	
Good heat resistance.	Lack of ductility and/or fracture toughness.
Industrial experience for some materials.	Severe irradiation-embrittlement for refractory metals.
	Remaining uncertainty in radiation response.

4.5.3 Material and component fabrication issues

Currently there are several manufacturers of C/C composites that may be suitable for reactor-core components (pitch-based matrix with pitch-based fibers). However, these manufacturers have not qualified any of their recent high-performance materials for nuclear applications. Additionally, only limited data is available for the newest, radiation-resistant SiC/SiC composites, because these materials have undergone rapid development within the last 10 years. Therefore, significant qualification efforts will be needed for both C/C and SiC/SiC components to be used in VHTR applications.

For C/C materials, there are limited mechanical and thermal-physical property data at elevated temperatures that will need to be augmented. In addition, the manufacturers and their prime candidate materials must be examined for repeatability, quality, and eventual size of manufacture, as many of the parts will be very large. More importantly, the weave patterns have never been examined in this application. Thus, this material must be baselined to determine if the parts indeed meet the specification required for both thermal and mechanical properties. The scale-up of parts will be aided by stress-analysis codes, which are quite mature for C/C; however, the codes will need to be adapted for the specific fiber architectures selected.

The first phase of the composite R&D effort above focuses on providing verification of the viability of the composite control rod concept which may be required if a prismatic design is selected. This phase will primarily provide the engineering data which are more relevant to the reactor system design and the confidence regarding the practical reliability of the NGNP-grade C/C in the reactor environment. For these purposes, properties and mechanisms governing the lifetime and reliability will be thoroughly characterized, including aging, creep / fatigue, fracture toughness, and the irradiated lifetime envelope. Again, the effect of neutron irradiation at very high temperatures on various mechanical and thermophysical properties will be the key to understanding and evaluating the composite lifetime / reliability issues. Full-scale prototype control segments will also be fabricated with the most promising constituents / architectures for C/C, and will be subjected to the complete baseline characterization. The fabrication program is also driven by the needs for ASTM testing guidelines and the ASME code development, both of which are also essential in all R&D phases.

Reference C/C materials have been identified and designed in terms of the selection of reinforcement fibers, designing appropriate reinforcement architecture, designing appropriate fiber-matrix interphase, and selecting the matrix densification process.

The articulating joint between the control rod segments is expected to be the weakest link in the control rod system in terms of mechanical strength. A conceptual design for the articulating joint and appropriate material and component design schemes will have to be established for the composite joint. More specifically, in the FY-07 - 08 timeframe, design options for the articulating joints will be fully identified and the technical issues for individual design options will be thoroughly discussed. Furthermore, it is strongly desired that sub-sized joint components and the key component elements for one or a few most promising design concepts are fabricated and tested for mechanical properties.

For C/C, a low cost manufacturing route has recently been developed, and others continue to be developed. The most mature of these new routes has been shown to reduce the cost of these composites by almost an order of magnitude. Currently, they have been used to produce plates up to 4 feet in size and nozzles for missiles up to 2 feet in diameter. However, the technology is not completely mature and needs to be completely evaluated to ensure that the type of matrix that is developed is identical to those methods used to produce C/C that have been utilized for the past 20 years. These low cost options only need to be characterized for their viability and materials compatibility. If it is shown that they produce composites with the same microstructure of the matrix and interfaces, then they will be a viable candidate for later years during the production testing of these materials during the NGNP program.

4.5.4 Effects of neutron irradiation

The primary objective of the irradiation program is to determine the practical life limit for ideal grade C/C, under a representative NGNP-relevant irradiation environment. The Phase-I irradiation experiment of composites was initiated at the beginning of the NGNP materials R&D program. Bend bar specimens of the Hi-Nicalon™ Type-S, reference NGNP-grade SiC/SiC and the FMI-222, pitch-based graphite fiber-reinforced pitch-based graphite matrix, ideal-grade C/C composite were irradiated in the target holder rabbit capsules in HFIR. C/C specimens that received 20 dpa-C or 2×10^{26} n/m^2 (E > 0.1 MeV) of irradiation will be subjected to post-irradiation examination in FY-07. Specifically, the effects of irradiation on dimensional stability, modulus of elasticity, matrix cracking stress, and flexural strength will be determined. In case significant irradiation-induced property changes are realized, detailed analysis of the irradiation effect mechanisms needs to be performed. The irradiation data to be generated in this task will be crucial to determine the allowable neutron dose for ideal-grade C/C composites.. The conditions and materials for the Phases-I and II irradiation are summarized in Table 20.

Table 20. HFIR irradiation matrix for NGNP composite Phase-I and Phase-II irradiation campaigns.

Phase	Capsule Group	Material	Temperature(°C)	Dose(dpa)
Phase-I	BS1	SiC/SiC Type S / ML	800	10
	BS2			20
	BS3			>30
	BC1	C/C FMI-222	800	10
	BC2			15
	BC3			20
Phase-II (Note: SiC/SiC will be deleted from the matrix when practical)	TS1	SiC/SiC reference NGNP	800*T1 tbd	10
	TS2			20
	TS3			30
	TS4	SiC/SiC reference NGNP	950*	10
	TS5			20
	TS6			30
	TS7	SiC/SiC reference NGNP	1100*T2 tbd	10
	TS8			20
	TS9			30
Phase-II	TS1	C/C reference NGNP	800*T1 tbd	5
	TS2			10
	TS3			15
	TS4	C/C reference NGNP	950*	5
	TS5			10
	TS6			15
	TS7	C/C reference NGNP	1100*T2 tbd	5
	TS8			10
	TS9			15

*Primary target temperature

*T1,T2 The low and high bounding temperature, respectively, to be finalized.

It is well understood that the isotropic graphite materials exhibit non-monotonic volume change during neutron irradiation at elevated temperatures, consisting of the early contraction phase and the following swelling phase. The mechanical integrity of graphite is severely compromised as the swelling phase begins, typically ~10 dpa-C. The point at which the swelling of graphite moves through the densification stage into the swelling phase and passes through "zero swelling" into positive net swelling is regarded as the materials life. Since the lifetime fluence of C/C is substantially influenced by irradiation temperature and initial quality of the material, it is necessary to determine the swelling behavior for each C/C under consideration. Moreover, C/C composites consist of two kinds of constituents each exhibiting anisotropic irradiation-induced dimensional changes, therefore the mechanical and physical properties are obviously modified significantly before the lifetime fluence is reached. Understanding of the detailed irradiation effects in C/C composite components is essential for the reactor design and development of design codes.

Design codes for reactor in-vessel components typically require the qualification of materials in terms of post-irradiation creep, fatigue, and fracture toughness, in order to determine the safety margin during the accidental high temperature excursion events, allowable number of scram events, and the maintenance schedule requirements. Assuming that the early screening irradiation program proves the viability of composites as the control rod material, it will be necessary to perform a subsequent irradiation program for the engineering material qualification. Post-irradiation creep deformation and rupture life is the primary issue for metallic control rods for gas-cooled reactors, since the irradiation-induced transmutation and solute segregation compromises the high temperature mechanical properties of heat-resistant metallic alloys such as Alloy 800H very significantly. Although it is expected that significant insight about the irradiation effect on creep deformation will be obtained through the planned in-pile creep experiment, it is likely that the design code requires the use of creep / rupture data generated through the standardized procedure. Low cycle fatigue life will also be important because it most likely dictates the requirement of control rod replacement when scram events are experienced. Verifying the sensitivity of fracture toughness to neutron irradiation is also important as fracture toughness is the key feature of composites. Strategy for addressing these qualification irradiation effect issues needs to be developed under close coordination with the design code development and test method standardization during the next few years.

It is expected that the NGNP control rods will be subjected to low stress, long-duration tensile loads within a high temperature irradiation environment. A significant concern for these materials is creep or environmental degradation under combined load and irradiation. It may be necessary to characterize the creep behavior of composite materials in the absence of irradiation with He atmospheres containing oxygen impurity levels that bracket the expected operating conditions for the NGNP.

4.5.5 Environmental Effect and Lifetime-limiting Issues

The high-purity He environment in the NGNP, provides some interesting issues for materials degradation at high temperatures. Carburization of metals is observed in low-oxygen-potential environments but is reduced in high-oxygen-potential environments. Control of the oxygen potential is seen as an effective means of reducing carburization of metals and alloys in the NGNP but the effects of increased oxygen potential on the corrosion rates of SiC/SiC and C/C will need to be established. A focus of this research will be to determine the corrosion mechanisms and rates associated with degradation of the fiber/matrix interphase in the C/C materials. This has been shown to be the critical mechanism that shifts the degradation or failure modes from fiber creep domination to interphase degradation.

Typical simulated advanced HTGR helium chemistries used in various previous test programs include the main impurities such as H_2, H_2O, CO and CH_4. The hot graphite core is considered as reacting with all free O_2 and much of the CO_2 to form CO, and with H_2O to form CO and H_2. In addition, in cooler regions of the core, H_2 reacts with the graphite via radiolysis to produce CH_4. The overall stability of the proposed helium environment that will be representative of the NGNP must be evaluated in order to ensure that testing proposed is performed in environments that have consistent chemical potentials.

4.5.6 Standards and codes

The need for continued ASTM guideline development has been highlighted as a critical issue for both C/C and SiC/SiC composites under NGNP. Currently there are few national or international full-consensus standards for evaluating advanced ceramics and ceramic matrix composites (CMCs) in particular. Technical and pragmatic issues related to standardization efforts for CMCs must be evaluated for full consensus standards [that is, American Society for Testing and Materials International (ASTM) Subcommittee C28.07 on Ceramic Matrix Composites, Comit Europen de Normalisation (CEN)

Subcommittee TC184/SC1 on Ceramic Composites, and International Organization for Standardization (ISO) Technical Committee TC206 on Fine (Advanced, Technical) Ceramics]. This task will provide for continued involvement of key personnel involved in these efforts.

The present effort by the NGNP Composite Working Group within ASTM Subcommittee C28.07 involves the test standard development for basic fast failure modes for tubular SiC/SiC structures, such as axial and hoop tensile properties at ambient temperature. The Working Group activities include continued development of testing standards, providing guidelines to materials characterization in fabrication and evaluation tasks, and providing coordination to the round-robin and robustness testing for the precision and bias determination involving multiple institutions.

The role of the Working Group and the task scope will have to be expanded in several ways, in order to accommodate the prospective needs by the design code development. First, similar test standards do not exist for C/C composite parts. Although the venue for C/C standard development has not yet determined, it is probably most appropriately handled by ASTM Subcommittee C28.07. Second, once the test standards for fast failure strength properties are established for the ambient temperature evaluation, they need to be expanded to the evaluation at elevated temperatures. Similarly, dependable testing guidelines will be necessary for fracture toughness determination of SiC/SiC and C/C tubular structures at both ambient and elevated temperatures. Furthermore, it will be preferred that the potential time-dependent failure modes such as slow crack growth, low cycle fatigue, and creep-fatigue are evaluated following the standard procedures, most preferably the full-consensus test standards developed within ASTM. The strategic plan of test standard and/or guideline establishment needs to be developed as a part of the path-forward of standardization and codification effort during FY-06 – 07.

As noted above, structural composites are emerging materials and therefore design codes have to be developed for applications as critical components in commercial nuclear energy systems. General requirements for control rod are:

1. Subcriticalize and maintain subcriticality during normal operation and abnormal transients without exceeding design conditions

2. Shut down the reactor operation safely and reliably during normal operation, abnormal transients, and accidental incidents

3. Compensate the reactivity variation arising from variations or time-dependent changes of reactor loading, gas composition, temperature, fuel burn-up, etc.

For these requirements, design codes for the control rod may conservatively be developed along the reference ASME code case N-47 (?), because it will be sufficient for control rod structural integrity to be assessed by codes equivalent with more demanding core support structures.

The design codes must ensure that the control rod does not degrade its functionalities by repeated use for a pre-determined period when the specified operation conditions are maintained. The operation conditions are typically specified by upper temperature limit, number of scram events, and other factors. When one of these conditions is exceeded, the control rod will have to be replaced but still has to be able to shutdown the reactor safely and reliably. In order to determine the specific conditions, it is necessary to fully identify the potential accidental scenarios for excess temperature events and the consequences, and how the material properties limit the upper temperature limit and the allowable number of scram events. Therefore, interactions of the codification activity with the material / component evaluation and test standard development are critically important throughout the R&D phases beyond the fundamental viability demonstration.

The design code development for composite control rods and other in-vessel components will most appropriately be performed by the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Section III. Involvement in the ASME activity and the fundamental code development strategy will be determined as a part of the path-forward of standardization and codification effort during FY-06 – 07, however, based on preliminary information this process may take several years to complete.

4.5.7 Thermal Insulation:

High-temperature fibrous insulation is being considered for use throughout the reactor system and the power conversion unit notably in the hot duct, upper plenum shroud, SCS helium inlet plenum, and turbo-compressor. These components, and hence their material constituents, are typically designed for lifetime operation. The insulation in these components is required to retain resiliency and physical characteristics during normal operating and conduction cool-down accident conditions. Operating conditions for fibrous insulation include low neutron fluence (<0.01 dpa), moderate gamma flux, and high temperatures. The choice of fibrous insulation, as compared to forms such as blown insulation or jelly-roll assemblies of materials such as Grafoil is essentially the previous experience with Kaowool for hot-duct application at Fort Saint Vrain and other HTGR's. The currently envisioned design requires fibrous insulation to operate at normal and off-normal temperatures of up to 1000°C and 1200°C, respectively. Mechanical loads on the thermal insulation result from differential thermal expansion, acoustic vibration, seismic vibration, fluid flow friction, and system pressure changes. Typical operating conditions are listed in Table 21.

Table 21. Conditions affecting materials selection for reactor internals thermal insulation and potential candidate NGNP materials.

Component	Sub-components	Normal NGNP operating conditions			Abnormal operating conditions
		Nominal Temp(ºC)	Neutron fluence with E.0.1 MeV	Medium	
SCS unit Metalworks Insulation	Conica shell at SCS HX	1000	3×10^{18} cm^{-2} Per 60 years	Helium	-1200ºC At start of cooldown Then - 1000ºC
	Outer shell of Thermal insulation Element unit	1000			1000ºC At start of cooldown
Hot gas duct	Inner shell Thermal insulation Element unit	650	2×10^{17}cm^{-2} Per 60 years	Helium	At start of cooldown
	Thermal insulation	600-1000			1000ºC At start of cooldown
	Metal support Bottom Plate Insulation				700ºC

Component	Sub-components	Normal NGNP operating conditions			Abnormal operating conditions
		Nominal Temp(°C)	Neutron fluence with E.0.1 MeV	Medium	
In-vessel metalworks	SCS entrance Structural tubes Insulation		2.0×10^{17} cm^{-2} Per year	Helium	-1200°C At start of cooldown Then-1000°C
Insulation	Upper Plenum Shroud Insulation	600			1200°C At start of cooldown
	Ceramic Floor Block	600	4.5×10^{16} cm^{-2} Per year	Helium	600°C
	Top Insulator Block	700	1.5×10^{16} cm^{-2} Per year	Helium	1100°C
	Bottom Insulator Block	1050	1.5×10^{16} cm^{-2} Per year	Helium	600°C

Test programs to support the acquisition of design and performance data were conducted on Kaowool and Quartz/Silica fibrous mats for the earlier HTGR programs. Limited irradiation effects test data is available for those specific material grades. Tests to determine fatigue properties as a function of acoustic noise were planned but not conducted. Insulation design surveys have indicated that a suitable insulation system, where significant structural support is not required, is the use of Al_2O_3 and SiO_2 mixed ceramic fiber mats (K_{th}<0.1 W/m-K) contained between metallic or carbon-carbon composite cover plates attached to the primary structure that requires insulation. However, the operating normal and off-normal temperatures (1000 and 1200°C) are aggressive for application of the previously considered Kaowool systems. As example, the pumpable Kaowool temperature limit for continuous operation is 1093°C. Maximum temperature rating is typically 1260°C for the highest performing Al_2O_3 and SiO_2 mixed ceramic fiber mat insulation. Typically, by reducing the fraction of silica in the wool, or through simultaneous reduction of silica and addition of ZrO_2, insulating mats can achieve continuous and maximum operating temperatures of 1300 and 1400°C respectively. High-purity alumina mat can achieve operating temperatures above 1500°C. However, these higher temperature mats would not take advantage of previous data and experience gained with the Kaowool product, therefore a premium would be paid for their use. For VHTR application, fibrous insulation will require both long-term thermal stability and combined thermal/irradiation stability testing. Additionally, the canisters are in direct contact with the hottest gas conditions in the reactor. Thus, the materials chosen for these canisters will need to withstand up to 1000°C for 60 years, and up to 1200°C for up to 50 hours and then 1100°C for 100 hours during a loss of flow condition (LOFC) followed by a conduction cool-down transient. For this reason non-metallic materials such as carbon-carbon composites may be required for some of these canisters.

The insulating materials previously discussed have fairly modest mechanical performance requirements, therefore low specific density fibrous materials can be considered. However, for applications such as the top and bottom insulator blocks, the ceramic floor block, and possibly the canisters holding the fibrous insulation of the hot gas duct, the mechanical loading and need for creep resistance suggests the use of monolithic or composite ceramic materials. Typical operating parameters for these systems are also provided in Table 21. Given that the operating temperatures are modest and the neutron fluence is low, achieving a lifetime material appears a desirable, attainable goal. Graphite is a potential candidate material for both top and bottom insulator blocks. The bottom insulator block will most likely be a refractory ceramic. However, consideration will be given to improved low-thermal conductivity graphites for all three functions along with commercially available refractory ceramics such as alumina, fused silica, mullite and composite materials. The current material of choice for the bottom insulator of the South African PBMR prototype reactor is high-purity fused silica. Assuming a high-quality, high-purity commercial material, radiation effects will likely not be an issue. However, a qualification program to determine the effect of neutron irradiation on the dimensional stability under combined neutron and gamma irradiation and the effect of low-dose neutron irradiation on the thermal conductivity will be required. There are currently no insulating materials of adequate irradiation experiment to be considered qualified for bulk insulation in the core.

The properties that will drive the selection of the monolithic insulation are the non-irradiated thermophysical properties in particular: thermal conductivity, compressive strength and fracture toughness, and cost. When comparing full density brick forms of mullite, alumina, and fused silica, significant differences in properties are noted. In particular, high-density alumina brick will possess significantly higher thermal conductivity as compared to mullite (and very similar to low-conductivity graphite) but exhibit extremely high compressive stress and somewhat higher fracture toughness as compared to mullite. Creep, which will be of particular importance, will also be lower for alumina as compared to the mullite. By taking the properties of fused silica into consideration it is readily shown that, due to its very low thermal conductivity, a much thinner insulating block is required, leading to a lower cost for not only the insulating component itself, but also the overall structure as such thickness reduction also reduces the axial build of the reactor and associated structures (I.e., pressure vessel.). However, it is noted that the data-base on irradiation performance of fused silica is as immature as the other ceramics under consideration and silica suffers from radiolysis under irradiation, possibly enhancing the overall irradiation degradation of fused silica.

The first step in the research program on ceramic insulation materials for the NGNP will be a comprehensive and detailed review of the potential candidate materials identified in Table 21. This has not been done to this point in the program as the insulators in general have been regarded as of lower priority. However, in order to meet the current schedule such work should be initiated in the upcoming year. Preparation of a materials test program in support of ceramic insulation materials requires knowledge and understanding of the materials requirements dictated by the operating conditions of those components which are now available. With this information specific commercial material candidates can be chosen and their non-irradiated and as-irradiated performance determined. The data include: physical properties (heat resistance, heat conductivity and heat capacity), long term thermal and compositional stability, mechanical strength at temperature, resistance to pressure drop, vibrations and acoustic loads, radiation resistance, corrosion resistance to moisture and air/helium mixtures, stability to dust release and gas release, thermal creep, and manufacturing tolerances and mounting characteristics. The acquisition of these data requires testing of insulation specimens on small assemblies of thermal insulation panels and application of appropriate ASTM standards. As with other elements of this program, a certain level of ASTM code development will be required. Moreover, application of current non-destructive evaluation techniques, especially in support of the monolithic insulators, should be included within this test plan. Specific test rigs and facility requirements include helium flow, vibration, and acoustic test equipment as well as an irradiation facility and hot cell. Prototype assemblies testing is not planned to include neutron irradiation. However, a fundamental understanding of how critical properties such as dimensional stability, thermal conductivity, and fracture toughness (friability) should be assessed by a combined neutron and gamma irradiation study. Please note, however, that none of this work is currently integrated into the FY-07 proposed program due to lack of funding.

4.6 Molten Salt/ Metallic Alloy Interactions at Very High Temperature

4.6.1 Introduction

The interaction between molten salts and metallic alloys at very high temperatures is not currently a part of the NGNP Materials Program; however, a discussion of this topic was determined to be appropriate for the program plan because there has been much discussion and interest in this area. This interest has resulted because high temperature heat transfer to vaporize and decompose the thermochemical cycle process fluid will require an efficient and corrosion resistant heat exchanger. Two candidate working fluids have been considered: helium and molten salts. Helium poses relatively better known compatibility issues and requires higher pumping powers and pressures than molten salts. While the compatibility issues associated with high temperature helium are being addressed for the NGNP; work in the current programs has not addressed issues associated with molten salts, which provide for much better heat transfer characteristics, lower pumping powers, and allow for the use of low pressure loops. Previous programs established Hastelloy-N for use with fluoride salts at temperatures up to 750°C and for a minimum service life of 30 years. The use of molten salts at temperatures greater than 750°C introduces new performance requirements, which results in its own set of materials compatibility issues at these temperatures. The candidate salt coolants also possess relatively high (>350°C) melting points that must be accommodated by the system design. For example, LiFNaF-K melts at 454°C, KF-ZrF4 melts at 390°C, and LiF-BeF2 melts at 460°C [70]. In order to determine the viability of molten salts as a heat transfer medium for nuclear hydrogen production, the materials compatibility and mechanical properties issues of possible structural materials must be addressed. The program would need to address at least the following issues:

1. Identify candidate structural materials (alloys, composites, other advanced materials) for a molten salt intermediate loop at temperatures in the range of 750 to 950°C.

2. Review available compatibility and mechanical property data to determine current limits of performance for candidate salts.

3. Select the most promising structural materials for the molten salt application.

4. Develop a preliminary test matrix to for evaluation of these materials.

5. Perform the evaluation and down-select material(s).

6. Perform long-term testing of down-selected material(s).

Because the expense and time required to complete a program of this scope are significant, it has been decided that this type of effort will not be started until it is known to be a NGNP program requirement. Further insight on this issue is expected following review of proposals associated with preconceptual engineering services in support of the NGNP

4.6.2 Long History of Molten Salt Work

After several years of research and development (R&D) activities on molten salt reactor systems, the Aircraft Reactor Experiment began operation in 1954 at the Oak Ridge National Laboratory (ORNL) to demonstrate the feasibility of operating a molten-salt fueled reactor at high temperature; fuel entered the core at 649°C and exited at 816°C and the reactor power level was 2.5 MW. Peak salt temperatures during operation reached ~850°C, and this reactor concept was design for a short period of continuous operation (~ 1 month). A second demonstration, the Aircraft Reactor Test, was under development for several years before it was discontinued upon termination of the Aircraft Nuclear Propulsion (ANP) Program in 1957-58. Subsequently, the Molten Salt Reactor Experiment (MSRE) was carried out at ORNL to demonstrate that the desirable features of the molten salt concept could be incorporated into a safe and reliable civilian power reactor. The 7.4 MW(t) MSRE, which operated at 549 to 649°C went critical in 1965 and was shut down in 1969 after a successful operating history. Because of the success of the MSRE, additional R&D was carried out in support of a molten salt breeder concept that was designed for a peak temperature of 705°C.

Areas such as salt chemistry, including phase behavior, purity requirements, radiation stability, etc. and materials for salt-containing structures, including physical and mechanical properties, corrosion, irradiation effects, etc. were thoroughly investigated. The majority of these studies were in support of applications at ≤700°C; however, the available data base, especially the earlier studies in support of the ANP program, affords useful technical guidance for applications at >700°C.

Based on early R&D, the nickel-base alloy Hastelloy N was developed by ORNL for use in the MSRE. Several hundred thousand hours of corrosion experience with Hastelloy N and fluoride salts were obtained in thermal convection loop tests[71,72,73] and pumped loop tests[74] in addition to experience from the MSRE. Corrosion studies also included graphite as well as stainless steels and some refractory metal alloys, (e.g., TZM).

4.6.3 Mechanisms of Corrosion

Two general mechanisms of corrosion, metal dissolution and oxidation of metal to ions, can occur in molten salt systems. Because of low solubilities of most structural metals in salt systems, one mechanism, metal dissolution due to solubility in the melt, is not a common form of attack. The second mechanism, oxidation of metal to ions, is more likely. In addition to anodic dissolution and cathodic reduction of an oxidant, in the salts that are electronic as well as ionic conductors, reduction reactions can occur in the melt as well as at the metal-melt interface. In many molten salt systems the rate controlling step is ion diffusion from the alloy into the bulk solution, rather than charge transfer.

Thermodynamic stability of the fluoride salt components versus alloy constituents is quite important because molten-salt corrosion is usually induced by reduction/oxidation (redox) reactions. In work leading to operation of the MSRE, ORNL demonstrated the excellent compatibility of Hastelloy N with fluoride salts containing LiF, BeF2, ThF4, and UF4. If the salt is pure and the metal clean, UF4 is the strongest oxidant in a fuel-salt system. The reaction,

$$\text{UF4 (d)} + \tfrac{1}{2}\,\text{Cr (s)} \leftrightarrows \text{UF3 (d)} + \tfrac{1}{2}\,\text{CrF2 (d)}$$

has a favorable free energy of formation at 500 - 800°C. Since CrF2 is one of the most stable structural metal fluorides, Cr can selectively react with extraneous (impurity) oxidants (fluorides) in the system. Because of the unusually high stability of alkali metal and beryllium fluorides, the corrosion potential of melts without uranium is likely controlled by redox equilibria involving impurities in the melt or gas phase, or a soluble redox buffer (M3+/2+) that is specifically added to the system. Gradients in the chemical activities of constituents caused by temperature differences can result in dissolution of metal in one region of the system with subsequent deposition in other portions of the system. The amount of attack will depend on the driving force and dissolution or deposition kinetics for reactions that result in transporting the corroding species to a different part of the loop circuit where they might deposit. In single material systems (e.g., capsule tests) at constant temperature, there would be no activity gradient and the amount of attack would be a function of the relative solubility of the respective corrosion product in the salt or, if the product is volatile, the partial pressure in the atmosphere immediately above the salt. Thus, equilibrium solubility principles would limit the amount of solute and degree of corrosive attack in this type of system. However, a non-isothermal system could also be subject to corrosion from thermal gradient mass transfer if the chemical potential of the corrosion product fluoride, at a given concentration, is a strong function of temperature, and deposition of the corrosion species occurs in the cooler regions of a loop operating with a relatively high temperature differential. Although thermodynamics limits the amount of solute in a specific volume of salt in a closed system (capsule test) at constant temperature, continued transport of material can occur to the cold region of a non-isothermal system.

Mass transfer can also occur when dissimilar structural materials are included in the same system. The dissimilar materials do not have to be electrically coupled. Two things are required for dissimilar material mass transfer to be a factor. First, an element contained in one of the materials has to have a strong tendency to form an alloy or compound with the second material. More importantly, an element in one of the materials must be subject to oxidative attack or suffer dissolution within the salt solution. The oxidative attack can be either by reaction with impurities or, if a reactive element, with the salt constituents. In either event, once the element goes into solution, it can then migrate through the solution and form the required product if the chemical driving force (activity gradient) is sufficient. If the product does form, this will allow the mechanisms of corrosive attack or dissolution to continue (i.e., favorable thermodynamics) although kinetics may be limited by solid state diffusion. If the attack is solely by reaction with impurities, the problem may dramatically slow with time. However, the reaction will not stop if there is solubility in the salt for a specific element.

In the case of previous fluoride corrosion studies at ORNL, the containment system was generally monometallic, and dissimilar metal effects were not encountered. However, graphite was a major nonstructural component in the reactor system and there was no evidence of graphite interaction with the structural materials at Molten Salt Reactor (MSR) temperatures (≤700°C). However, for higher temperatures and materials other than Hastelloy-N, there is no data.

Even if redox reactions with major salt constituents are limited, materials can interact with oxidizing impurities present in the salt or in the system (capsule or loop). The presence of HF within the salt, which is used for impurity removal in the purification process, as well as moisture or oxides residing on the container or loop walls, can all lead to adverse corrosive reactions. Normally HF is completely removed during the purification process, and its presence in the salt is due to contamination of the salt at high temperatures with moisture. HF can react either directly with the material forming a fluoride as noted in Equation (1) or indirectly with oxides present on the wall of the container as noted in Equations (2-4). All of these reactions should cease when the impurities have reacted.

$$\text{HF} + \text{metal} \rightarrow \text{metal fluoride (dissolve in melt)} + \text{H2} \tag{1}$$

$$\text{HF (gas)} + \text{oxide (wall)} \rightarrow \text{H2 (gas)} + \text{F (wall)} \tag{2}$$

$$\text{H2O (gas)} + \text{alkali F (melt)} \rightarrow \text{HF (gas)} + \text{alkali OH (melt)} \tag{3}$$

$$\text{Alkali OH} + \text{metal} \rightarrow \text{alkali oxide} + \text{metal OH} \tag{4}$$

As alluded to earlier, oxidation of one of the constituents of the material by the reduction of a less stable impurity metal fluoride initially in the salt such as iron or nickel fluoride can occur[71,72]. Examples of this include:

$$\text{Cr} + \text{FeF2} \rightarrow \text{CrF2} + \text{Fe} \tag{5}$$

$$\text{Zr} + \text{2FeF2} \rightarrow \text{ZrF4} + \text{2Fe} \tag{6}$$

In order to control the redox potential of the fluoride salt in the MSRE, the melt chemistry was made slightly reducing by the addition of a small amount of beryllium metal that reduced approximately 1% of the UF4 solute to UF3:

$$Be + 2UF4 \leftrightarrows BeF2 + 2UF3 \tag{7}$$

As a result of this addition, the 100/1 ratio of UF4/UF3 minimized the corrosion of Hastelloy N by UF4 (where Cr is the most active metal present and therefore the first to be oxidized) because of the equilibrium between the chromium and uranium components:

$$Cr + 2UF4 \leftrightarrows CrF2 + 2UF3 \tag{8}$$

The redox potential of systems that do not contain uranium may be controlled in another fashion – by addition of a small amount of solute that can exist in two oxidation states, by addition of a sparge gas that is set at the proper redox potential, or by electrochemically setting an applied potential equivalent to the desired redox potential. These approaches need to be evaluated.

The results of ORNL's work for the MSRE showed that Hastelloy N corrosion could be characterized as void formation in the base metal due to chromium diffusion to the salt metal interface where it was oxidized by impurities in the salt or by UF4. The latter effect, oxidation by UF4, was controlled in the MSRE by the addition of beryllium metal as a means to control the UF4/UF3 ratio, as previously explained using Equations 7 and 8. The results of this work generally indicates that nickel-based alloys with the required mechanical properties are the materials of first choice to be evaluated for use in any molten fluoride salt environment.

4.6.4 Candidate Materials for Use with Molten Fluoride Salts

Based on past experience, an initial candidate list of materials for evaluation in the chosen heat fluoride salt is proposed. This list is presented in Table 22. Alloys high in chromium content (like Alloy 617) may only be viable in very low impurity content salt that is maintained at the proper redox (reducing) state.

Table 22. Potential candidate materials for use in molten fluoride salts

Material	Fluoride salt corrosion resistance	Fabricability	Highest use temperature °C	Comment
Haynes 242	Very good	Good	540	
Hastelloy N	Excellent	Good	800	
Hastelloy X or XR	Needs evaluation	Good	900	May need a protective coat
Inconel 617	Needs evaluation	Good	900	ASME Sect. VIII Div. 1 May need a protective coat

Material	Fluoride salt corrosion resistance	Fabricability	Highest use temperature°C	Comment
Haynes 230	Fair	Fair	899°C	ASME Sect. VIII Div. 1 May need a protective coat
Haynes 214	Very good	Poor-fair	1000	
MA 956	Good	Poor-fair	1300°C	No longer in production. May need a protective coat
MA 754	Very good	Poor-fair	1177°C	No longer in production
Hastelloy B3	Needs evaluation	?	?	Service temperature limited by brittle intermetallic formation. Use below 750°C.
Cast Ni superalloys	Very good		>900°C	For pumps. May need protective coat
Nb-1Zr	Very good		>900°C	
SiC (CVD)	Very good	poor	Above that needed	Need a near net shape and the ability to seal

4.6.5 Testing

Because of the lack of data for materials compatibility with molten fluoride salts and candidate alloys at the temperatures of interest, an extensive test program is needed if a molten salt heat transfer system would be used for even a limited extent in the design of the NGNP These fluoride salts must be carefully purified to avoid reactions of the structural materials of interest with impurities and techniques must be developed to perform real-time monitoring of the salt. Once a subset of these materials is determined, their mechanical properties at temperature, if unknown, would need to be determined. Those that possess required compatibility and mechanical properties would need to be evaluated in flowing thermal-convection loops in order to determine mass transfer effects. These proposed tasks are separated in three areas. This work is currently not in the NGNP Materials Program for FY-07.

4.6.5.1 Salt chemistry Measurement and Control. Thermodynamic stability of the fluoride salt components versus alloy constituents is quite important because molten-salt corrosion is usually induced by reduction/oxidation (redox) reactions. As a result, an alloying element will tend to go into solution if it forms a more stable species than those that exit in the melt. Hence, impurities present in the salt, or the gas phase, as well as moisture or oxides residing on the containment walls, can all lead to adverse corrosive reactions. Thus it is necessary that the concentration and identity of impurities in the melt be well characterized so that:

1. Adverse reactions from impurities do not cloud an understanding of the materials salt interaction.

2. Electrochemical and spectro-chemical instrumentation can be further refined to allow for real-time monitoring of corrosion effects in an operating system and to provide an understanding of the corrosion mechanism (controlling factors).

3. On-line, side-stream clean-up of salts in an operating reactor can be performed efficiently and effectively.

4. Techniques be implemented that may allow for control of redox potential of the salt and hence the corrosion rate of structural materials.

4.6.5.2 Identification of Monolithic Material and Clad-Material Candidates. There is no well characterized material for operation above 750°C. Materials for service are required especially for pumps, because a "low wear" contact surface material would be required. There are needs for:

1. Cast high-nickel material with/without a hard surface coating.

2. Coatings (cermets, carbides) for pump bearings immersed in salt and valve seats.

3. Graphite, and C/C and SiC/SiC composites for use in the presence of alloys in the salt.

Corrosion effects must be evaluated so that:

1. Identity and concentration of alloy element migrating into the salt can be ascertained (thermodynamic stability effects).

2. Effect of dissimilar materials (activity gradients) in the salt on the corrosion of each other can be ascertained.

3. Identity of diffusing species from a high strength substrate through a corrosion resistant cladding can be ascertained and efficacy of possible diffusion barriers determined.

4. Down selection of materials of appropriate strength and identification of alloying elements that must be controlled can be performed.

4.6.5.3 Behavior of Materials Under Flow Conditions. Because an operating system is not isothermal, corrosion reactions driven by thermodynamic considerations or impurity reactions, which can have different solubility limits at various temperatures, can have additional impacts. Alloying elements, which can go into solution at one temperature and come out of solution at a different temperature can build up in critical flow paths and create significant porosity at other sites. This is particularly applicable to piping and heat exchangers. Hence evaluation of temperature gradient assisted corrosion effects would need to be initiated.

4.6.6 Metallurgical and Mechanical Issues

In addition to the major compatibility issues, the candidate structural materials listed in Table 22 are expected to be exposed to temperatures up to 950ºC under certain loading conditions. Therefore, common issues for high temperature applications must also be considered. The consideration should include the highest service temperature and strength, aging effects on microstructural stability and mechanical properties, effects of product form and grain size, manufacturability, and ASME codification.

For NRC review/license application, defect-tolerance and flaw assessment should also be investigated. If the candidate material is already considered for metallic reactor internal or helium-helium IHX, most of the metallurgical and mechanical property issues should have been covered in as a part of the NGNP Materials Program. Therefore, only limited additional information will be required for the molten salt IHX application, which may include optimum metallurgical conditions for improved performance specific to molten salt IHX design, microstructural stability and mechanical properties degradation in molten salt environment. For candidate materials that are not covered in the program, all the issues related to high temperature services must be addressed. To efficiently study the metallurgical and mechanical fitness for molten salt IHX application, the investigation and testing should be closely coordinated with activities addressing corrosion and compatibility to provide adequate information for eliminating unqualified materials at the earliest stage.

4.6.6.1 Maximum Service Temperature. Most of the candidate materials listed in Table 22 have demonstrated high temperature service capabilities in air environment. Alloy 617, Haynes 230 and Hastelloy X are good to at least 900ºC. Hastelloy N is relatively weak but may be usable for some applications at temperatures less than 800ºC. Haynes 242 and Hastelloy B3 are limited in service temperature by the formation of a brittle intermetallic phase [Ni4Mo, Ni2(Mo,Cr)]. Properties at high temperature are significantly reduced and room temperature toughness drops to nil following formation of this phase. The MA alloys are strengthened by dispersed oxides that are insoluble in the metal matrix at high temperatures, and possess good strength at high temperatures well above the required 950ºC. The cast Ni superalloys and Nb-1Zr alloy have good strength properties to above 900ºC. It must be noted that there is a difference between the maximum temperature at which a material has been tested, the maximum temperature the material can serve with reasonable strength, and the maximum temperature the material is codified for certain applications.

4.6.6.2 Manufacturability. For molten salt IHX application, the major manufacturability issues that should be considered include formability and weldability. The workability of Haynes alloys, Alloy 617 and Hastelloys are quite good. All can be produced as foil, sheet, tubing, and piping. However, Haynes 214 may have some formability issues that must be investigated. MA alloys are specialized alloys produced by powder metallurgy processing, and have been produced as foil, sheet, and tubing. Depending on the processing, some MA alloy product forms such as extruded tubes may be anisotropic and creep strength is significantly weaker in the hoop direction due to longitudinally elongated grain structure. Because of the mechanical alloying process, manufacturing of MA alloys is complicated and very expensive. The major vendor, Special Metals Corporation, has stopped the production of MA 956 and MA 754. However, similar alloys, PM 2000 and PM 1000, are available from Plansee. The Nb-1Zr alloy could be produced as foil, sheet, and tubing forms and will be very expensive.

Weldability of some of the candidate materials listed in Table 22 is a significant issue. The Haynes alloys, Alloy 617 and Hastelloys are quite weldable with the exception of Haynes 214, which has a problem with cracking in the heat affected zone due to the high aluminum content. The MA alloys possess good oxidation and high temperature properties but are notoriously difficult to weld. All fusion welding methods, which are the major process used for metallic components construction, inevitably destroy the fine dispersion of the strengthening oxides and significantly decrease the high temperature strength of the weld. High energy intensity welding methods such as laser beam welding have been investigated to join MA alloys but the success was very limited because no matter how narrow a fusion zone achieved, disturbance of the fine oxide dispersion in the melted metal during the welding process can not be avoided. Solid state welding methods have the potential to solve the joining problems of MA alloys. Among these techniques, inertia welding and friction stir welding are the most promising. Both methods are currently being investigated for joining MA alloys, particularly MA 956. The inertia welding has been used successfully to weld 50.8-mm diameter MA 956 tubes for the Fossil Energy Program, and friction stir welding has successfully produced both butt and lap welds of 1-mm thick MA

956 sheets for the Gen IV Program in FY06. The weldability of the cast Ni-based alloys is not good either. In addition to the common welding problems in all the cast metallic materials, the high alloy contents make the welding more difficult. The Nb-1Zr alloy is also difficult to weld. Usually, it is joined by an Electron Beam process. However, there still tends to be an aging embrittlement issue.

4.6.6.3 ASME Codification. At present, the construction rules for the intermediate heat exchanger of the NGNP have not been identified. Possible codes that cover high temperature components include ASME VIII-1, III-NH, B31.1, and B31.3, to name a few. Regardless of which construction code will be applicable, the materials must be approved by the Code committee and listed in ASME Section II for construction. The allowable stresses or stress intensities are provided in Part D of Section II or in ASME Code Cases that are associated with specific construction code. Of the candidate material listed in Table 22, several have reached code status. These include:

1. Haynes 242
2. Hastelloy N
3. Hastelloy X
4. Haynes 230
5. Hastelloy B3/B2 (N10675/N10665)

None are included in ASME Section III Subsection NH. The status of several alloys is provided in Table 23. It should be pointed out that although the Code temperature limit may be low, data for several alloys extend well beyond the Code limit.

Table 23. Present ASME Code status of the candidate materials including some considered for molten salt IHX applications

Material	UNS	ASME B&PV Code Section / Temperature Limit (°C)	
718	N06718	VIII-1 / 621	III-NH / 550
N	N10003	VIII-1 / 704	
617	N06617	VIII-1 / 982	I / 899
X	N06002	VIII-1 / 899	
242	N10242	VIII-1 CC2319 / 538	
B2	N10665	VIII-1 / 427	III / 427
230	N06230	VIII-1 / 899	I / 899

5. Materials Program

This section contains a discussion of the NGNP Materials Program workscope performed in FY-05 and FY-06 and planned activities for FY-07 and years beyond FY-07. The planned workscope is currently uncertain, therefore, the discussion will be based on activities that the NGNP Materials Program believes to be required to support the Preliminary Project Management Plan discussed previously. The NGNP Materials Program continues to evolve and program direction, as noted in the discussion below, reflect several changes in issues and needs from FY-06 to FY-07. The primary programmatic changes that have been made and a discussion of the possible effect of the selection of the NGNP design and a vendor team on materials issues discussed previously are summarized below:

1. As noted in Section 4.1, commitment to the planned series of graphite irradiations in the ATR and HFIR extends only to AGC-1(ATR) and to HTV-1 and 2 (HFIR). Planning for other irradiations in this series is currently on hold pending selection of an NGNP reactor vendor and subsequent discussions with that vendor.

2. As noted in Section 4.1, the qualification of only two graphites will proceed in FY-07. These graphites are PCEA (prismatic) and NGB-18 (pebble bed). Following the selection of the actual NGNP design and the reactor vendor, one of these graphites will be dropped from the program and development will continue only on the graphite selected for the NGNP design to be implemented.

3. As noted in Section 4.1, it is now recognized that qualification activities for graphite can not be completed by the beginning of final design based on the Option 2 approach in the Preliminary NGNP Project Management Plan. Therefore, core graphite for the NGNP will be part of a provisional license for less than the expected total graphite irradiation dose and graphite irradiations are expected to continue into the NGNP construction phase. Following completion and analysis of all graphite irradiations, further licensing discussions with the NRC are envisioned.

4. As noted in Section 4.1, a number of graphite related code and standards activities will be accelerated to support licensing activities.

5. As noted in Section 4.1, a new activity to identify the disposal issues related to the C-14 burden in irradiated graphite removed from the NGNP will be initiated.

6. As noted in Section 4.2, a materials program for qualification of potential reactor pressure steels is not envisioned. This is primarily due to a lack of funding resources to perform this work and the knowledge that much of the work required for the qualification of Grade 91 steel is currently being performed by AREVA. Therefore, in FY-07 it is envisioned that the selection of the steel to be used for the Class 1 boundary will be made based primarily on the reactor design and vendor team selected for the NGNP.

7. As noted in Section 4.3, a limited qualification program for two potential metallic alloys (Alloy 617 and Alloy 230) for very high temperature applications will be continued. These alloys are primarily directed at the development of the IHX. As noted in item 8 below, it is envisioned that the selection of the reactor design and the NGNP vendor team will largely resolve the issue of the size and type of the IHX required for the NGNP and therefore will indicate the extent and focus of future qualification activities in this area.

8. As noted in Section 4.4, the time required to resolve of the design methodology issues noted is long and appears to be greater than the time allowed to support the Option 2 approach. Therefore, in FY-07 an approach will be developed based on the proposals received and the results of the ANL study to adequately define the risk issues noted for the IHX. It is envisioned that the approach adopted will support the Option 2 schedule, reconcile the code and qualification issues noted for the possible construction materials and integrate the direct/indirect cycle risks into the NGNP planning process.

9. As noted in Section 4.5, a limited program directed at the further qualification and codification of C/C composites for potential applications associated with the NGNP control rods is planned. It is envisioned that continued effort in this area will be strongly influenced by the results of a study on the use of composites for critical NGNP core applications and the reactor design and vendor team selected. It is, however, unlikely that C/C materials will initially be used in the core of the NGNP in critical application because of the lengthy process and time required for the development of appropriate ASTM standards and adoption of an ASME codification process for these materials. Based on prior reactor licensing experience, it is unlikely that these materials will be accepted for use in the core of the NGNP by the NRC without completion of the ASME Section III process for the use of these materials. The current alternative to the use of C/C materials for control rod components is Alloy 800H, however, as noted in Section 4.5.1, this material has potential issues for a prismatic designed VHTR system. Therefore, if a prismatic design for the NGNP is selected, the resolution of this issue will become a priority.

10. As noted in Section 4.6, a qualification program to investigate the corrosion effects of the potential use of molten salt heat transfer systems in the NGNP is not planned. It is improbable that these systems will be proposed for the NGNP but may be considered for retrofit in the future following plant construction for specific applications.

5.1 Discussion of FY-05 and 06 Materials Program

5.1.1 Graphite

Graphite will be the primary structural component in the core of the NGNP. Graphite will be used as the primary core structural component for any thermally moderated NGNP design envisioned. The graphite used for the Fort St. Vrain (H-451) is no longer available and one or more new nuclear graphites require qualification. This section describes the effort performed in this area in support of the NGNP to date.

5.1.1.1 Graphite Selection Strategy. A graphite selection strategy was developed by interaction and discussion with the GIF and described in the Graphite Collaboration Plan developed by the GIF VHTR Materials and Components Project Management Board. The Graphite Collaboration Plan describes the activities being conducted internationally to develop a design database for the NGNP and other VHTR concepts. Moreover, the graphites being used by the GIF partners in their international programs and the selection strategy developed are identified in the collaboration plan.

As a result of the GIF plan, GIF representatives met at the facilities for SGL Carbon and Graftech in January 2005 to review the work on developing new nuclear grade graphites. At the meetings with SGL and Graftech, the new graphite grades NBG-18, NBG-17, and PCEA were discussed.

It was agreed that one or more billets of the following graphite grades be purchased for inclusion in the graphite program. The graphites purchased are categorized into major, minor, and experimental grades. The graphites in these grades are given in Tables 24 and 25.

Full sized billets of major grades were purchased and smaller portions of billets were either donated or purchased for the minor grades.

Table 24. Major Grade Graphite.

Grade	Manufacturer	Comments
PCEA	Graftech International	Extruded, candidate for high dose regions of VHTR concepts
NBG-17	SGL	Vibrationally molded, candidate for high dose regions of VHTR concepts
NBG-18	SGL	Vibrationally molded, candidate for high dose regions of VHTR concepts
IG-430	Toyo Tanso	Vibrationally molded, candidate for high dose regions of VHTR concepts

Table 25. Minor Grade Graphite.

Grade	Manufacturer	Comments
PGX	Graftech International	Molded, candidate for low dose regions of VHTR concepts
PCIB	Graftech International	Isostatically molded, candidate for low dose regions of VHTR concepts
NBG-10	SGL	Extruded, candidate for low and high dose regions of VHTR concepts
NBG-25	SGL	Extruded, candidate for low dose regions of VHTR concepts
HLM	SGL	Extruded, candidate for low dose regions of VHTR concepts
2020	Carbone of America	Isostatically molded, candidate for low dose regions of VHTR concepts
PPEA	Graftech International	Extruded, candidate for low dose regions of VHTR concepts

5.1.1.2 Irradiation Experiments. Graphite from these billets was used for precharacterization activities and production of irradiation samples for AGC-1 and HTV1 and 2. Small amounts of other graphite grades were also selected for insertion into AGC-1 for specific purposes. The actual graphites selected for AGC-1 are given in Table 26. A report on graphite testing and qualification specimen selection strategy was written[75].

Table 26. AGC-1 graphite materials test matrix

Graphite	Reactor Vendor	Proposed Use	Capsule Location	Remarks
H-451	General Atomics	Prismatic fuel element and replaceable reflector	Creep	Historical Reference Only a few samples
IG-110	JAERI, INET	Prismatic fuel element, replaceable reflector, and core support pedestals Pebble bed reflector	Creep	Historical Reference Only a few samples Currently being used in the HTTR and HTR-10

Graphite	Reactor Vendor	Proposed Use	Capsule Location	Remarks
PCEA	AREVA	Prismatic fuel and replaceable block	Creep	AREVA wants to construct the entire graphite core out of the same graphite
NBG-18	PBMR AREVA	Pebble bed reflector structure and insulation blocks Prismatic Fuel element and replaceable reflector;	Creep	Candidate for PBMR replaceable reflector
NBG-17	AREVA PBMR	Prismatic Fuel element and replaceable reflector Pebble bed reflector structure and insulation blocks	Creep	AREVA wants to construct the entire graphite core out of the same graphite. NBG-17 is finer grain than NBG-18
IG-430	JAERI	Prismatic fuel element, replaceable reflector, and core support pedestals	Creep	JAERI wants to use this graphite in the GTHTR 300
HLM		Prismatic large permanent reflector	Piggyback	Fort St. Vrain permanent reflector. Similar to PGX
PGX	AREVA JAERI	Prismatic large permanent reflector	Piggyback	AREVA may use this material; preference is to use PCEA or NBG-17 for Permanent reflector. HTTR permanent structure.
NBG-25		Core support candidate	Piggyback	Isostatic fine grain
2020		Prismatic core support pedestals and blocks	Piggyback	Fine grain isotropic NPR candidate material
PCIB		Core support candidate	Piggyback	Fine grain isotropic
BAN			Piggyback	Experimental graphite with potentially superior irradiation life
NBG-10	PBMR	Prismatic Fuel element and replaceable reflector Pebble bed reflector structure and insulation blocks	Piggyback	PBMR's original choice for replaceable reflector Price/performance will be the basis between NBG-18 and NBG-10
PPEA		Needed to provide comparison with PCEA	Piggyback	Provides direct comparison of pitch coke and petroleum coke graphite performance
HOPG		Needed to determine change in crystalline structure	Piggyback	Provides insight to single crystal changes during neutron irradiation
A3 Matrix		Needed to determine fuel compact irradiated material behavior	Piggyback	Provides dimensional change and thermal conductivity data for matrix materials

5.1.1.2.1 AGC Irradiations — A report documenting the initial experimental plan and design for the AGC-1 was written in FY-05[76,77]. An updated report[78] was written in FY-06 that included design changes to the experiment required for irradiation in the South Flux Trap of the ATR. It was determined in FY-06 that the AGC-1 gas system design used to control the experiment in the ATR and the fabrication of the AGC-1 capsule could not be completed in FY-06 as originally planned. This will cause a delay for insertion of the AGC-1 that will be discussed in Section 5.2.

A report[76] written in FY-06 contains the latest update regarding the objectives, technical plan, design, and schedule for the AGC-1 irradiation experiment.

The data contained in this experiment are critical to the design licensing of the NGNP graphite components and support ongoing work in the area of model development, e.g., irradiation effects model such as dimensional change and creep strain, structural modeling, and fracture modeling. Moreover, the data will be used to underpin the American Society of Mechanical Engineers (ASME) design code currently being prepared for graphite core components.

The report reviews the background and theory of irradiation induced creep in graphites, and details the graphite grades to be irradiated in the experiment along with the rationale for their inclusion. Details of the irradiation test conditions are reported. Detailed AGC-1 layout plans are given for each of the specimen channels in the capsule, and the specimens are tabulated by grade, location, and anticipated fluence. The process of pre and postirradiation examination is reported, and details of the tests to be performed and the data to be acquired are given.

The second half of the report details the design to date for the AGC-1 capsule. Detailed discussions are given for the capsule requirements and how the design incorporates those requirements. The experimental plan is fully developed because all details and theory are known at the present time. However, this report is a snapshot of the AGC-1 design. A final design review will occur later in FY-06 and the outcome of this design review could result in further required changes to the AGC-1 design. Capsule insertion in the ATR is expected in November 2007. The final design will use an assumed neutronic model of the ATR core, because the actual November 2007 core configuration is not know at this time. This prevents the development of a final neutronic, thermal, and mechanical analysis. The complexity of the cross-sectional geometry requires a translator to be developed to read the output of the neutronic analysis and provide a database for use by the finite element thermal analysis. The finite element model is expected to use over 150,000 elements. The database will alleviate the need to enter the gamma heat generation in every element by hand. As noted previously, ATR operations required the schedule for design and fabrication of the AGC-1 capsule and gas control system be realigned to fit ATR's available limited resources. A resource loaded schedule was prepared and accepted by ATR operations. This schedule split the AGC-1 gas capsule and gas control system into two separate activities. The final design for the AGC-1 capsule is expected by the end of September 2006. Fabrication of the capsule is slated to start the second quarter of FY-07. The AGC-1 gas control system design and fabrication will start in FY-06 and will finish at the end of FY-07. This extended period of time is necessary to allow ATR personnel to install tubing and equipment in restricted ATR areas during planned outages. The planned schedule for the AGC-1 Irradiation experiment is given in Table 27. A schedule for other AGC irradiations is not currently available.

Table 27. Planned Schedule for AGC-1

AGC-1 Activity	Milestone Date
Preliminary AGC-1 test train design review meeting	May 25, 2006
Preliminary AGC-1 gas control system design review meeting	July 17, 2006
Final AGC-1 test train design review	August 15, 2006
Complete AGC-1 mock-up assembly	September 21, 2006
Complete bench-top testing	September 21, 2006
Final AGC-1 gas control system design review	December 12, 2006
AGC-1 ATRC run	February 6, 2007
Complete AGC-1 test train assembly	March 5, 2007
Complete installation of AGC-1 gas control system in ATR	August 16, 2007
Insert AGC-1 test train in Cycle 141A and complete SO testing of gas control system	November 17, 2007
Remove AGC-1 test train in Cycle 142B and transfer to ATR Canal	April 19, 2008

A study in support of a preliminary design of the AGC-1 sizing and dry storage apparatus was performed in FY-06. This study was needed because following shutdown of the hot cells at the INL Reactor Test Complex (RTC), there is currently no capability to transfer the AGC-1 experiment following ATR irradiation to the INL Materials and Fuels Complex (MFC) for disassembly or to ORNL for post irradiation examination (PIE). This study reviewed options for appropriately sizing the AGC-1 experiment to allow loading into the GE-2000 cask. This experiment sizing operation needs to be performed without water from the ATR transfer pool entering the experiment and contacting the graphite specimens which could degrade the information obtained during PIE. It was determined that this type of operation could create ALARA concerns for the ATR operational staff. Actions on further developing this capability have been tabled pending the determination of heat loads and the actual source term for the AGC-1 experiment following irradiation.

At the preliminary AGC-1 design review meeting held May 31, 2006, this issue was reviewed and alternate procedures were proposed to extract the core section of the experiment. These alternate procedures will be pursued in the future with ATR operational staff.

5.1.1.2.2 HFIR Irradiations — Thirty-six NBG-10 graphite flexure bars were successfully irradiated in a series of 18 High Flux Isotope Reactor (HFIR) PTT capsules at ORNL. Nuclear Block Graphite-10 (NBG-10) is a medium-grain, near-isotropic graphite manufactured by SGL Carbon Company at their plant in Chedde, France. NBG-10 graphite was developed as a candidate core structural material for the Pebble Bed Modular Reactor (PBMR) currently being designed in South Africa, and for prismatic reactor concepts being developed in the USA and Europe. NBG-10 is one of several graphites included in the US-DOE Very High Temperature Reactor (VHTR) program.

Thirty-six NBG-10 graphite flexure bars have been successfully irradiated in a series of eighteen HFIR PTT capsules at ORNL. The capsule irradiation temperatures were 294±25, 360±25 and 691±25°C. The peak doses attained were 4.93, 6.67, and 6.69 x 10^{25} n/m^2 [E>0.1 MeV] at ~294, ~360, and ~691°C, respectively. The high temperature irradiation volume and dimensional change behavior, and flexure strength and elastic modulus changes of NBG-10 were similar to other extruded, near-isotropic grades, such as H-451, which has been irradiated previously at ORNL. The low temperature (~294°C) irradiation volume and dimensional change behavior was also as expected for extruded graphites, i.e., exhibiting low dose swelling prior to shrinkage. This behavior was attributed to the relaxation of internal stress arising from the graphite manufacturing process and specimen machining.

The room temperature thermal conductivity of NBG-10 appears to be saturated within the fluence range of this study. The rate of reduction appears to correlate with the limited data available in the literature. In order to better correlate this data with H-451 a program to measure the thermal conductivity at the irradiation temperature, along with an appropriate isochronal annealing study, is recommended.

A cursory SEM microstructural examination was conducted and although the observations are only qualitative, they illustrate the mechanism of irradiation damage and its effect on the graphite microstructure at low neutron dose. A more detailed study involving quantitative image analysis combined with SEM is indicated.

This information is detailed in *Initial Post Irradiation Examination Data Report for SGL NGB-10 Nuclear Grade Graphite*, ORNL/TM-2005/518[79]. While the data reported in this report does not represent a complete database for NBG-10 graphite, the data obtained gives a measure of confidence that the current generation of nuclear graphites will behave in a familiar and well understood manner. A final report[80] was issued in FY-06 that provides all results of the PIE performed. While the data reported here do not represent a complete database for NBG-10 graphite, they give a measure of confidence that the current generation of nuclear graphites will behave in a familiar and well understood manner.

A report documenting the experimental plan and preliminary design of the HTV-1 and 2 was written in FY-05[81]. An updated report giving the objectives, technical plan, schedule, and final design details of the High Flux Isotope Reactor (HFIR) target irradiation experiments HTV-1 and 2 was written in FY-06[82]. Experiments HTV-1 and 2 will be irradiated in the HFIR target region to a peak irradiation dose of ~2.2 x 10^{21} n/cm^2 [E > 0.1 MeV] or ~1.6 displacements per atom (dpa) and ~6.5 x 10^{21} n/cm^2 [E > 0.1 MeV] or ~4.8 dpa, respectively. Each capsule has multiple temperature zones allowing irradiation temperatures of 900, 1200, and 1500°C.

The key data to be obtained from these capsules include:

1. Elastic constants (Young’s modulus, E; Shear modulus, G; Poisson’s ratio, ?)
2. Dimensional Change
3. Thermal conductivity, room temperature, and elevated temperature
4. Compressive strength
5. Microstructural characterization

These data are critical to the design of the NGNP and the high-temperature graphite irradiation creep capsule (AGC) planned for irradiation in the Advanced Test Reactor (ATR) at Idaho National Laboratory (INL). Moreover, the data supports ongoing work in the area of model development; e.g., irradiation effect models such as dimensional change, structural modeling, and fracture modeling. Moreover, the data will be used to underpin the American Society of Mechanical Engineers (ASME) design code currently being prepared for graphite core components.

The report reviews the background and theory of irradiation-induced dimensional change in graphites, and details the graphite grades to be irradiated in the experiment along with the rationale for their inclusion. Details of the irradiation test conditions are reported. Detailed capsule layout plans are reported for HTV-1 and -2, and the specimens are tabulated by grade, location, and anticipated fluence. Details of the capsule design, including the capsule thermal analysis are discussed. The process of pre- and post-irradiation examination is reported and details of the tests to be performed and the data to be acquired are given. Finally, the schedule for HTV-1 and -2 is reported.

The primary objective of irradiation capsules HTV-1 and -2 is to provide high-temperature irradiation dimensional change data to support the design of the high-temperature (1200°C) AGC creep experiment being planned for the ATR at INL, and design data to support the development of the next generation nuclear plant (NGNP). Specifically, data for the effects of neutron irradiation on the dimensional stability, strength, elastic modulus, and thermal conductivity of candidate NGNP graphites will be obtained in the temperature range 900-1500°C.

The planned schedule for Capsules HTV-1 and -2 is shown in Table 28.

Table 28. Planned schedule for Capsules HTV-1 and -2

TASK	Due Date
Issue draft experimental plan and preliminary design report	9-30-05
Issue draft final capsule design report	8-31-06
Initiate capsule construction	8-31-06
Complete preirradiation examination of HTV-1 and -2 graphite specimens	9-30-06
Complete assembly of HTV-1	3-31-07
Complete assembly of HTV-2	3-31-07
Insert HTV-1 and -2 in HFIR	6-30-07
Complete irradiation of HTV-1	8-31-07
Complete hot cell disassembly of HTV-1	10-31-07
Complete irradiation of HTV-2	12-31-07
Complete hot cell disassembly of HTV-2	2-28-08
Complete graphite PIE	6-30-08
Issue draft HTV-1 and -2 PIE report	9-30-08

The PIE of initial high temperature scoping graphite irradiations for mapping elevated temperature swelling (METS) was performed in FY-06. A report has not been issued on this work to date.

5.1.1.3 Standards and ASME Code Activities. Staff participated on the ASME Graphite Project Team on Core Supports. A summary of committee activities were detailed in the report *Status of ASME Section III Task Group on Graphite Core Support Structures, INL/EXT-05-00552*[83]. This report was updated and reissued in FY-06[84].

A major activity of D02.F over the several years has been drafting a materials specification for nuclear graphite. This activity was initiated in June 2002 when a discussion paper proposing a nuclear graphite specification was accepted by D02.F. The first draft (rev.1) of the Standard Specification was presented to the committee in Jan 2003 and in February 2005 version 11 of the standard specification was sent to sub-committee ballot for approval. The scope of the standard specification is: This standard specification covers the classification, processing, and properties of nuclear grade graphite billets with dimensions sufficient to meet the designer's requirements for fuel elements, moderator or reflector blocks, in a high temperature gas cooled reactor. The graphite classes specified here would be suitable for reactor core applications where neutron irradiation induced dimensional changes are a significant design consideration.

The subcommittee ballot results were discussed at the June 2005 meeting. A revised standard specification (revision 13) was approved for main committee ballot in June 2005 and final balloting and release of the specification was completed.

A second nuclear graphite materials specification "Standard Specification for Anisotropic Nuclear Graphite" is also being prepared and it is anticipated that this specification will be ready for subcommittee balloting in 2007. The scope of the second materials specification is: This standard specification covers the classification, processing, and properties of nuclear grade graphite billets with dimensions sufficient to meet the designer's requirements for fuel elements, moderator or reflector blocks, in a high temperature gas cooled reactor.

Another activity of D02.F that has progressed in FY-05 is the development of a graphite air oxidation test method. A graphite oxidation test stand was assembled at ORNL to allow participation in the planned oxidation round-robin. Similar systems exist at the Korean Atomic Energy Research Institute (KAERI), Carbone USA, and GrafTech (two locations). A round robin trial of the standard test method commenced in August 2005 and has not been completed to date.

A report written in FY-06[85] that provides the status of the development of ASTM DO2.F standard test method for the air oxidation of graphite. The draft of a "Standard test method for oxidation rate and threshold oxidation temperature of manufactured carbon and graphite in air" was approved at the December 6-8, 2005 meeting in Norfolk, VA of the ASTM D02-F0 subcommittee on Manufactured Carbons and Graphites. The draft was based on a quality control method in use at one major industrial graphite producer; the method was further improved and developed by ORNL. The ASTM subcommittee decided to proceed with the organization of the inter-laboratory test.

The equipment for the inter-laboratory test has been installed and tested at ORNL, and became operational in August 2005. Similar equipment is in advanced stage of setting up at INL.

Four different graphite materials were selected for the round-robin test; identical specimens, machined at ORNL, were distributed to participating organizations. Currently five different organizations are participating in the inter-laboratory test. The tests have been completed (or are about to be completed) at several organizations; other organizations are in different stages of progress with the tests. Additional organizations expressed interest and may join the test at a later time.

Preliminary results obtained at ORNL show that (1) the repeatability of the test, in similar conditions, on random samples from a uniform batch of materials is good; (2) the test shows differences between the properties of materials of different origin tested so far. More work will be needed in order to statistically process the results returned by all participating organizations (when available) in order to determine how reproducible the results between laboratories are. It is expected that the significance of the test results and will become more evident at the completion of the inter-laboratory test. The examination of all data collected may possibly suggest that minor modifications in the test procedure might be necessary in order to improve the consistency of results.

Additional data will become available from the participating organizations before the next meeting of the ASTM D02-F0 subcommittee in December 2006.

Oxidation of graphite by air is a heterogeneous reaction involving a porous solid and a gas. Depending on conditions, graphite oxidation can be controlled by one or more of the three idealized steps:

1. Mass transport of oxygen or reaction products (CO and CO_2) across the relatively stagnant gas film at the gas – solid interface;

2. Mass transport of oxygen from the exterior surface to an active site in the bulk of the solid and of the reaction products in opposite direction;

3. Chemical reaction (carbon oxidation) at the active sites.

Temperature is of prime importance in determining which of these steps controls the graphite oxidation in air. Other factors, such as sample geometry, air permeability, and intrinsic surface reactivity, oxygen partial pressure, and flow conditions have secondary importance.

Experience has shown that the variation of the reaction rate with the temperature can be divided in the three main regimes shown schematically in Figure 43. In the low temperature range (regime 1) graphite oxidation is controlled by the basic laws of chemical kinetics and the reaction proceeds uniformly throughout the volume of material (if porosity is uniformly distributed in the volume). The reaction rates have a marked temperature dependence (given by the Arrhenius equation), with a significant increase in the rate as the temperature increases. In this range, the measured (apparent) activation energy is by definition equal to the true activation energy of the chemical reaction. In the highest temperature range (regime 3) oxidation is restricted to a narrow layer at the external surface of the material, and is limited by the boundary layer developed at the solid-gas interface. Because gas transfer processes have very low activation energies, the reaction rate has very little dependence on temperature. A transition domain (regime 2) exists at intermediate temperatures, where oxidation is controlled by the pore-diffusion rate. The activation energy observed in this regime is approximately half of the true activation energy.

The simplest method of determining the reaction regime is to vary the surface-to-volume ratio of the graphite. If the rate is dependent only on the total volume (or mass) of graphite, then reaction is in regime 1. If the rate is proportional to the external surface area, the reaction is in regime 3. If the oxidation rate is sensitive to the surface-to-volume ratio, the reaction is in regime 2 or an intermediate regime[86].

The temperature boundaries between these regimes depend much on sample properties (high content of impurities with catalytic effect accelerates the rate of chemical reaction) and test conditions (specimen size and geometry, flow conditions). In regime 1, the rates normalized to sample volume (or mass) can be used because of lack of oxidation profiles. In regimes 2 and 3 it is reasonable to use rates normalized per geometrical surface. Considering erosive mass loss, only regimes 1 and 2 are the most important, because oxidative strength loss of carbons occurs by an increase in porosity. In regime 3 there is practically no strength loss because oxidation does not penetrate into the pores of material and remains restricted to the exposed surface. Consequently, in regime 3 the oxidation rate is not material dependent and kinetic measurements are not necessary[87].

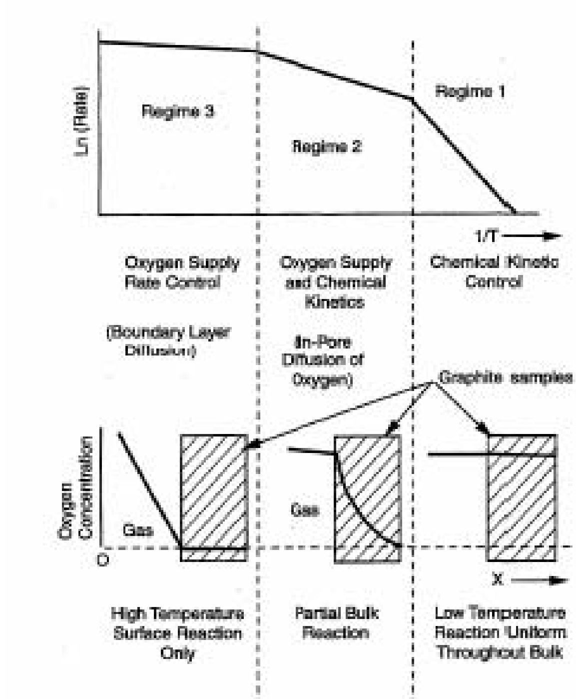


Figure 43. Schematic showing the effects of temperature on oxidation rates of graphite. The semi-logarithmic plot of oxidation rates versus the reciprocal of absolute temperature is known as the Arrhenius plot. The slope of the linear segments is correlated with the activation energy. However, only in the low temperature range (regime 1) the linear dependence shown in this scheme is truly determined by the basic laws chemical reaction rates[88].

Committee D02.F, which includes ORNL and INL staff, also worked on developing a new test method for determining the Critical Stress Intensity Factor (KIc) of graphite. The proposed method, a single edge notched beam tested in three point flexure, was adopted from the ASTM standard method C 1421 “Standard Test method for Determination of Fracture Toughness of Advanced Ceramics at Ambient Temperatures”. An initial ruggedness test of this method was performed at ORNL in 2004 and the results discussed at the December ‘04 meeting. The committee recommended some changes in the proposed method be made and a second ruggedness study was conducted in early 2005. Details of this work are found in the report *Development of a Fracture Toughness Testing Standard for Nuclear-Grade Graphite Materials Status Report*, INL/EXT-05-00487[89]. The results of the second ruggedness test were reviewed by the committee at the June 2005 meeting and approval was given for the round robin of the draft test method to proceed. Specimens were machined from two grades of graphite: Carbone 2020 (fine grain graphite) and GrafTech PGX (medium grained graphite). A third graphite SGL R4650 (ultra-fine grained graphite) was shipped from SGL to ORNL and specimens were subsequently machined. The draft test method and round robin specimens were distributed to participants in September 2005. The draft standard test method developed is documented in ORNL report *Standard Test Method for Determination of Fracture Toughness of Graphite at Ambient Temperature*[90]. The round robin testing of specimens associated with this test method is under way.

ASTM C781, *Standard Practice for Testing graphite and Boronated Graphite Components for High-Temperature Gas-Cooled Nuclear Reactors*, was comprehensively reviewed by committee D02.F in FY-05.

The June 2005 meeting the D02.F committee also conducted a thorough review of C 709, *Standard Terminology Relating to Manufactured Carbon and Graphite*. Activities of the ASTM Subcommittee DO2.F are documented in *Status of ASTM Subcommittee DO2.F Graphite Activities*, ORNL-GEN4/LTR-05-003 [91].

5.1.1.4 Modeling. Isotropic and near-isotropic nuclear grade graphites are proposed for use in the NGNP. During reactor operation graphite core components and core support structures are subjected to complex stresses such as combined loading from neutron irradiation induced dimensional change and thermal gradients. Moreover, static and seismic stresses act on the core components. These stresses, acting singularly or in combination, should not be sufficient to cause failure of the graphite core components. Thus it is desirable to be able to predict the probability of failure for a given stress state. Various theories of failure have been suggested for graphite, such as the maximum principal stress criterion, or the Weibull theory of fracture. One promising approach to the prediction of failure probability is a fracture model previously described by Burchell[92] that uses a fracture mechanics based failure criterion and that captures the essence of the microstructural processes that control fracture in graphite. A report was written in FY-06 that documents the progress made to date to utilize this model for graphite failure prediction for a nuclear environment[93].

The Burchell model requires several input parameters to model a particular graphite grade and/or specimen geometry.

A graphite specimen or component is modeled as a rectangular monolith with a volume equal to the stressed volume of the component or specimen, and a breadth such that the stressed cross-sectional area is identical to that of the component or specimen. The model accounts for the orientation of a specimen or component with respect to the materials forming direction (i.e., texture effects), by applying appropriate values of the mean particle size, mean pore size, and standard deviation of the pore size distribution.

The applicability of the Burchell model was first demonstrated by correctly predicting the uniaxial tensile strength distribution of H-451 graphite. The model's ability to predict the volume dependency of the strength of graphite, in the manner of the Weibull theory, is demonstrated. An extensive data set for the multiaxial strength of H-451 graphite (obtained from large specimens) in the first (tension plus circumferential) stress quadrant, and fourth (compression plus circumferential) stress quadrant was obtained as part of the nuclear graphite characterization program conducted at ORNL in the early 1990's.

The Burchell model was successfully applied to predict the H-451 graphite failure surface in the first and fourth stress quadrants. In the first stress quadrant the combination of the Burchell model and the principal of independent action allowed the fracture surface to be accurately predicted. In the fourth stress quadrant failure in compression is assumed to occur because of the circumferential stress induced by the Poisson strain. It is concluded that the principal of independent action is not applicable in the fourth stress quadrant where the stresses causing failure act in the same sense. However, the failure surface was accurately predicted by modeling the multiaxial specimen under an "effective" uniaxial (circumferential) stress. Under these circumstances the Burchell model conservatively predicts the failure surface in the fourth stress quadrant.

Finally, the extension of the Burchell fracture model to predict the effect of neutron irradiation on the strength of nuclear graphite is reported. Evidence is presented from recent irradiation experiments confirming the effects of irradiation on the microstructure of graphite. By varying the Burchell model's inputs in the manner expected on irradiation the neutron dose dependency of graphite strength is qualitatively predicted.

The robust performance of the Burchell fracture model in predicting the strength/probability of failure of graphite, under both uniaxial and multiaxial stress conditions, indicates that the model should be further developed within the framework of the ongoing failure theory evaluation exercise being conducted by the American Society of Mechanical Engineers (ASME), Project Team on graphite core components.

5.1.2 High Temperature Design Methodology

The area of high temperature design methodology (HTDM) is directed at the requirements and methodology for the successful ASME codification of high temperature metallic alloys required for the NGNP. The alloys which are currently being qualified by the NGNP Materials Program are Alloys 617 and 230. Other alloys may be potentially useful; however, current funding limitations will not allow the addition of other alloy types. A discussion of other potential alloys that could be used is given in Survey of Metallic Materials for Irradiated Service in Generation IV Reactor Internals and Pressure Vessels, ORNL/TM-2005/519[94]. Neither of these alloys is currently ASME code approved for Section III (nuclear) applications and neither is currently listed in ASME Section III, Subsection NH for elevated temperature applications. A prior Section III code case for Alloy 617 has expired and requires a complete rewrite. Both of these alloys are allowed for Section VIII (non-nuclear) applications. These alloys were selected for qualification and codification for the NGNP because they are very mature alloys and considerable experience has been gained over many years in the use of these alloys for construction of critical components for the fossil and petrochemical industries. These alloys are also quite stable and developed for very high temperature applications; however, neither has been extensively qualified for nuclear applications in the past. The use of these alloys is primarily directed toward the potential use of an intermediate heat exchanger (IHX) system in the design of the NGNP. The IHX could either be used for complete primary to secondary isolation (indirect cycle) of the helium coolant or partial isolation between the primary and a high temperature heat source application such as the production of hydrogen gas. The alternative to an indirect cycle is called a direct cycle which allows primary helium to directly contract the balance of plant. The AREVA and the PBMR designs described previously use an indirect and direct cycle, respectively. If the PBMR design was selected, it is probable that much of the effort associated with HTDM could be downgraded to a lower priority or removed from the program.

5.1.2.1 Controlled Atmosphere Test System. A new system for performing high-temperature creep-fatigue tests in a controlled environment was procured, installed, and successfully checked out. The system performs creep-fatigue tests with extended hold times on metallic specimens in strain control at temperatures up to 1100°C. The specimen, grips, and extensometer are enclosed in a stainless steel environmental chamber with associated vacuum and gas control accessories. The specimen is induction heated. A purchase order for the system was placed with MTS Systems Corporation (Eden Prairie, MN) in January 2005. The system was delivered in August 2005; installation and initial checkout was performed by MTS personnel from September 6-16. A final creep-fatigue checkout test at 1000°C under vacuum conditions (1x10-4 Torr) was successfully performed on September 21. A report was written that documented this effort[95]. A photograph of the system is shown in Figure 44.

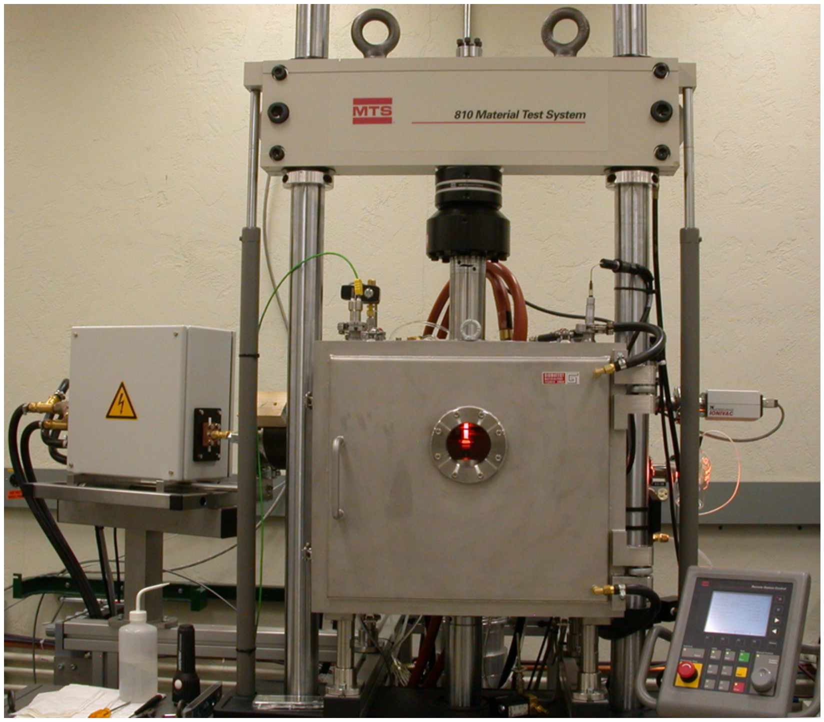


Figure 44. Environmental chamber installed on creep-fatigue load frame.

5.1.2.2 Materials Handbook. The current status of the materials handbook is documented in Initial development of the GEN IV Materials Handbook[96]. Activities in preparing existing data on Alloy 617 for the Gen IV Materials Handbook through data mining and assessment were summarized in *Assessment of Existing Alloy 617 Data for Gen IV Materials Handbook, ORNL/TM-2005/510*[97]. Status of existing data was reviewed and assessment approaches were discussed. Data classification was used to provide a reference for quality and reliability evaluation. A tracking system was developed so that all data elements can be traced back to their original source for background review whenever needed to facilitate convenient data processing and the future input into the Gen IV.

Materials Handbook, formats for data editing and compilation were established. Existing data that was the most germane to Gen IV nuclear reactor applications were evaluated for their data types, material status, testing conditions and other background information. Acquisition of European data on the alloy for nuclear applications was also reported.

5.1.2.3 Investigation into Alloy 617 Specification. An investigation was conducted in an effort to refine the standard specifications of Alloy 617 for VHTR applications and is documented in ORNL/TM-2005/504, *Development of a Controlled Material Specification for Alloy 617 for Nuclear Applications,* by Ren and Swinderman[98]. Historical data generated from various heats of the alloy were collected, sorted, and analyzed. The analyses included examination of mechanical property data and corresponding heat chemical composition, discussion of a previous Alloy 617 specification development effort at the ORNL, and assessment of the strengthening elements and mechanisms of the alloy.

Based on the analyses, literature review, and knowledge of Ni base alloys, a tentative refined specification was recommended and given in Table 29. The CCA designations noted in this table were developed previously at ORNL for work that was performed in support of the Fossil Energy Advanced Research Materials Program. The Gen IV 617 specifications are the minimum and maximum specifications recommended.

Table 29. Recommended Tentative Chemical Composition of Alloy 617 for VHTR Materials Testing

Heat	Ni	Cr	Co	Mo	Nb	Fe	Mn	Al	C	Cu	Si	S	Ti	P	B	N
ASTM Min	44.5	20.0	10.0	8.0	-	-	-	0.8	0.05	-	-	-	-	-	-	-
ASTM Max	-	24.0	15.0	10.0	-	3.0	1.0	1.5	0.15	0.5	1.0	0.015	0.6	-	0.006	-
CCA 617 Min		21.0	11.0	8.0				0.80	0.05				0.30		0.002	
CCA 617 Max		23.0	13.0	10.0		1.5	0.30	1.30	0.08	0.05	0.3	0.008	0.50	0.012	0.005	0.050
GenIV617 Min	44.5	22.0	13.0	9.0				1.20	0.07				0.40		0.002	
GenIV617 Max		24.0	15.0	10.0		1.0	1.0	1.40	0.10	0.2	0.3	0.008	0.60	0.010	0.005	0.040

Based on the analyses of mechanical properties and chemical compositions of historical data from various heats and the strengthening elements and mechanisms, a tentative refined specification of Alloy 617 was recommended for VHTR materials testing in the future. For the reasons discussed in the report, the specifications for Co. Mo, Fe, Al, C, Cu, Si, S, Ti, P, B and N were restricted based on the standard ASTM chemistry specifications. The other elements remain unchanged for the time being as listed in the ASTM standard specifications

The GEN IV 617 specification noted in the table was subsequently referred to as a controlled material specification (CMS) for Alloy 617 and the procurement of CMS Alloy 617 and Alloy 230 is described in an FY-06 report[99]. The results of the development of the CMS Alloy 617 have been inconclusive to date.

5.1.2.4 Testing Performed on Alloy 617 Joints and Base Alloy Samples. Activities at the INL were focused on improved understanding of creep-fatigue-environment interactions in candidate high-temperature alloys and joints made from them. Initial work was performed on Alloy 617 due to the fact that it appears to be closest to gaining code approval in ASME Code Section III.

The first task in the study of creep-fatigue-environment interactions in Alloy 617 base metal and joints was the production and characterization of joints prior to creep-fatigue testing. The results of initial microstructural and mechanical characterization of three types of high temperature Alloy 617 joints: fusion welds, high-temperature braze joints, and diffusion bonds were documented in *Microstructure and Strength Characteristics of Alloy 617 Welds,* INL/EXT-05-00488[100]. In the absence of an IHX design, the joints produced and studied in this program were "typical" for the alloy, i.e., joining parameters that are those commonly used in industry and dimensions are those convenient for production of test specimens. It is recognized that overall joint behavior is system and geometry dependent; the joints tested were designed for ease of interpretation of fundamental mechanical and environmental degradation mechanisms.

With the exception of a problem of braze joint wetting, the characteristics of the high-temperature Alloy 617 joints have been as expected.

Creep-fatigue tests have been performed on Alloy 617 samples with and without fusion welds at 800^0C and 1000^0C in air and vacuum. A report was written to document the status of this effort in FY-05[101]. These results will be updated for FY-06 in a report to be issued in August, 2006. No significant mechanical properties testing has been performed to date with specimens exposed to the impure Helium environment expected to be used in the NGNP.

5.1.2.5 HTDM in Support of ASME Code Development. High temperature design methodology (HTDM) includes the integration of simplified design methods and material data generation towards development of ASME B&PV Code for elevated temperature design procedures that address time-dependent failure criteria and assure adequate life. Data generation includes design data needed to quantify criteria such as uniaxial creep-rupture data, as well as specific data used in the development of criteria such as mutliaxial strength criteria and creep-fatigue interaction criteria. The work performed for simplified methods development in FY05 is documented in *Simplified Design Criteria for Very High Temperature Applications in Generation IV Reactors*, ORNL/TM-2004/308, Revision 1[102] and in *High Temperature Design Methods Development Advances for 617: Status and Plans*, ORNL/TM-2005/515[103].

HTDM also will provide experimentally based constitutive models – the foundation of inelastic design analysis required by ASME B&PV Section III Division I Subsection NH (NH). These equations are required to characterize the time-varying thermal and mechanical loading of structures. The equations require full time-histories of tensile, fatigue, and creep test data at many test conditions. The ability to predict the time history of stress and strain of reactor components is critical in integrating with damage and lifting models in predicting time-dependent failure modes A report that summarizes the initiation of scoping tests to provide time dependent input for constitutive equation development was written in FY-05[104]. Specimens were made of Alloy 617 produced by Special Metals. Scoping tests were conducted at 800 and 850°C (1472 and 1562°F) at different stress levels for creep rupture times of 1,000 hours or more.

A report written in FY-06[105] that provides a comparison of Ashby deformation maps for several nickel-based alloys. The study revealed that for several nickel-based super alloys power law creep does not dominate at stresses below ~10 MPa. Instead, the diffusion of atoms through the matrix of the grain (Nabarro Herring creep) accounts for the majority of strain.

No test data are available for Alloy 617 or 230 that approach the extremely low stresses and long durations required for the NGNP. The current database includes test results for higher stresses, requiring extrapolation of the database for times relevant to the 60 year plant life of the NGNP, or even fractions of the plant life for intermediate heat exchangers that are expected to require replacement at least two to three times over the plant life. The deformation mechanism identified in the existing database for Alloy 617 is power law creep. Uncertainty exists whether or not this mechanism remains dominant at significantly low stresses and very long times envisioned for several IHX design concepts. If a different deformation mechanism dominates at lower stresses, extrapolation of power law creep data to lower stresses could predict non-conservative (lower) creep rates by several orders of magnitude.

Alloy 617 or 230 are also expected to experience diffusion-controlled creep when subjected to the low stress levels of interest to the IHX for the NGNP for 100,000 to 200,000 hours; although there are no test data on these specific alloys to verify the prediction. Consequently, the creep rates expected in an IHX material may be several orders of magnitude higher than those predicted by the power law creep model in the current database. This will result in shorter expected service lives for IHX designs, as the 1% strain limit currently in ASME Section III Subsection NH (elevated temperature design nuclear class 1 component design code) will be exceeded more quickly for a given design stress. The service life could also be limited due to reduced creep rupture lives, although creep rupture data are not available or do not exist to ascertain any differences. Hence, extrapolations of current database results based on a power law creep model may be highly non-conservative.

Since a deformation mechanism map is not available for Alloy 617 or 230, another nickel alloy (Ni-13.5%Cr-1%ThO_2) was used as a 1st order estimate. Isochronous curves were generated with the model for comparison with German isochronous curves to determine the effects of diffusion-controlled creep on design stress limits for 100,000 hour service. A range of grain sizes were evaluated. The design stress (based upon a strain limit of 1% consistent with ASME code) was predicted to be reduced by at least 50%, and for fine grains was predicted to decrease by several orders of magnitude.

The design options to be considered for the IHX for the NGNP will require different grain sizes; certain compact designs will require thin sections that will require finer grains than those designs that include thicker sections. The predicted isochronous curves for these grain sizes will place varying restrictions on permissible stresses. The creep deformation mechanisms used to produce these isochronous curves must be verified experimentally for Alloy 617 and 230. A recommended test matrix was developed to address this need; testing has already commenced on large grain material

Confirmation or revision of the isochronous curves with experimental test results will be very valuable in providing material limitations for conceptual and preliminary design efforts, the selection of an IHX design concept, and the selection of a nickel-based super alloy (617 or 230). Efforts are planned in the next fiscal year to acquire fine grain 617 and/or 230 for testing and validation. Elastic and simplified inelastic methods will be used in conjunction with the isochronous curves in the early design stage process; R&D continues in this area. Comprehensive testing will be required in support of the development of unified constitutive models for use in the final design of the IHX. Depending upon the product form (grain size) and alloy selection (617 or 230), varying levels of extensive testing will be required to generate sufficient data for determining stress allowables for the IHX material for ASME codification. Testing will also need to address creep-fatigue testing of joints or weldments to assess loss of ductility from the fabrication process, possible grain growth, and environmental effects. Structural testing, both in representative coupon type samples as well as small scale IHX models will be required to validate an IHX design, including the to-be-developed ASME Section III stress allowables and design criteria for the IHX material.

The activities listed in the previous paragraph are consistent with the High Temperature Design Methodology R&D Program Plan and High Temperature Metallic Materials Test Plan developed several years ago for DOE by ORNL. Initiation of activity to support the codification of an IHX within ASME Section III is expected in 2006 through the existing DOE-ASME Cooperative Agreement. Realization of bi-lateral agreements with France (AREVA) would greatly facilitate the design process for an IHX, as AREVA appears to be the only stakeholder with significant effort, commitment, and progress on the development of an IHX – including testing to down-select a nickel-based super alloy for the final design. Formation of a working group within the ASME Section III Subgroup of Elevated Temperature Design is also anticipated (this activity would be in addition to the DOE-ASME task on IHX codification); the ability of the working group to freely share information on materials, design concepts, expected normal operating conditions as well as transient conditions, and other relevant information will likely be restricted until formal agreements have been reached between relevant stakeholders and DOE.

A report was written in FY-06[106] that summarizes the progress and status of efforts to evaluate and improve upon elevated temperature design (ETD) methodology in the ASME Section III relevant to the NGNP. Design criteria for ETD are categorized as Load Controlled or Deformation Controlled criteria. These criteria are briefly summarized below and discussed in the report; a summary of the developments in modified or additional criteria are discussed, and a brief summary of additional efforts that are required to resolve existing issues is given. The acceptance of existing or future design procedures for ETD by the NRC will play a significant role in the ability to obtain a construction and operating license for the NGNP.

Load controlled stress criteria for ETD of nuclear components exist to ensure against short term (single load applications) and long term (creep rupture) failure modes due to primary loads. Primary loads are viewed as loads that cause damage under monotonic loading, whereas both primary and secondary loads contribute to producing damage under cyclic loading. Currently, the methodology is limited to elastic analysis, where modifications of elastic analyses are incorporated to account for steady state creep stress distributions. This report summarizes current technology available for implementing inelastic design analysis for load controlled stress criterion as an alternative or additional design methodology, namely the use of limit analysis or reference stress analysis. Equivalent reference stress criteria are proposed for use in ETD methodology in ASME; the criteria are based greatly upon research conducted in Great Britain and utilized in the British Nuclear Code R5. These criteria are formulated to be consistent with and maintain the intention of current ETD criteria in ASME Section III (specifically Class 1 nuclear component design in ASME Section III Subsection NH (NH), although NH serves as the foundation for many other criteria in Section III). Limit analysis provides a consistent and structured methodology in the classification of stresses as primary or secondary – even for complex structures and loading. Proper classification of stresses are one of the most fundamental and critical aspects of the design. Application of current stress classification methods can be difficult, and often times requires judgment from engineers. Limit analysis provides a consistent methodology to classify stresses in these scenarios. Furthermore, limit analysis can be used to address discontinuity effects in load based design criteria; currently load based design criteria do not *directly* address discontinuities – they are actually addressed in deformation based design criteria. Finally, limit analysis is a common method utilized by engineers that is easily implemented within existing commercial software for structural analysis (e.g. finite element analysis). Typically, finite element codes provide engineers with excessive amounts of information that is difficult, if not impossible, to apply and interpret with ASME design criteria and intent. Limit analysis lends itself to easy and proper interpretation and integration of the proposed design criteria, even when conducted with finite element software. As such, the approach incorporates current technology for use in the design of Gen IV reactors along with ASME ETD design criteria without losing the fundamental bases for which the ASME ETD code was founded to ensure against recognized failure modes of materials and structures.

Deformation controlled design criteria for ETD address the effects of cyclic loading and secondary stresses on various failure modes such as excessive strain accumulation, fatigue, creep-fatigue, and buckling. Full inelastic analysis is very time consuming, expensive, and typically is difficult to interpret and apply the results to structural design within the ASME code. Such analyses are typically reserved for very complex and critical structures, usually in late stages of the design process. Early design stages typically utilize simplified methods, either elastic analyses or simplified inelastic analyses, to size structures, components, and down select one or more materials for conceptual and preliminary design stages. They can also be used for investigating sensitivity of material properties and load cycles. Current elastic analysis (A-Tests) and simplified inelastic analyses (B-Tests) in ASME-NH address the effects of secondary cyclic loading (e.g. cyclic thermal stresses) on accelerating damage or failure above and beyond the effects due to primary stresses alone. The application of these 'Tests' to materials and structures at *very high temperatures* is uncertain; in fact, the draft code case for Alloy 617 does not permit use of these simplified analyses for temperatures above 649^{o}C (1200F). The uncertainty lies partially in the inability to easily distinguish between plasticity and creep at temperatures above 649^{o}C. An analytical model that attempts to account for the decrease in yield strength with temperature was developed (the Alpha Model) and is discussed in the report. Finite element elastic-plastic simulations confirm the Alpha Model within the assumptions made to develop the model. The development of the Alpha Model reveals an assumption made in the current B-1 Test that *may* lead to non-conservative designs. Full inelastic finite element simulations that include elastic, plastic, and creep behavior were conducted as a means of predicting the 'true' behavior of the material and structure under realistic loading conditions. While results are preliminary, simulations of a thin tube under constant primary load with cyclic secondary stresses (cyclic linear thermal gradient through a thin tube wall) of Alloy 617 reveal significantly different structural behavior than predicted by both the B-1 Test and the Alpha Model. The results also indicate that the B-1 Test is not conservative for Alloy 617 under the loading conditions investigated.

The acceptance of limit analysis for use in Stress Controlled Design Criteria for ETD will require approval by members of the ASME Subgroup of Elevated Temperature Design (SG-ETD). Issues that may need further clarification are how creep damage evolves and is predicted in structures with gross structural discontinuities (e.g. intersections of nozzles and cylinders, or stepped cylinders) and redundant structures that can result in large amounts of accumulated strain. A critical review of available testing results from research programs in Europe that were used to validate and adopt limit analysis (reference stress analysis) for prediction of creep rupture and deformation needs to be conducted; additional structural validation testing will likely be required. Similarly, the assessment that the B-1 Test in NH is not conservative at very high temperatures for Alloy 617 is a significant issue that the SG-ETD will need to consider, review, verify, and resolve. This will include reconciliation of previous analytical and experimental results that were reported to demonstrate the conservatism of the B-Tests for stainless steel structures during the Liquid Metal Fast Breeder Reactor (LMFBR) program. The Gen IV High Temperature Design Methodology R&D plan developed by ORNL several years ago anticipated the need for additional analyses development, materials testing, and structural testing to validate structural design methodology in support of Gen IV reactors, including the NGNP. Efforts continue within the NGNP program to address these findings.

A report that documents the status of the development simplified methods and constitutive equations was written in FY-06[107]. This report is a companion document to the report noted above in July 2006. This report summarizes additional progress and the current status of efforts in 2006 to evaluate and improve upon elevated temperature design (ETD) methodology for proposed acceptance and use in the American Society of Mechanical Engineers (ASME) Section III relevant to Generation IV (Gen IV) reactors, specifically the NGNP. Design criteria for ETD are categorized as Load Controlled or Deformation Controlled criteria. This report does not address Load Controlled design methods and criteria; a summary of the status and proposed methods and criteria was covered in the report noted above. A summary of the developments in simplified design methods for cyclic loading (deformation controlled criteria) is provided in the August report referenced. The objective of this portion of the project is to gain acceptance of modern analysis techniques and tools in support of simplified methods without loss of the supporting fundamental theories that bound structural behavior and serve as the foundation for design and construction codes for pressure vessels and nuclear reactor components. Furthermore, a summary of the status of refurbishment of creep machines for testing of Ni-base superalloys for materials research relevant to the design and selection of an internal heat exchanger for the NGNP is provided. The acceptance of existing or future design procedures as well as acceptable design curves for heat exchanger materials for ETD by the Nuclear Regulator Commission (NRC) will play a significant role in the ability to obtain a construction and operating license for the NGNP.

Deformation controlled design criteria for ETD address the effects of cyclic loading and secondary stresses on various failure modes such as excessive strain accumulation, fatigue, creep-fatigue, and buckling. Early design stages typically utilize simplified methods, either elastic analyses or simplified inelastic analyses, to size structures, components, and down select one or more materials for conceptual and preliminary design stages. They can also be used for investigating sensitivity of material properties and load cycles. However, the application of current simplified methods to materials and structures at *very high temperatures* is uncertain; in fact, the draft code case for Alloy 617 does not permit use of these simplified analyses for temperatures above 649°C (1200F). The uncertainty lies partially in the inability to easily distinguish between plasticity and creep at temperatures above 649°C.

A review of shakedown and ratcheting theory for bounding solutions provided by simplified methods was conducted. The review supports the case that modern numerical tools, such as finite element analyses, can be utilized by designers to conduct calculations that were otherwise impossible prior to the existence of the computer. These design methods indeed can be utilized without loss of understanding of the fundamentals of the problem and the design process; the methodology also provides solutions that are bounded, meaning that the actual structure behavior will be less severe than predicted by the simplified methods. Modern simplified methods are available for designers' use at much lower costs than ever before. Another positive aspect of modern simplified design methods is the ability to address specific geometries and loading conditions, as opposed to the extrapolation of solutions for simplified one-dimensional problems with restricted load histories such as the classical Bree problem. The use of simplified methods is of critical importance in pre-conceptual, conceptual, and preliminary design stages. In fact, they may also be very useful in final design stages as well. The realization that numerous modern computational and analytical tools can be utilized in the design process without loss of the understanding of the fundamental issues related to the design and safe operation of nuclear reactor components at elevated temperatures will be of great value in satisfying historical concerns raised by the Nuclear Regulator Commission and the Atomic Committee on Reactor Safeguards during review of the Clinch River Breeder Reactor and the PRISM reactor in the 1980's and 1990's. Furthermore, the future approval of alternative simplified design methods, rather than currently approved simplified methods or full inelastic analysis, by members of the ASME Subgroup of Elevated Temperature Design (SG-ETD) will also provide confidence and justification for various other design methods utilized and developed by reactor vendors.

Design methods must incorporate design curves for various material properties in order to avoid various failure modes, e.g. creep rupture or excessive strain accumulation. Creep rupture curves are utilized to protect against sustained primary loads. Isochronous curves are utilized in the prediction of strain accumulation and the enhanced accumulation of creep strain due to cyclic loading. The draft code case for Alloy 617 lacks very long term creep test data at elevated temperatures. The intermediate heat exchanger (IHX) operation conditions and intended service life will be extremely challenging in terms of a successful design and selection of an adequate and affordable material.

Refurbishment of creep machines at ORNL and testing initiated on Alloy 617 at 950°C at very low stresses are partially complete. Test plans were developed and reported in July 2006 to target anticipated changes in active deformation mechanisms for very long term service at elevated temperature that are not captured in the draft code case for Alloy 617. Several minor repairs and replacements were required to update the creep machines and specifically target the low stress level testing. A hydraulic loading system was developed to gradually apply the test force such that pertinent data could be obtained during the loading phase of the creep test. The bending strain specification has been redefined for low load applications, mandating that the absolute bending strain remain below a limit of 100 $\mu\varepsilon$. This translates to less than 1% error in a creep test to 1% strain. Also, grip heaters have been installed to limit conduction losses from the sample and maintain a temperature gradient of less than ±2°C across the gauge length. Since the majority of technical issues have been resolved, the remaining tasks involve applying these modifications to additional creep machines. Results of the creep tests will provide experimental data in support of extrapolation of temperature and times relevant for IHX designs. Testing is currently limited to large grain Alloy 617. IHX designs include options that utilize much thinner product forms that require fine grained material. An ORNL report in July 2006 (discussed above) provided first order estimates of expected creep behavior in terms of isochronous curves for fine grained Alloy 617; these predictions were based upon mechanistic material models. The predictions are also applicable to Alloy 230. A lack of data for fine grained material requires additional testing to verify the models for use in the design and material selection process. Efforts in this direction are still required and will commence upon approval and funding.

A report that documents the status of negligible creep rules applicable to the NGNP reactor pressure vessel and applicable steels to be used for construction of the vessel will be written in FY-06 but is not available for inclusion in this report.

5.1.2.6 ASME Section III, Subsection NH Code Support. ORNL and INL staff attended quarterly ASME B&PV Code meetings in support of VHTR and Gen IV reactor needs, specifically ASME Section III Division I Subsection NH – the Subgroup on Elevated Temperature Design. Broad plans for R&D activities to support ASME Codification, primarily NH, for HTDM have been laid out and reported in *R&D Plan for Development of High-Temperature Structural Design Technology for Generation IV Reactor Systems*, ORNL/TM-2004/309[108]. These plans have not changed except for the overall timeline in accordance with allocation of funding by the government. However, increased interaction with stakeholders and NH members at ASME B&PV Code meetings and GIF meetings, along with currently funded VHTR & Gen IV activities have resulted in development of more detailed plans. These plans address the need to update and expand appropriate materials, construction and design codes within ASME B&PVC for application in future Generation IV nuclear reactor systems that operate at elevated temperatures. Implementing new materials and design methodologies, or simply updating current ASME B&PVC, requires a tremendous amount of review, discussion, and validation of all proposed codes and standards, with an agreed upon consensus from experts in appropriate areas. To this extent, numerous tasks and activities required for code evaluation and development for Gen IV reactors were identified as a result of the meetings in FY-05. These are supplementary tasks that are closely tied to the related activities and documented in the *Updated Generation IV Reactors Integrated Materials Technology Program Plan*, Revision 1[109]. An updated version of this plan was issued in December, 2005[110].

A summary of the status and plans associated with HTDM testing to be performed on Alloy 617 in support of an updated Code Case has been documented[111] A discussion of tasks currently funded directly by DOE and being performed by the ASME in support of the NGNP is given in Section 1.7. An updated status of the activities of Subsection NH and the DOE supported ASME activities noted above was written in FY-06[112].

5.1.3 Environmental Testing and Thermal Aging

Prior experience with high temperature gas cooled reactors has shown that helium on the primary side of the reactor will have significant levels of impurities during reactor operation. The expected composition range for impurities in the NGNP has been examined and is shown in Table 30, along with values used previously for a number of reactor programs[113]. Impurities in the helium arise from a number of sources including impurities in the graphite, lubricants in pumps and valves, and leakage into the system. Note that results of measurements from the gas cooled Fort St. Vrain plant in the United States are not included in Table 5 since there were substantial leaks of steam into primary circuit from the power generation circuit in this plant.

Table 30. Composition helium environments (advanced HTGR) used in past tests[114]

Program	H_2 (µatm)	H_2O (µatm)	CO (µatm)	CO_2 (µatm)	CH_4 (µatm)	N_2 (µatm)	He (atm absolute)
NPH/HHT	500	1.5	40		50	5–10	2
PNP	500	1.5	15		20	<5	2
AGCNR	400	2	40	0.2	20	<20	2

NPH: Nuclear process heat

HHT: High-temperature helium turbine systems

PNP: Prototype Nuclear Process Heat

AGCNR: Advanced Gas-Cooled Nuclear Reactor

The overall stability of the proposed helium environment that will be representative of the NGNP must be evaluated in order to ensure that testing performed includes the effects of NGNP helium at elevated temperatures. Therefore, testing of both the helium environment to be used for mechanical properties and general corrosion evaluations of the candidate materials to establish their overall compatibility with that environment are planned at temperatures up to at least 50 °C above the proposed operating temperature for the various metallic components.

5.1.3.1 Low Velocity Helium Loop. A closed circuit low flow velocity test loop has been designed and assembled at the INL, see Figure 45. This loop has the ability to expose coupons and mechanical test specimens in controlled impurity atmosphere at high temperature for long periods Details of the system are discussed in *Controlled Chemistry Helium High Temperature Materials Test Loop*, INL/EXT-05-00653[115]. A detailed report on the kinetics of gas reactions and potential degradation effects of NGNP helium on metallic alloys and low velocity helium loop was written in FY-06[116].No significant testing activity has been performed in this loop to date.

Figure 45. Photograph of the assembled low velocity controlled chemistry test loop.

5.1.3.2 Fabrication of Long Term Thermal Aging Test Specimens. One ¾ inch thick plate of ASTM B 168-01 was purchased from the Special Metals Corporation. Mechanical test bars and 6” square plates were machined from the plate. The bar specimen was machined to the dimensions in Figure 46. An additional 0.5” plate of the same ASTM specification was purchased from Special Metals Corporation and All Metals and Forge (produced by Haynes International) for additional 6” square aging specimens. All coupons were cut from plate stock using water-jet cutting. The design of these specimens is identical to that chosen for creep and creep fatigue testing to minimize potential for variability in test results arising from sample geometry. The small round tensile specimens will be tested in the low velocity test loop under a controlled helium environment for long times at elevated temperatures. The 6 inch square specimens will be thermally aged in air for long times at elevated temperatures.

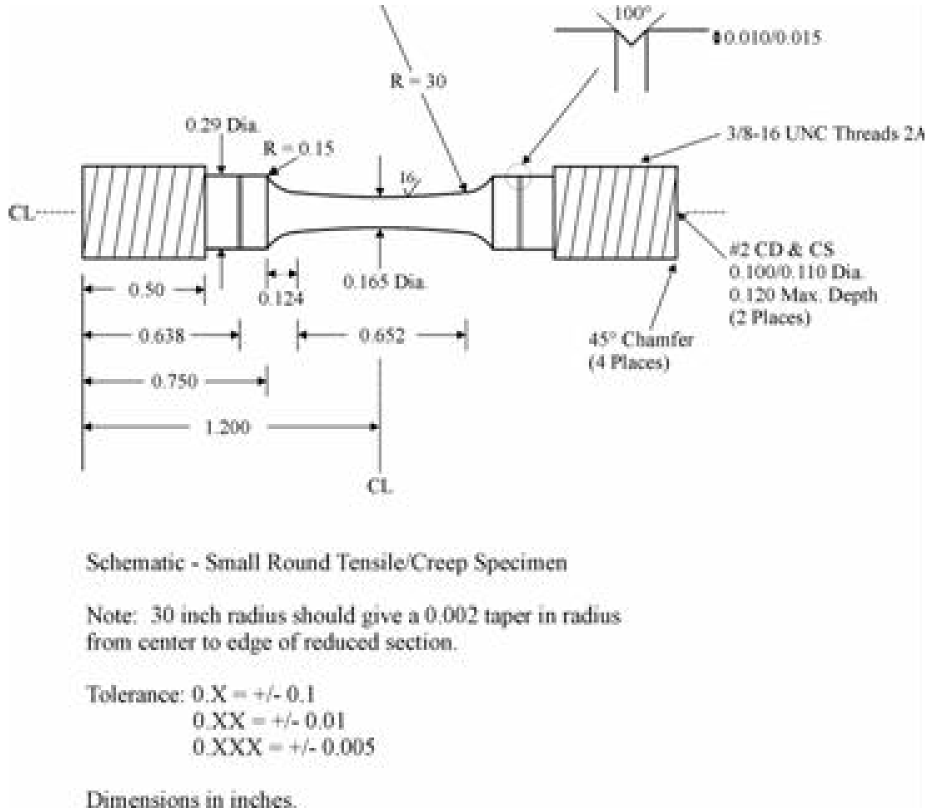


Figure 46. Schematic of the mechanical testing coupons used for long term aging and environmental exposure effects testing.

5.1.3.3 Long Term Thermal Aging and Environmental Effects Program. This work was documented in *Aging and Environmental Test Plan*, ORNL/TM-2005/523[117]. This joint ORNL/INL report proposed tests for alloys of interest. While Alloy 617 is of prime interest to the VHTR program, Alloy 230, Alloy 214, Alloy 800H, Alloy X, and Alloy XR have some properties favorable for use in this reactor. An array of mechanical properties tests and microstructural evaluations are proposed including hardness, tensile, fatigue, creep, creep-fatigue, and crack growth. These tests will be performed on as-received, heat-treated, aged, and environmentally exposed materials based on budget. Materials in the form of sheet, plate, tube in the welded and unwelded condition will be evaluated. These tests, which are by no means exhaustive, are intended to evaluate materials' performance, relate mechanical properties to microstructural features, and provide some data in support of the Alloy 617 Code Case.

A report will be written in FY-06 that documents the status of the upgrade of the centralized environmental creep testing facility at the ORNL but is not available for inclusion into this report.

Reports will be written in FY-06 that provide a summary and analysis of environmental and thermal aging testing performed on Alloy 617 (and its variants) and Alloy 230 at the INL and the ORNL, however, these reports are not available for inclusion into this report.

5.1.3.4 Alloy 617 Aging Effects. This work was documented in *A Review of Aging Effects in Alloy 617*, ORNL/TM-2005/511[118]. This review identified a number of issues of importance regarding aging phenomena in Alloy 617 that need to be addressed in further experimental and analytical studies.

Computational thermodynamics suggested that the equilibrium phases vary with temperature in the range of interest to the VHTR. The critical temperatures and the weight percentages of the equilibrium phases depend on the composition within the specified chemical ranges for the alloy. Additional studies are needed to further explore the variability in the content of the equilibrium phases associated with aluminum, titanium, carbon, nitrogen and boron. Both base metal and filler metal compositions need to be examined.

The kinetics of precipitation of non-equilibrium phases appear to vary from one set of experimental observations to another. Investigations are needed to establish the processes by which these phases are formed and replaced.

Hardness data were found to be valuable in mapping the kinetics of the property changes as a function of exposure history. Significant differences in the hardness values were observed from one heat to another with parallel exposures. Explanations were not put forward. Hardness testing should be encouraged as a practical tool to accompany other types of materials evaluation that involve high temperature exposures.

The room temperature tensile yield and ultimate strengths were increased by aging in the temperature range of 600 to 760°C. The rate of change in these properties was temperature-dependent and the maximum change varied with temperature and from one heat to another. The room temperature ductility decreased as the aged strength increased and the ductility decreased after long aging times at temperatures above 760°C with no apparent minimum. More research is needed to establish whether or not a ductility minimum could exist for very long times at VHTR temperatures. More experimental work should be planned to assess whether or not strength reduction factors will be needed. These factors would relate to “residual” strength at both high and low temperatures.

The fracture toughness, as measured by the Charpy V notch energy, was greatly reduced by aging to long times. Values ranged greatly from one investigation to another. For nearly identical exposure conditions, energies ranged from less than 10 to near 100 Joules. CVN testing is expected to be part of the research efforts on radiation effects so it may not be necessary to incorporate CVN testing into a separate aging program.

The information obtained from the literature review has not been verified by testing performed by the program to date.

5.1.3.5 Assessment of Past Helium Test Environments. This work was documented in *Potential Helium Test Environment for Next Generation Nuclear Plant Mater*ials, ORNL/TM-2005/92[113]. An analysis of potential helium environments for the NGNP was performed in FY-05. In the absence of designed system data with associated projected leakage rates, previously known environments and the factors that contributed to these environments were evaluated. Based on this evaluation, a possible range of composition for the helium environment for the NGNP has been chosen. The need for this earlier selection of a range of composition is necessitated by the requirement to begin testing possible materials for various applications that are outside of previously used materials/gas-composition/temperature operating conditions and/or test programs. It is anticipated that as the reactor system is specified with greater clarity, the reactor operating helium environment will be reviewed and the compositional range narrowed.

The nominal compositional range selected for materials testing for the NGNP was chosen as 400/2/40/0.2/20/<10 (μ atm) for $H_2/H_2O/CO/CO_2/CH_4/N_2$. In addition, it is recommended that test systems at various testing sites ascertain their ability to attain this compositional range at their test temperatures. Attainment of this compositional range may be accomplished by various approaches including varying flow rates, blending chambers, materials of construction of test chambers, and type and number of purification stations. Dynamic equilibrium within the test system should be assumed by achieving outlet and inlet gas composition within less than 10% of each other. Once dynamic gas/gas equilibrium has been demonstrated, it is necessary to establish the boundaries of the protective gas chemistry for the various selected materials, at least within the range of expected operating temperatures. It is anticipated that these protective chemistries will be within the gas compositional range selected in this report.

5.1.3.6 Review of Existing Data on the Effects of Impure Helium on Alloy 617. This work was documented in *Effects of Impure Helium Environmental Effects on Surface and Near-Surface Microstructures of Reactor Candidate Materials*, ORNL/TM-2005/525[119]. This review was performed in FY-05 to outline the available information on environmental effects of impure helium on Alloy 617, Alloy 800H, and Hastelloy X. These materials are of interest because past testing programs have established Alloy 617, Alloy 800H and Hastelloy X as reference materials for very high temperature applications in helium-cooled reactors.

After exposure, these materials demonstrate a fairly continuous surface layer, beneath which, there is an internally oxidized region, and a depletion zone. The microstructures consist of a mixture of primary carbides, similar to those observed in the as-received alloy, and intermediate to fine intra- and intergranular carbide precipitates (associated with aging and/or environmental effects). The carbides appeared to be preferentially precipitated along certain crystallographic directions. Differing results that arose from the various programs involve details of the surface scales formed, including the continuity and thickness of the scale and the phases present in the scale (type of oxide and/or carbide), and the amount of carburization observed after exposure of the same alloys in the different simulated helium-cooled reactor environments.

Much of these differences are probably associated with the differences in the actual test environments especially with respect to the degree of “dryness” of the environment. Environments depleted in water are likely to produce increased carburization. While such depletions are less likely to occur in operating reactors, the possibility of a lack of formation of continuous protective oxide layers, which would result in increased carburization, must be addressed. This is especially important for components such as heat exchangers, which would have thin cross-sections for which the environmental effects, as distinct from aging effects, will be most pronounced.

5.1.4 Develop and Qualify Metallic Alloys for Irradiation

In order to evaluate the irradiation effects of candidate alloys (such as the steels proposed for the reactor pressure vessel) under relatively low flux test reactor conditions, plans were made to design a new irradiation facility. This facility would replace the irradiation facility that was previously shutdown at the Ford Test Reactor at the University of Michigan. It was planned to design this irradiation facility to accommodate a relatively large complement of mechanical test specimens. The facility would include temperature control to allow for irradiation at the temperatures of interest and operate at a flux low enough to provide results representative of the conditions anticipated for the NGNP. The irradiation facility was anticipated to be a joint DOE facility with the NRC and be established in one of the material test reactors previously surveyed.

Having already identified reactors that were good candidates to host the low-flux RPV facility through contacts and visits in FY-04, the focus of work in this task in FY-05 had two primary parts:

1. To finalize the agreement between DOE-NE and the NRC Office of Research regarding how interaction of DOE- and NRC-sponsored RPV material experiments in the jointly sponsored irradiation facility would be coordinated, managed, and funded

2. Maintaining contact with and updating technical and financial input from potential candidate host reactors.

It was decided that DOE and NRC would require that a Memorandum of Understanding between the two sponsoring organizations and a draft MOU was prepared and meetings were held with both DOE and NRC to exchange information about technical approaches under consideration and relative roles of the organizations involved. An eventual agreement in principal was reached among the technical staff involved but concurrence on the legal aspects of the MOU was not reached.

Since final selection of the host reactor must await final agreement and issuance of the MOU, all further decisions on site selection and other preparations for the irradiation facility were deferred until FY-06, since they are heavily dependent upon the host reactor selection. During FY-05, a decision was made to close the Studsvik reactor in Sweden, so it was removed from consideration as a host for the irradiation facility. In contrast, information was received from the staff at the JRC reactor in Petten, Netherlands, that resulted in it being added to the list of primary candidates for the facility.

Progress was not made on this issue in FY-06 and this project is currently on hold pending the selection of the NGNP design and vendor team.

5.1.5 Composites

Fiber reinforced ceramic composites have been identified as possible material candidates for high temperature nuclear reactor components for the NGNP. Specific components of interest are control rod cladding and guide tubes within a VHTR design. These ceramic components require high thermal stability, good fracture toughness, and high irradiation stability during service. Current control rod design is composed of segments of ceramic composite tubes containing high neutron cross-section material (i.e. B_4C). Each segment (approximately 1-meter in length) will be joined to the next segment by an articulating joint to allow maximum flexibility of the rod during emergency use. The control rods will be used for both emergency shut-down of the reactor and controlling the active core.

Two ceramic composite systems have been identified as possible candidates for this specific application: carbon fiber reinforced carbon (C_f/C) and silicon carbide fiber reinforced silicon carbide (SiC_f/SiC) composites. C_f/C composites have been fabricated and used in a wide variety of different applications for decades, mainly in the aerospace industry. SiC_f/SiC have many similarities to the C_f/C composites but have only been readily available for a relatively short period of time. Both candidate composite systems were chosen due to their availability and past experience in irradiation environments.

The large market for carbon-based composites along with a wide variety of fabrication techniques to accommodate complex geometry components makes this material system a “mature technology.” There is little doubt that the control rod components consisting primarily of tubes and end-cap pieces can be fabricated using these materials. However, based upon fairly extensive studies on carbon-based materials, these composites have demonstrated irradiation instability over time and exposure levels in an irradiation environment. Even at relatively low dose levels (~ 7-8 dpa) the bundles of fibers within a composite can shrink or swell significantly creating large cracks and general degradation within the larger composite structure.

Therefore, while there is no doubt that C_f/C composites will perform sufficiently well at beginning of life they will eventually need to be replaced as the material properties become compromised over time and dose. It has been estimated that C_f/C composites will need to be replaced at least three times over the lifetime of the VHTR (nearly 60 years and up to 30 dpa).

SiC_f/SiC composites, however, have been shown to be structurally stable to dose levels where C_f/C composites become significantly compromised (~ 8 dpa). It is thought that this material system may be stable enough to withstand a dose of 30 dpa, or the equivalent of the lifetime in the NGNP. However, the challenges for SiC_f/SiC composites lie in their fabricability, material supply, and the cost of manufacture. Because of these challenges and other considerations, it was decided in FY-06 to drop these materials from the program.

5.1.5.1 Summary of SiC Tube Architecture and Fabrication. This work is documented in *Summary of SiC Tube Architecture and Fabrication*, ORNL-GEN4/LTR-05-007[120]. As a part of the FY-05 NGNP Composites R&D task activities, the Phase-I SiC/SiC composite materials were fabricated following the successful completion of selections of appropriate tube architecture and composite's constituents, definition of material specifications, and designing of a tubular test specimen for the elevated temperature axial tensile test. The Phase-I materials include small diameter double-shouldered tubes and flat plates of bi-axially braided Hi-Nicalon™ Type-S / multilayered PyC/SiC interphase / CVI SiC matrix composite (Reference NGNP-Grade) for baseline properties characterization and tube - plate properties correlation study, and small diameter straight tubes of bi-axially braided Hi-Nicalon™ composite, which is in otherwise identical with the Reference NGNP-Grade, in support of ASTM / ISO testing standard development.

5.1.5.2 Status of Irradiation of Multilayer SiC/SiC and Graphite Composites. This work is documented in *Status of Irradiation of Multilayer SiC/SiC and FMI-222 Graphite Composites*, ORNL/TM-2005/508[121]. The objective of an irradiation experiment is to prove mechanical integrity retention of the advanced nuclear grade SiC/SiC composite and to provide side-by-side comparison of irradiation effects of candidate SiC/SiC and graphite composites for NGNP control rod application. More specifically, the goal of this task is to obtain data of flexural strength, proportional limit stress, fracture mode, elastic modulus, swelling, and thermal conductivity of Hi-Nicalon™ Type-S multilayered-interphase CVI SiC-matrix composite and FMI-222 pitch fiber graphite composite after neutron irradiation to fluence levels of 10dpa and greater at elevated temperatures in the HFIR. All the Phase-I irradiation capsules, except one that failed leak test, have been constructed and started irradiation in the peripheral target tube (PTT) facility of HFIR before June, 2005. A subset of the 10 dpa capsules have been completed and disassembled and post-irradiation examination (PIE) was completed in FY-06. A report that documents the results of the PIE will be written in FY-06 but is not available for inclusion in this report.

A report that documents the experiment plan and specimen design of composite irradiations planned in the future for the ATR will be written in FY-06 but was not available for inclusion in this report.

5.1.5.3 Environmental Effects on SiCf/SiC Composites. It is assumed that the fundamental irradiation response of the microstructure will be similar for all preform architectures and component geometries. However, using different preform architectures (i.e., weave angles, fiber tow counts, weave structures, etc.) can lead to differences in the macroscopic mechanical responses in the composite structure due to infiltration efficiency, fiber bending stresses, or matrix/fiber interface characteristics. The environmental conditions these materials will be subjected to may also change the overall creep response of the composite (i.e. creep crack growth for fiber reinforced materials).

PNNL has extensive experience in environmental degradation of SiC. They have developed a creep crack growth model to predict the environmental factors on the overall creep of the SiCf/SiC composite structures. To improve the accuracy of the model predictions a limiting "reactor environment" for elevated temperature tests must be determined. Most likely, the limiting environmental species in the He loop will be the H_2/H_2O ratio. Assuming these species are the most damaging to the composites PNNL will determine the degradation potential for various H_2/H_2O ratios using both modeling and experimental tests.

Slow crack growth tests have been performed in high purity Argon (expected to be no different than He) at 1100º C, 1150º C, 1200º C, and 1300º C. These tests require analysis for crack growth rates but it was observed that failure for these Type-S fiber composites at 1300º C was very rapid, which suggests an upper temperature limit below 1300º C for this composite system.

PNNL is using materials that were on hand and purchased in 2004 from GE Power Systems. The SiCf/SiC materials are 8-harness satin weave, 8 ply, Hi-Nicalon Type-S fiber composites. They are intended to be a surrogate until the newer Hyper-Therm materials arrive. The 4-point bend slow crack growth tests were all performed on un-notched bend bars and can be analyzed to give crack growth rates in Argon due to fiber creep. An activation energy analysis will be performed and compared to creep of single Type-S SiC fibers.

Studies will continue up to 1400°C in pure Argon or pure He. Then, testing will begin using impure He that is tailored to simulate actual VHTR operational environments. A crack growth model will be developed to explore crack growth and time-dependent bridging in Type-S materials.

These studies will continue through FY-06 and not be continued in FY-07. A summary of the work performed in FY-06 is planned but has not been issued to date.

5.1.5.4 Composites Development Activities. The geometry effect in composite tubes were documented in *Status of Geometry Effects on Structural Nuclear Composite Properties*, INL/EXT-05-00756[122]. Irradiation creep has been identified as a primary degradation mechanism for the structural ceramic composites being considered for control rod applications within the VHTR design. While standard sized (i.e. 150-mm long or longer) test specimens can be used for baseline non-irradiated thermal creep studies, very small compact tensile specimens will be required for irradiated creep studies. It must be demonstrated that the smaller test samples used in an irradiated study will adequately represent the true response of larger composite tubes used for control rod applications. To accomplish this, two different test programs are being implemented to establish that small, flat test specimen are representative of the mechanical response for large, cylindrical composite tubes; a size effect study and a geometry effect study. This is discussed further in the report noted.

The status of creep testing of composite tubes, is documented in *Creep of Structural Nuclear Composites*, INL/EXT-05-00747[123]. One of the primary degradation mechanisms anticipated for composite core control rod components is high temperature thermal and irradiation enhanced creep. As a consequence, high temperature test equipment, testing methodologies, and test samples for very high temperature (up to 1600º C) tensile strength and long duration creep studies have been established. A more general discussion of these activities is documented in *Structural Ceramic Composites for Nuclear Applications*, INL/EXT-05-00652[124].

5.1.5.5 Testing Plans for Failure Mode Assessment of Composite Tubes Under Stress. This work is documented in *Summary of Testing Plans for Failure Mode Assessment of Composite Tubes Under Stress*, ORNL-GEN4/LTR-05-002[125].

Potential failure modes for SiC/SiC composite control rod sleeves have not been assessed before, due both to the lack of general comprehensive property data for nuclear-grade SiC/SiC composite parts in cylindrical geometries, and to the lack of dependable engineering design and accident scenarios for the relevant reactor systems. This report briefly summarizes the typical properties of candidate SiC/SiC composite, preliminary analysis on stress state in control rods, considerations of potential failure modes, mechanisms, and the desired test plan for the SiC/SiC control rod sleeves and guide tubes

The maximum axial tensile stress in a control rod sleeve due to the dead-weight is estimated to be between 1.25 - 2.5 MPa depending upon radius and wall thickness. However, transient stresses may occur if a control rod is stuck in the core and the operators actively pull on it. The maximum applied stresses are therefore unknown but should be determined during the testing program (i.e. evaluation of residual strength after the creep test might be required to determine an upper stress limit in the event of a stuck control rod).

Additional stresses will be imposed due to the thermal gradients across the axial length of a rod section. Thermal gradients may impose stresses in two different mechanisms; thermal stresses in a usual meaning that is caused of differential thermal expansion, and the internal stresses developed by differential irradiation-induced swelling, which is significantly temperature dependent for SiC. The maximum thermal gradient in practical applications is not known yet. A preliminary analysis was performed using ABAQUS code and assuming the design data from the HTTR, Japan, in which the largest thermal gradient of ~3.3 K/cm occurs at the vertical position of the topmost fuel element[126].

The maximum radial (trans-thickness) temperature gradient in control rod sleeve wall is expected to be small, according to the design data of HTTR[126] ($\Delta T << 10$K across 3.5mm-thick wall made of superalloy 800H). However, the maximum radial temperature gradient in SiC/SiC sleeve of NGNP may be more significant because of the potentially larger power density and the lower trans-thickness thermal conductivity (> ~5 W/m-K for SiC/SiC at ~1000°C[127] compared to ~25 W/m-K for alloy 800H at ~800°C[128]), and thus this will have to be addressed more precisely when the design data for NGNP are made available. Also, transient thermal stresses in occasions such as reactor start-up, scram, or any other accident scenarios will have to be addressed.

Radial or hoop stresses may also be imposed from the thermal and irradiation-induced deformation of neutron absorbing materials contained. There should be no gas pressure within the control rods. The vast majority of gas released from B_4C will be He, which does not require containment. In case B_4C pucks are used, irradiation-induced swelling of B_4C may create significant stresses at around the contact points. Such stresses may be minimized by employing geometries like spherical pebbles.

For the case of guide tubes, the static axial tensile stress will be negligibly small. In the event of a stuck control rod, tensile stress corresponding stress in the control rod will occur in a guide tube. The maximum applied stresses are therefore unknown but should be determined during the testing program. Radial or hoop stresses will also be imposed from the irradiation-induced densification or swelling and differential thermal expansion of graphite blocks. These stresses are again difficult to calculate since the total graphite densification / swelling and the tolerance between the tubes and graphite blocks are not known. It is preferred that the guide tube tolerance is designed to impose stresses to guide tubes no more than a few MPa.

As suggested by the result of preliminary stress analysis, it is likely that the magnitudes of static stresses in the control rod sleeves and guide tubes are sufficiently lower than the typical matrix microcracking stress of the nuclear-grade SiC/SiC composites. At applied stresses below the matrix microcracking stress, CVI SiC/SiC composites without exposed interphase do not undergo either static or dynamic fatigue. Therefore, static stresses applied to the control rod sleeves during normal operation are extremely unlikely to cause failure of sleeves, which are structurally sound, by mechanisms other than irradiation creep. In the off-normal event of stuck control rod, the failure mode that has to be primarily considered is the axial tube tensile. It will be possible that tube bending stresses occur and contribute to failure in a seismic event. Once matrix microcracks are introduced during a recovery operation in such events, slow failure mechanism such as interphase recession and/or fiber creep may limit the residual life time of the component.

Irradiation creep is the only potential failure mechanism for CVI SiC/SiC composites under stresses well below the matrix micro-cracking stress at temperatures of interest. Presently, the irradiation creep strain rate of SiC is almost unknown because of very limited availability of experimental data; neutron-irradiation creep compliance of CVD SiC at >1100°C was very roughly estimated to be ~10^{-12} Pa^{-1} dpa^{-1} by Price[129], whereas proton-irradiation creep compliance of SCS-6 CVD SiC fiber was measured to be ~10^{-11} Pa^{-1} dpa^{-1} at 450 - 1200°C by Scholz[130,131,132]. This corresponds to 0.1% strain at 2.5MPa and 40 dpa. SiC/SiC composites may undergo significantly different irradiation creep deformation from CVD-SiC. The NGNP SiC/SiC composite R&D program had planned to provide the first dependable irradiation creep data; however, as noted previously, this program was terminated in FY-06.

5.1.5.6 Survey of Potential Vendors for C/C Composites. This work was documented in *NGNP Carbon Composites Vendor Survey*[133]. The report contains Export Controlled Information and therefore will not be discussed here.

5.1.5.7 NGNP Carbon Composites Literature Review. This work was documented in *NGNP Carbon Composites Literature Review and Composite Acquisition*[134]. This report details the research performed over the last few decades on the irradiation response of various types of C/C composites and makes recommendations into the types of composites that would be best for dimensional stability. In addition, this report details the current status of purchasing C/C composites for evaluation.

5.1.5.8 ASME Code and ASTM Standards Activities. A report was written in FY-05 that reviewed the development of standardized test methods, design codes and databases for SiC/SiC[135]. A report, *Roadmap to NRC Approval of Ceramic Matrix Composites in Generation IV Reactors*, INL/EXT-06-11425[136] was written in FY-06 to further explore and determine a roadmap for ASME Code and ASTM standards activities related to nuclear composites and NRC approval for use of these materials in the core of the NGNP. This report discusses the steps and the process required but does not establish an actual roadmap or a projected timetable for completion of this process.

This result and other studies have strongly inferred that a roadmap associated with these activities and possible criteria for NRC acceptance of these materials cannot be established at this time and therefore it is unlikely that composites will be used in initial critical core applications in the NGNP.

5.1.6 Studies

Several studies were performed in FY-06. These studies were performed to clarify several issues that resulted from a review of the NGNP Program. The purposes of these studies are given below:

1. Determination of the need for use of composites in critical NGNP core applications particularly associated with the control rod drive application

2. An examination of the issues associated with materials selection of the reactor pressure vessel and Class 1 boundary

3. An examination of the use of a intermediate heat exchanger in the NGNP and the issues associated with materials selection

4. Performance of a conceptual design study of a test loop to be to be used to characterize the performance of materials and critical NGNP components

5.1.6.1 Composites Study for NGNP Core Applications. The results of this study were documented in *Analysis of Potential Materials for the Control Rod Sleeves of the NGNP*, INL/EXT-06-11614[137]. The recommendations and conclusions from this report are currently being reviewed and final information is not available for inclusion in this report.

5.1.6.2 Examination of NGNP RPV and IHX Issues. The results of these studies, which were being performed by ANL, are not available for inclusion into this report.

5.1.6.3 Test Loop Conceptual Design. A report that documents a conceptual design for a test loop and a preliminary cost estimate for further work that needs to be performed prior to construction was prepared in FY-06[138]. This study had a number of objectives that collectively constitute the development of an early stage conceptual design for a high-temperature gas test loop. The objectives include the following:

1. Investigate existing gas test loops to determine their capabilities and how this system might best complement them

2. Develop a preliminary test plan to help identify the performance characteristics required of the test unit

3. Develop test loop design requirements

4. Identify potential test loop locations with an emphasis on characteristics of the building or site rather than on naming a specific location. It was assumed that the test unit would be built at INL and would be operated by INL staff

5. Develop a conceptual design including process flow sheet, mechanical layout, and equipment specifications and costs

6. Develop a preliminary test loop safety and environmental plan.

This report documents the results of these tasks which were primarily accomplished by Brayton Energy, LLC of Hampton, New Hampshire. INL engineers provided project management, technical guidance and review, and input on tasks requiring knowledge of INL design and operational requirements.

The broad requirement is for testing of components to be installed in the primary and secondary coolant loops of the NGNP. The primary loop operates with pure helium at 7 MPa (about 1000 psi) and 950º C (about 1740º F). The secondary loop operates with a helium-nitrogen mixture at similar maximum conditions. The leading test priority is for various designs of the intermediate heat exchanger (IHX). Further test requirements are for isolation valves for emergency shutoff of the primary loop, various construction materials, novel fabrication and joining methods, insulation concepts for the high temperature piping, and instrumentation strategies including nondestructive evaluation methods for online monitoring of the IHX or valves. A detailed description of anticipated tests is presented in Appendix A of the report.

To flexibly provide a broad range of test scenarios, the test loop configuration will feature a single primary loop with a centralized heat source feeding three parallel secondary loops. The primary and secondary test loops will be manually reconfigurable without the use of high-temperature valves which are themselves a target of development. Efficiency of piping changes is a design priority to allow a timely succession of tests, many lasting only hours or days.

The centerpiece test articles will be intermediate heat exchangers, likely one or two full-scale core units (a small fraction of the number required for a commercial system). The primary test issues are heat transfer performance and internal stresses from thermal expansion. Analysis of performance and durability require the measurement of internal temperature gradients during steady-state and transient operation. Representative fault conditions and transient maneuvers will be simulated, during which potentially damaging stresses may be imposed on the IHX test article. The strategy for core life prediction is to validate finite element computer models against measured thermal and strain profiles, with these models then providing analysis tools covering a range of scenarios beyond those directly simulated.

Test and qualification programs for military high temperature heat exchangers have been reviewed for this report. The proposed High Temperature Gas Loop (HTGL) plan builds upon this experience and further recognizes the great potential value of three advanced diagnostic capabilities beyond standard temperature, strain, and pressure measurements. These capabilities are thermal imaging telemetry, in situ precision coordinate measurement for monitoring of dimensional changes, and leak detection online at full test conditions. These methods, if developed as part of this program, offer the prospects of both better quality data during IHX testing and, in eventual commercial service, ways to monitor IHXs online for early warning of degradation or incipient failure.

Testing of isolation valves involves several challenging requirements. A full-scale high-temperature valve for a NGNP hot piping system is 1 to 2 meters in diameter, requiring special installation and large actuation drives. Each of the three test bays is equipped to handle this scale of equipment. The test objective is primarily to verify sealing while avoiding high temperature galling. Operation at high temperature and contact pressure in a high purity helium environment is known to initiate “helium-welding,” a form of diffusion bonding between metal parts. The expected test format requires repetitive actuation of the valve under hot flowing conditions with measurement of leakage and valve rotational torque. Some extended dwell periods at high temperature and pressure are also anticipated to fully qualify this equipment. Since the HTGL configuration provides three test bays, extended operation of NGNP valve and IHX tests may run concurrently.

A spectrum of extended endurance tests for materials samples is also planned for investigation of corrosion, fatigue, and creep. These tests will be performed on fabricated parts such as valve bodies, gas turbine engine parts, or on simple weld and braze coupons. Test compartments located in the primary and secondary loops provide exposure to a range of gas velocity and temperature. Tests requiring altered gas compositions—for instance the addition of carbon monoxide, water, or graphite particles—are also planned for a secondary loop, to keep from possibly damaging the primary loop equipment.

Because the test loop itself operates at conditions equal to or exceeding those of the NGNP system, the loop has many of the same novel design problems. A number of the components in the loop will be new designs, new applications of conventional designs, or use of new materials. Though these components will be installed with the expectation that they will perform as needed, they will be undergoing *de facto* usage testing of their own. A partial list of examples includes the following:

1. Sealing systems for hot high-pressure helium

2. Pipe design with internal insulation to reduce the temperature at the pressure boundary (This has complications related to insulation performance in pressurized helium, installation of a flow liner that allows the insulation layer to breathe, and insulation mating methods at pipe flanges.)

3. Circulators for high temperature helium.

The French HELITE test loop program recognized this problem and had, as part of its own design effort, a development and testing effort on the equipment that makes up the test loop (as opposed to the IHX test article itself). To address this technical risk, the program plan should include an equipment development program including comprehensive equipment testing at the manufacturer. The system design should include monitoring instruments installed on such assemblies, and a layout that allows relatively simple replacement of such components as alternative concepts for them are developed.

This system will have significant instrumentation and data collection needs. The basic operation of the test loop requires automated monitoring and control of various flows, temperatures, pressures, and equipment speeds to provide well-regulated test conditions. The control system will be capable of recognizing off-normal conditions and safely shutting down the unit without operator intervention when the conditions warrant. The key control variables will be scanned at relatively high frequency to quickly detect and correct any aberrant behavior.

The test function of the unit calls for comprehensive data collection, online analysis, presentation, and archiving capabilities. In the IHX tests, a large number of temperature and strain data points must be collected, perhaps hundreds of channels of data. In the transient IHX tests, these channels must be logged at a rate of about once a second for a number of hours. All three test bays must have similar capability of collecting data from various types of instruments which will be provided as part of the test articles. The data collection system (as well as physical access to the system) must provide individual security to each test bay because users might require protection of the performance data from their proprietary designs.

Measuring some of the required data will involve more than just installing the proper type of direct sensor. IHX and valve testing will require leak checking, in the first case between the primary and secondary loops while both are hot and pressurized, and in the second case through a closed valve with 8 MPa pressure behind it and atmospheric pressure downstream. Methods to do this must be conceived and tested as part of the overall project design. One concept for the IHX leak testing is to use gamma-emitting radioactive isotopes of xenon or krypton as inert tracer gases. The radiation from these can be detected with great sensitivity through the system's walls while it is operating. The same leak checking system might be used with valves, or they could use a standard pressure rise test if the necessary piping is installed. Other minor systems will be needed to monitor and control gas composition in the secondary loops for long-term corrosion testing. There trace amounts (2-50 ppm) of gases like CO, CO2, H2O, CH4, and H2 will be maintained in the secondary gas mixture to simulate the atmosphere generated by reaction of the graphite in a NGNP reactor core. Production of a gas stream containing low levels of suspended graphite particles will also be needed to test for erosion/corrosion. Although both gases and graphite particles simulate conditions in the NGNP's primary loop, it is less risky to simulate them in the test system's secondary loops at similar temperature and pressure.

A somewhat more involved test program planned for several years in the future covers the interface of the high temperature gas reactor with the thermochemical sulfur-iodine process for making hydrogen from water. In this process, heat at about 850º C is used to decompose sulfuric acid vapor. That process step (which includes vaporization of liquid sulfuric acid, decomposition of the vapor to SO3, and decomposition of that to SO2 all in one vessel) will be validated at pilot scale in this test loop. Since this will be a process test rather than just a component test, additional piping, pumps, and chemical storage will be required in the test bay and they will be provided by the test article developers. The test bay, however, must be designed now with space for this equipment and with walls, floors, and ventilation appropriate for acids and toxic gas handling. After the sulfuric acid decomposer unit has been tested—and quite speculatively at this point—the high temperature test loop might be considered as the thermal energy source for an integrated test of the complete sulfur-iodine process. The significance of this use is simply that, to keep the option open, the test loop should be laid out to allow installation near the primary helium loop of a process system at least as large as the test loop itself.

A similar alternative concept for one of the secondary loop test bays is to characterize the use of molten salt as the working fluid in an intermediate heat transfer loop. Molten salts are potentially advantageous if the heat must be transported over a great physical distance since pumping costs are lower for liquids than for gases. The system to be tested is not well-defined at this point, but the requirements for space and spill control should be comparable, though not necessarily identical to, those for the sulfur-iodine testing.

To develop the specifications for this HTGL, the current NGNP and other high-temperature reactor initiatives in the U.S., Japan, China, the Republic of South Africa, France, the Netherlands, and Germany were reviewed. The survey considered three classes of applications outlined in Table 31. Both direct and indirect power conversion systems can also use an IHX to provide high temperature heat for hydrogen or process heating applications[139]. As it is not apparent which of these applications will advance or fail during research and development, planning must allow for testing each of them.

The basic design conditions for the test loop are listed in Table 32. These were developed after consideration of the detailed data in Table 33, which lists some critical design parameters for several IHX candidates under development.

Table 31. The HTGL will allow testing of all anticipated NGNP categories and processes.

Category	Cycle	Secondary fluid	Temperature °C
Power Generation			
	Steam turbines	Steam	<650
	Gas turbines — direct and indirect	H_2	<900
	Gas turbines — supercritical CO_2	CO_2	<700
	Gas turbine plus steam turbine bottoming cycle	He + N_2	<900
Hydrogen production			
	Sulfur Iodine	$H_2SO_4 + H_2O$	<900
	High-Temp Electrolysis	Steam	<950
	Solid Oxide Fuel Cell	Steam	<950
	Calcium-Bromine cycle	$CaBr_2 + H_2O$	<750
	Potassium hydroxide	$KOH + H_2O$	<800
Process heat			
	Heavy oil extraction	Steam	<400
	Oil shale/sands extraction[140,141,142]	Steam	<950

Table 32. Basic design conditions for the primary and one secondary loop of the HGTL.

Primary Loop	Values
Composition	pure He
Mass flow	0.80 kg/s
Supply temperature to IHX	950ºC
Return temperature from IHX	550ºC
Secondary Loop	**Values**
Composition	80% N_2 , 20% He
Mass flow	2.12 kg/s
Supply temperature to IHX	500ºC
Return temperature from IHX	900ºC

Table 33. Survey summary results showing temperature, pressure, and mass flow rate for primary and secondary loops.

Industrial sponsor/ Source	**INL study**	**INL study**	**INL study**	**AREVA IHX**	**PBMR, Westing-house, BNFL**	**KAPL/ MIT**	**MIT**	**Chinergy (China)**	**MHI**	**MHI**	**PBMR Inc.**
Application	**Direct gas turbine**	**Power & hydrogen**	**Power & hydrogen**	**Indirect power**	**Direct gas turbine**	**Direct gas turbine**	**Indirect Power**	**Indirect Power**	**S-I SO_3 Decomp**	**H_2SO_4 Decomp**	**Oil Shale Recovery**
Primary gas delivery temp., °C	900	881	881	850	510	505-900	700	700	880	688	600-1000
Fluid (Secondary loop)	He	molten salt H_2S0_4	He	20%He/ 80% N_2	He	He	CO_2	H_2O (steam)	688		molten salt
Primary side IHX Inlet Press., MPa	7	7	1.95	6	8.94	4.5	20	6.90	2.1	2.1	9
Primary mass flow @ 2MW, kg/s	1.07	1.07	1.07	0.84	0.998	1.587	2.5		2.0	1.6	1
Primary side outlet temp., °C	575	548	520	443	139	190	547				
Secondary side inlet press., MPa	1.95	7	0.1	5.5	8.9	9	20	6.90		4.1	
Secondary mass flow @ 2MW, kg/s	1.070	1.070	1.070	2.21	1.05	1.587	11.1				
Secondary side inlet temp., °C	558	530	501	350	106	155	397		527	391	
Secondary side outlet temp., °C	883	882	800	800	498	470	550	600	850	527	
Note refs.	Direct electric, parallel IHX[143]	Indirect electric, parallel SHX. For General Atomic's S-I H_2 process[144,145]	Indirect electric, parallel SHX. S-I process modified to reduce IHX stress[146,145]	Indirect electric, parallel IHX. Near-term primary temperature is 850˚C, goal is 950˚C (VHTR)[147,148]	Eliminates the IHX, but incorporates a recuperator. Recuperator specification is listed[149,150,151]	KAPL nuclear submarine concept employs a direct cycle gas turbine similar to that of PBMR Inc.	Super-critical CO_2.[152,153]	A joint venture of INET with China Nuclear Eng'g & Construction Co.[154]	Small scale ceramic and metallic units tested[155,156]	[156,157]	[154,158,159]

5.2 Discussion of Planned FY-07 Materials Program

The planned activities listed for FY-07 are dependent on NGNP Materials Program budget and other factors which are currently not clarified. The activities listed are those that have been previously submitted to DOE for planning purposes and are considered important activities to support the NGNP Program.

5.2.1 Graphite Selection Strategy

This activity is complete and further work in this area is not planned in FY-07.

5.2.2 Graphite Irradiation Experiments

1. Complete AGC-1 gas control system design, fabrication, installation in the ATR and testing. Completion of this work is dependent on continued support from ATR operations and adherence to the planned AGC-1 schedule given in Section 5.1.1.2.1.

2. Secure an AGC-1 fabrication facility and develop required assembly fixtures

3. Complete AGC-1 specimen pre-irradiation characterization activities

4. Complete the fabrication and assembly of the AGC-1 experiment such that it is ready for insertion in the ATR in November 2007. Completion of this work is partially dependent on continued support from ATR operations and adherence to the planned AGC-1 schedule given in Section 5.1.1.2.1.

5. Continue to work with ATR operations regarding the design, fabrication and installation of a canal sizing and transfer apparatus that is required following irradiation of the AGC-1. Goal of this work is to perform a final design review and operational checkout of the apparatus. Completion of this work is dependent on continued support from ATR operations and adherence to the planned AGC-1 schedule given in Section 5.1.1.2.1. Delay in the irradiation of the AGC-1 would likely delay this effort.

6. Complete and document qualification of the GE-2000 cask for receipt and handling capabilities at the ORNL, MFC the ATR facilities. It is currently planned to perform irradiation, sectioning and loading of the shortened AGC-1 capsule into the GE-2000 cask following irradiation at the ATR. It is planned to transfer the capsule to MFC for disassembly. It is planned that MFC would load the specimens following capsule disassembly into the GE-2000 cask for shipment to ORNL. It is planned that ORNL would perform PIE on the specimens.

7. Complete HTV-1 and 2 specimen pre-irradiation characterization activities.

8. Complete construction and qualification of HTV-1 and 2.

9. Complete irradiation of capsules HTV-1 and 2 in HFIR at the ORNL.

10. Complete final design of HFIR high dose tensile creep capsule.

5.2.3 Graphite Qualification and Licensing

1. Draft NRC White Paper on initial irradiation timeline and support development of licensing strategy for the NGNP first core graphite components. Goal is to issue a report on NGNP irradiation and licensing strategy for NGNP first core graphite components with cooperation from the NRC.

2. Develop draft production specifications and vendor qualification plans for NBG-18 and PCEA grade nuclear graphites. This assumes that the nuclear graphite vendors that develop these graphite grades are willing to collaborate on these activities.

3. Collaborate with reactor vendors regarding graphite components dpa limit.

5.2.4 Graphite Billet Characterization Activities

1. Continue graphite billet characterization plan and purchase of addition billets as appropriate.

5.2.5 Graphite Modeling

1. Continue CARES development and incorporate failure theories into CARES.

2. Perform multiaxial modeling testing on NGNP graphites.

5.2.6 Graphite Program Technical Oversight and Coordination of Working Group

1. Provide technical oversight of the NGNP Graphite Program.

2. Coordinate the activities of the GIF M&C PMB Graphite Working Group

5.2.7 ASTM Standards and ASME Code Activities Associated with Graphite

1. Continue to support the ASME graphite core support working group.

2. Prepare final version of ASTM DO2.F standard test method for air oxidation of graphite that incorporates round robin test results. Submit for subcommittee ballot.

3. Prepare final version of ASTM DO2.F standard test method for fracture toughness of graphite that incorporates round robin test results. Submit for subcommittee ballot.

4. Develop a draft ASTM standard test method for ultrasonic non-destructive examination (NDE) of graphite components.

5.2.8 High Temperature Design Methodology

1. Continue creep and creep-fatigue testing of Alloy 617 and Alloy 230.

2. Continue to evaluate the effect of joints on the properties of Alloy 617 and Alloy 230

3. Procure sheet product form of Alloy 617 and Alloy 230 for testing and evaluation.

4. Perform evaluations on Alloy 617 and Alloy 230 sheet materials directed at the use of these materials for a compact heat exchanger IHX design.

5. Continue to assess effects of off-normal heat treatment and welding procedures on creep strength of Grade 91 steel.

6. Initiate study on thick section properties of Grade 91 steel.

7. Perform testing in support of ASME Section III, Subsection NH expansion of Alloy 800H temperature range.

8. Continue constitutive model development for Alloy 617 and initiate modeling for Alloy 230.

9. Continue to generate additional data and improve understanding of low temperature Grade 91 creep and negligible creep rules for this alloy.

10. Continue development of simplified methods for deformation based criteria with expansion of NGNP relevant conditions.

5.2.9 Aging and Environmental Effects Studies

1. Perform mechanical property testing on Alloy 617 samples exposed to long term aging conditions at elevated temperatures.

2. Complete aging studies on Alloy 617 and initiate aging studies on Alloy 230.

3. Assess effects of long term exposure to air, helium and other inert gases at very high temperatures on Alloy 617 and Alloy 230 using microstructural examination techniques.

4. Perform creep testing in helium on Alloy 617 and 230.

5. Perform testing on samples fabricated from Alloy 617, Alloy 617 CCA and Alloy 230 in the low velocity helium test loop in various controlled atmospheres. Perform post exposure mechanical property and microstucture examination on the specimens.

5.2.10 ASTM Standards and ASME Code Activities Associated with HTDM

1. Continue to provide support for ASME Subsection NH.

5.2.11 ASTM Standards and ASME Code Activities Associated with Composites

1. Develop ASTM standards in support of planned C/C composite irradiations. This activity includes the development of fracture toughness testing standards.

5.2.12 Planned Composite Irradiations

1. Design, fabricate and characterize C/C tensile samples for Phase 2 HFIR irradiations.

2. Initiate HFIR Phase 2 irradiations on C/C composites.

3. Perform PIE on HFIR 15dpa Phase 1 composite irradiations.

5.2.13 NGNP Component Testing Facility

1. Perform a preliminary design and design review of the NGNP component test facility.

5.2.14 University Programs

1. Continue support to the University of Michigan for NGNP alloys for high temperature service.

5.2.15 Manage the Materials Program

1. Manage the materials program and the Materials Review Committee (MRC).

2. Revise the NGNP Materials Program Plan.

5.3 Discussion of Planned Materials Program Plan Beyond FY-07

The NGNP Materials Program workscope has not been determined for FY-08 and beyond. It is assumed that the workscope will be strongly influenced by the results of previous workscope performed, NGNP trade studied performed, the NGNP vendor team selected, the NGNP design selected and other important considerations such budget that have currently not been defined. Therefore, it was determined that it is not reasonable to carefully examine workscope for the out years until the program is more thoroughly defined. Much of this information is expected to be available near the end of FY-07; therefore, it is reasonable to assume that the next program plan revision will address this issue.

6. Program Cost and Schedule

This section contains schedule information related to the overall NGNP Program (Section 6.1) and the planned NGNP Materials Program Schedule and costs for FY-07 (Section 6.2). This information contained in this section is subject to change and should be used for general planning purposes only.

6.1 NGNP Program Schedule

The revised NGNP Program Schedule is not available at this time.

6.2 NGNP Materials Program Cost and Schedule Information

The information provided uses the same format given in Section 5.2 and is summarized in Table 34. The tasks are not listed in any particular order of priority. Priority will be determined by program management during FY-07.

Table 35. NGNP Materials Program FY-07 Task, Cost and Schedule Information

Reference Information From Section 5.2	**Report**	**Level**	**Performing Organization**	**Estimated FY-07 Cost ($K)**	**Planned FY-07 Schedule. Report Due date is Shown in Bold**											
					Oct-06	**Nov-06**	**Dec-06**	**Jan-07**	**Feb-07**	**Mar-07**	**Apr-07**	**May-07**	**Jun-07**	**Jul-07**	**Aug-07**	**Sep-07**
5.2.2-1	Yes	2	INL	$665	x	x	x	x	x	x	x	x	x	x	x	X
5.2.2-2	No	3	INL	$75	x	x	x	x	x	x	x	x	x	x	x	x
5.2.2-3	Yes	3	ORNL	$200	x	x	x	x	x	X						
5.2.2-4	No	2	INL,ORNL	$500	x	x	x	x	x	x	x	x				
5.2.2-5	Yes	3	INL	$350	x	x	x	x	x	x	x	x	x	x	x	X
5.2.2-6	Yes	3	ORNL, INL	$200	x	x	x	x	x	x	x	X				
5.2.2-7	Yes	2	ORNL	$300	x	x	x	x	x	x	x	x	x	x	x	X
5.2.2-8	No	2	ORNL	$300	x	x	x	x	x	x	x	x	x	x	x	x
5.2.2-9	No	2	ORNL	$300	x	x	x	x	x	x	x	x	x	x	x	x
5.2.2-10	Yes	3	ORNL	$250	x	x	x	x	x	x	x	x	x	x	X	
5.2.3-1	Yes	2	INL	$50	x	x	x	x	x	x	x	x	X			
5.2.3-2	Yes	2,3	INL,ORNL	$450	x	x	x	x	x	x	x	x	x	x	X	
5.2.3-3	No	2	INL,ORNL	$100	x	x	x	x	x	x	x	x	x			
5.2.4-1	Yes	2,3	ORNL,INL	$900	x	x	x	x	x	x	x	x	x	x	x	X
5.2.5-1	Yes	3	INL,ORNL	$500	x	x	x	x	x	x	x	x	x	X		
5.2.5-2	Yes	3	INL,ORNL	$250	x	x	x	x	x	x	x	x	x	x	X	
5.2.6-1	No	3	INL,ORNL	$300	x	x	x	x	x	x	x	x	x	x	x	x
5.2.6-2	No	3	ORNL	$75	x	x	x	x	x	x	x	x	x	x	x	x
5.2.7-1	No	3	INL,ORNL	$200	x	x	x	x	x	x	x	x	x	x	x	x
5.2.7-2	Yes	3	ORNL,INL	$75	x	x	x	x	x	x	x	x	x	x		
5.2.7-3	Yes	3	ORNL,INL	$200	x	x	x	x	x	x	x	x	x	x		
5.2.7-4	Yes	3	INL	$150	x	x	x	x	x	x	x	x	x	X		
5.2.8-1	No	3	INL,ORNL	$450	x	x	x	x	x	x	x	x	x	x	x	x
5.2.8-2	Yes	3	INL	$250	x	x	x	x	x	x	x	x	x	x	x	X

Reference Information From Section 5.2	Report	Level	Performing Organization	Estimated FY-07 Cost ($K)	Planned FY-07 Schedule. Report Due date is Shown in Bold											
					Oct-06	Nov-06	Dec-06	Jan-07	Feb-07	Mar-07	Apr-07	May-07	Jun-07	Jul-07	Aug-07	Sep-07
5.2.8-3	No	3	INL	$100	x	x	x	x	x	x	x					
5.2.8-4	Yes	3	INL,ORNL	$400								x	x	x	x	X
5.2.8-5	Yes	3	INL	$200	x	x	x	x	x	x	x	x	x	x	x	X
5.2.8-6	No	3	INL	$50	x	x	x	x	x	x	x	x	x	x	x	x
5.2.8-7	Yes	3	ORNL	$150	x	x	x	x	x	x	x	X				
5.2.8-8	Yes	3	ORNL	$150	x	x	x	x	x	x	x	x	x	x	X	
5.2.8-9	Yes	3	ORNL	$200	x	x	x	x	x	x	x	x	x	x	X	
5.2.8.10	Yes	2	ORNL,INL	$427	x	x	x	x	x	x	x	x	x	x	x	X
5.2.9-1	Yes	2	ORNL	$450	x	x	x	x	x	x	x	x	x	x	X	
5.2.9-2	No	2,3	INL,ORNL	$150	x	x	x	x	x	x	x	x	x	x	x	x
5.2.9-3	Yes	3	ORNL,INL	$400	x	x	x	x	x	x	x	x	x	X		
5.2.9-4	Yes	3	ORNL	$300	x	x	x	x	x	x	x	x	x	X		
5.2.9-5	Yes	3	INL	$275	x	x	x	x	x	x	x	x	x	x	X	
5.2.10-1	No	3	INL,ORNL	$100	x	x	x	x	x	x	x	x	x	x	x	x
5.2.11-1	Yes	2,3	ORNL,INL	$800	x	x	x	x	x	x	x	x	x	x	x	X
5.2.12-1	Yes	2,3	ORNL,INL	$375	x	x	x	x	x	x	x	x	x	x	X	
5.2.12-2	No	2	ORNL	$150	x	x	x	x	x	x	x	x	x	x	x	x
5.2.12-3	Yes	2	ORNL	$350	x	x	x	x	x	x	x	x	x	x	X	
5.2.13-1	No	3	INL	$500	x	x	x	x	x	x	x	x	x	x	x	x
5.2.14-1	Yes	3	U of M	$150	x	x	x	x	x	x	x	x	x	x	x	X
5.2.15-1	No	3	INL,ORNL	$825	x	x	x	x	x	x	x	x	x	x	x	x
5.2.15-2	Yes	3	INL	$200	x	x	x	x	x	x	x	x	x	x	X	
Totals				$13,792												

Appendix A
Annual Progress Report for SiC/SiC Composites for Control Rod Structures for NGNP for Period January-October, 2005

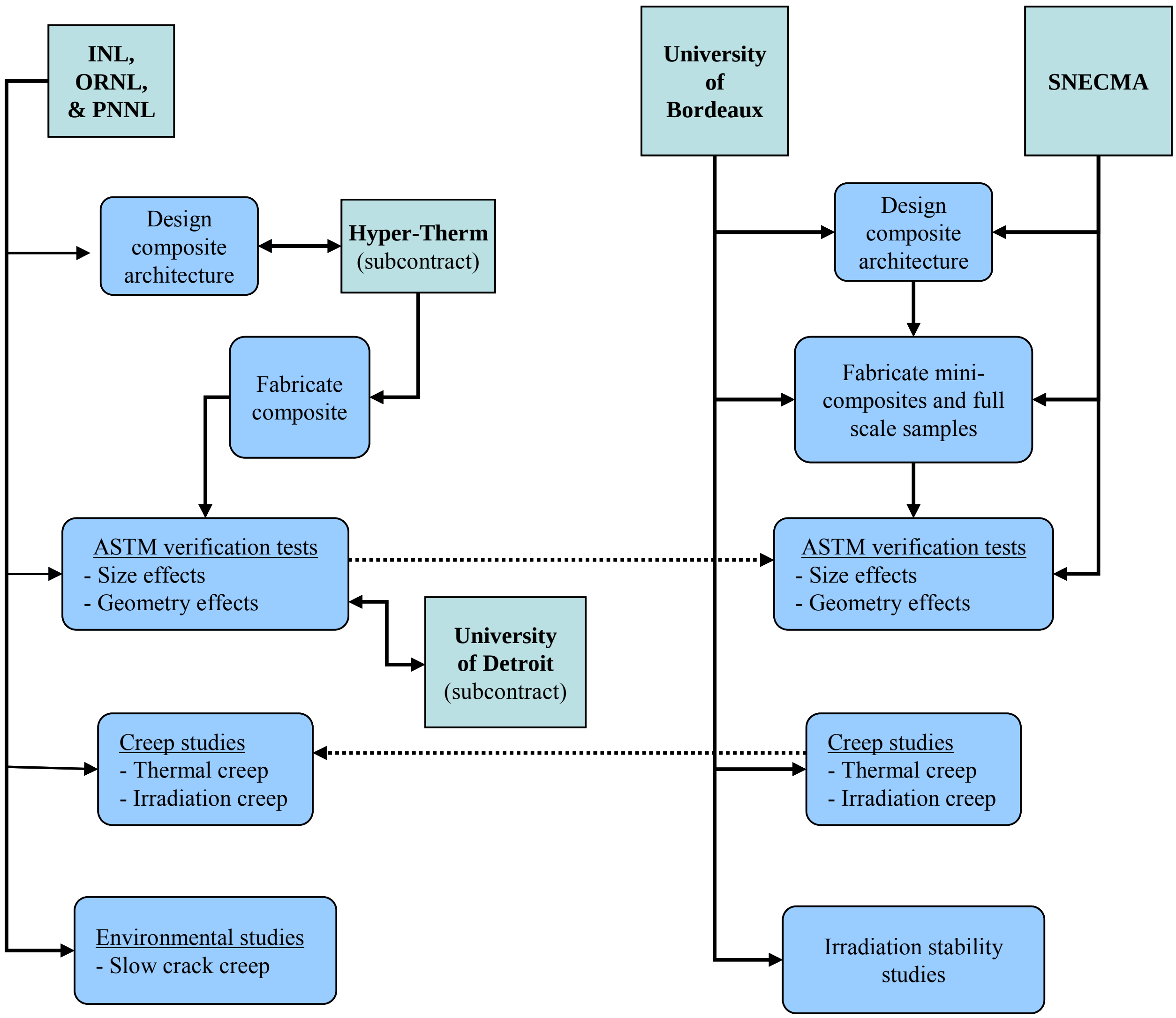
Project Number and Title:	2004-004-F : SiC/SiC for control rod structures for NGNP
Lead U.S. Investigating Organization:	INL
U.S. Principal Investigator:	W.E. Windes
Lead Collaborating Investigating Organization:	University of Bordeaux
Lead Collaborating Principal Investigator:	Prof. J. Lamon
Other collaborating institutions:	Oak Ridge National Lab, Pacific Northwest National Lab, SNECMA, and Commissariat à l'Energie Atomique (CEA)
Reporting period:	1-1-05 to 10-15-05
NTD/SIM:	Bill Corwin
Work package #:	GEN IV I0202J01

A-1. Project Status Summary

A three-year I-NERI grant between U.S. - French research institutions (INL, ORNL, PNNL, CEA, and University of Bordeaux) has been approved for research and development of SiC/SiC composites. The proposed research will investigate the issues surrounding the development of tubular geometry SiC/SiC composite material for control rod and guide tube applications in a very high temperature reactor (VHTR) design. Mechanical, thermal, and radiation-damage response of both the USA and the French fabricated composites will be studied during this time.

The project is designed to take full advantage of the innovative SiC/SiC technologies developed by our French collaborators (Prof. Jacques Lamon at the Universite de Bordeaux, Apessac, France). This research group has pioneered the use of 2D woven SiC/SiC composites and also nanoscale-multilayered pyrolytic carbon/silicon carbide interphases. The French emphasis on basic science and improved composite design will compliment the USA program which is much more focused upon application oriented testing and verification of these composites for use in the VHTR. The French will benefit from the U.S.'s full-scale composite testing and irradiation program allowing them to thermo-mechanically test their improved composite designs in a full-scale test sample. Thus, both programs compliment each other with little to no overlap of research. Initial meetings have discussed data exchange, sharing modeling experience, and test sample exchanges between the two programs. Further meetings in the coming months will provide detailed schedules and individual research plans for specific tasks.

A-2. Project Organization



A-2.1. Task 1: Irradiation stability studies (ORNL & University of Bordeaux)

Irradiation stability experiments for both candidate composites have been initiated within the High Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory (ORNL) to determine the limiting dose level for each type. Irradiation test capsules are currently being exposed to 10, 20, and 30 dpa dose levels. Samples exposed to 10 dpa irradiation doses have been completed; samples to dose levels of 20 and 30 dpa are scheduled for completion in following years. Preliminary indications show that SiC composites may be stable for the full lifetime of the reactor (up to 30 dpa) while C_f/C composites become compromised at 8 dpa.

University of Bordeaux will fabricate mini-composite samples with improved interphase and architecture designs for future irradiation stability studies. If these mini-composites prove to be more stable than the current composite structures large, full-scale composites will be fabricated for future irradiation creep studies.

A-2.2. Task 2: Composite fabrication (ORNL/PNNL/INL)

Since the control rods will be composed of tubular segments containing the high neutron cross-section material (i.e. B_4C), tubular test specimens were designed and fabricated. In addition, small, flat tensile specimens were fabricated from composite plate materials in anticipation of the need for small, flat irradiation specimens that will actually fit within the reduced volume of a material test reactor core (discussed in Task 4 below). Therefore, both tubular and flat plate tensile specimens were designed and fabricated for future mechanical testing. Actual testing of both tubular and flat, "dog-bone"-shaped tensile composite specimens will begin next year.

Specific attention was paid to the architectural fiber preform design as well as the materials used in construction of the composites. For maximum irradiation stability and moderate composite strength, both composite types used a simple ±45° bi-axial braiding architecture. Much stronger three-dimensional weaving was considered unnecessary for these moderately low stressed components. A much more important consideration was material selection, which resulted in significantly higher irradiation stability for both composite types. SiC composites used Hi-Nicalon Type-S fiber preforms with chemical vapor infiltration. These preform materials have been demonstrated to have superior irradiation stability to other SiC material systems.

A-2.3. Task 3: ASTM standards (All)

An ASTM working group has been established that will oversee the ASTM round-robin tests anticipated for creating standardized mechanical and environmental test procedures for ceramic composites. Our French collaborators at the University of Bordeaux and at SNECMA have been asked to participate in these round-robin mechanical tests. They are determining whether they wish to participate in an American standard development since they have a similar standardization organization, the European Committee for Standardization (CEN). If they decide to participate in the testing program they have agreed to attend the next American Ceramics Society meeting held at Cocoa Beach Florida where we will discuss the methods and procedures for the round-robin tests.

Currently, there is no precedent for using ceramic composites within a nuclear reactor. Consequently, no ASTM standards or ASME code cases exist for using ceramic composite components in a nuclear core. This research program will use the test procedures and methodology established during these studies to create standardized mechanical and environmental test procedures for use in validating a structural ceramic composite for use in a nuclear reactor system. ASTM (or equivalent) standards are being created for the composite architectures used, the high temperature test methods developed (both for tensile strength and creep tests), and environmental testing of SiC composites. In addition, ASTM round-robin test methods will be used to validate that these test standards are truly international standards that can be applicable to all reactor designs.

The development of international test standards will require close collaborations between the U.S. national laboratories and international collaborators. To this effect, a strong tie to France research institutions will be necessary to establish both national and international test standards in qualifying ceramic composites for nuclear reactor applications. These international collaborations will allow the researchers to share data, materials, and test samples, as well as provide a basis for working groups to create the new standards.

A-2.4. Task 4: Creep studies (INL/University of Bordeaux/SNECMA)

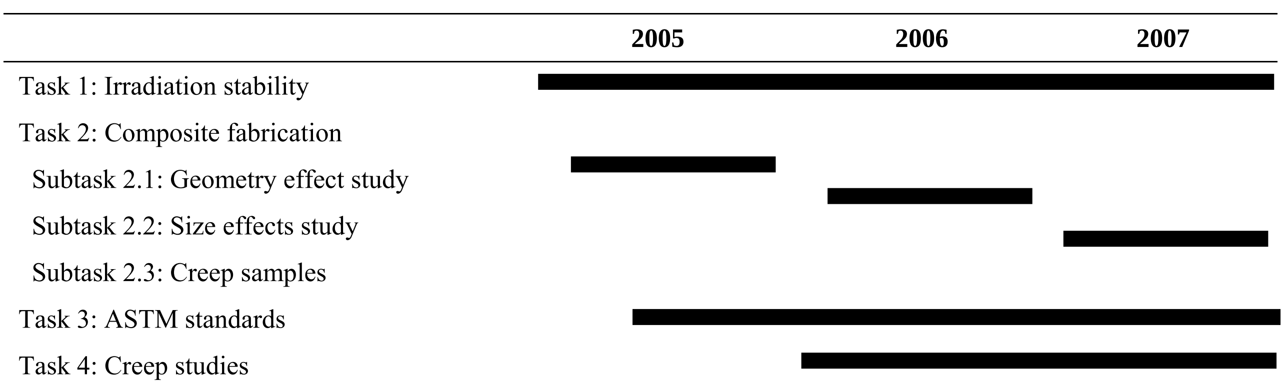
Concurrent to the irradiation stability studies, mechanical and environmental test capabilities are being established for temperatures up to 1600º C. Test equipment, high temperature testing methodologies, VHTR reactor core conditions, and composite sample design for high temperature tensile strength and long duration creep studies have been discussed and designed. Very high temperature test frames capable of attaining and sustaining temperatures around 1600 ºC have been purchased and are currently being installed. These high temperature frames will have ceramic environmental chambers capable of providing a helium environment similar to the anticipated environment inside the VHTR. Therefore, the tests will simulate the thermal and environmental conditions inside a VHTR (without radiation) providing the baseline non-irradiated creep data. Both small tubular and flat plate specimens will be tested in these high temperature tests next year.

Dependant upon the irradiation stability results of the French mini-composites discussed previously, University of Bordeaux and SNECMA will fabricate full-scale composite test samples for insertion into an irradiation creep capsule based upon the dimensions and design of the USA program. These composites will be irradiated at the same time, temperature, and dose levels as the current SiC composite architectures. PIE and characterization after irradiation will compare the “new” composites to the current USA composites.

High temperature, non-irradiated creep studies will be performed next year to determine a baseline creep response of the tubular geometry components. However, once the non-irradiated creep response has been established the enhanced creep effects resulting from neutron irradiation will need to be determined by irradiating composite test samples. Irradiation doses comparable to a lifetime exposure (~30 dpa) will be used to quantify the creep response of these composite components. The design of an irradiation creep capsule capable of simulating the stress, temperature, and final dose levels expected for a VHTR control rod cladding will be initiated next year for irradiation enhanced creep studies of these composites.

A-3. Project Milestone/Deliverable Summary

Milestone/Deliverable Description	Planned Completion Date	Actual Completion Date
Task 1: Irradiation stability	10-07	
Task 2: Sample fabrication	06-05	10-05
Task 3: ASTM standards	10-07	
Task 4: Creep studies	10-07	

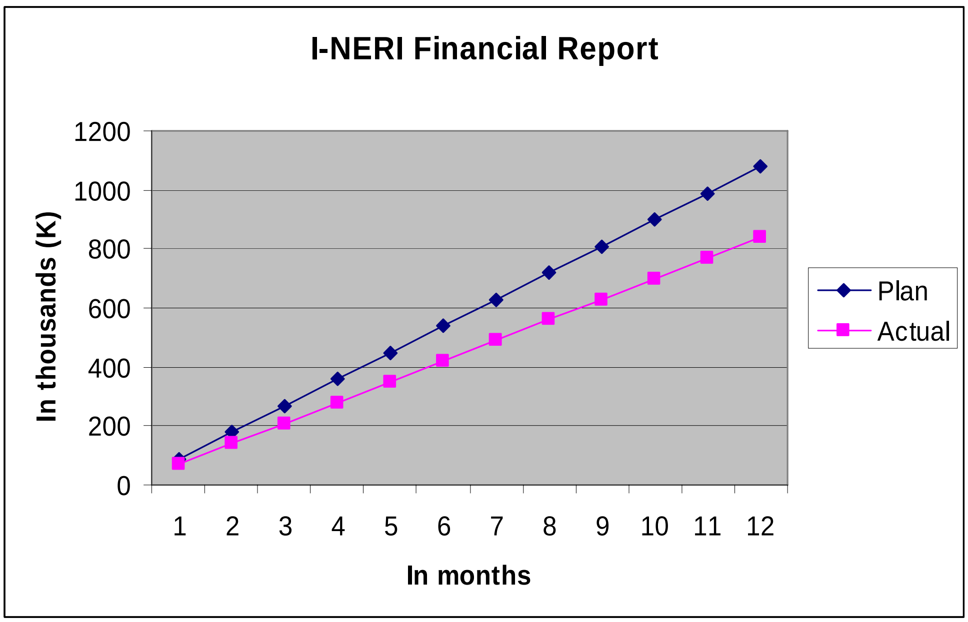


Source: 2004-004-F : SiC/SiC for control rod structures for NGNP

A-4. Financial Performance Summary

Monthly Expenditure Summary:

	FY 2004 Totals	2005 Oct	Nov	Dec	Jan	Feb	Mar	Apr	May	Jun	Jul	Aug	Sep	FY 2005 Totals
Monthly Cost														
- Plan	0	90	90	90	90	90	90	90	90	90	90	90	90	1080
- Actual	0	70	70	70	70	70	70	70	70	70	70	70	70	840
Cumulative														
- Plan	0	90	180	270	360	450	540	630	720	810	900	990	1080	1080
- Actual	0	70	140	210	280	350	420	490	560	630	700	770	840	840



Source: 2004-004-F : SiC/SiC for control rod structures for NGNP.

Under expenditures occurred due to issues with sample procurement and a delay in the high temperature load frame assembly. Fabrication of composites is much more complex since they are engineered products. The proper fiber material, weave architecture, infiltration methods, infiltration materials, and final heat treatments can produce significantly different composite structures. A great deal of time was spent to carefully design the composite architecture and sample dimensions to provide viable test results. Using either an incorrect architectural design or incorrectly sized samples would have skewed the results and rendered questionable data.

In addition, the ceramic inserts needed for the high temperature mechanical test frames are necessarily custom fabricated requiring a significantly longer time to procure. Final assembly of the high temperature test frames was delayed while these ceramic inserts are being fabricated.

Appendix B: Quarterly Progress Report for NERI Project Being Performed at the University of Michigan Related to and Supplementally Funded by the NGNP Materials Program

Project Title: Alloys for 1000°C service in the Next Generation Nuclear Plant NERI 05-0191

Covering Period: April 15, 2006 through June 30, 2006 Quarter 1 Year 2

Date of Report: July 27, 2006

Recipient: University of Michigan 2355 Bonisteel Blvd Ann Arbor, MI 48109-2104

Award Number: DE-FC07-05ID14660

Principal Investigator: Gary S. Was, 734 763-4675, gsw@umich.edu

J. W. Jones, 734 764-7503, jonesjwa@umich.edu

T. Pollock, 734 615-5150, tresap@umich.edu

Project Objective: The objective of the proposed research is to define strategies for the improvement of alloys for structural components, such as the intermediate heat exchanger and primary-to-secondary piping, for service at 1000°C in the He environment of the NGNP. Specifically, we will investigate the oxidation/carburization behavior and microstructure stability and how these processes affect creep. While generating this data, the project will also develop a fundamental understanding of how impurities in the He environment affect these degradation processes and how this understanding can be used to develop more useful life prediction methodologies.

Status:

B-1. Effect of CO/CO2 ratio on Oxidation Behavior

B-1.1 Introduction

Helium gas used as coolant in very high temperature gas-cooled reactor (VHTR) contains CO, CO2, CH4, H2 and H2O as the main impurities. Depending on the temperature, alloy composition and helium gas chemistry oxidation/carburization/decarburization of the metallic alloy can occur. Simplified helium gas chemistry containing ppm levels of CO and CO2 was selected for a 500 hr corrosion test with alloy 617 at 1000°C. In the (He-CO-CO2) environment, the oxygen partial pressure and hence oxidation potential in the gas mixture is established by the reaction:

$$2\ CO_2 \rightarrow 2\ CO + O_2$$ [160]

whereas, the carbon potential is established by:

$$2\ CO \rightarrow CO2 + C$$ [161]

Corrosion behavior of alloy 617 in this environment can be understood by comparing the oxidation and carburization potential of the impure helium gas to that of the thermodynamic stability diagram of alloy 617. The purpose of the initial experiment on alloy 617 was to determine the effect of CO/CO2 ratio on the oxidation rate at 1000°C.

B-1.2 Experimental procedure

Exposure tests were conducted by flowing He gas with a known impurity gas concentration through a 3-zone tube furnace at a temperature of 1000°C. The furnace houses 7 quartz tubes, each containing three test samples of alloy 617 of dimension (6 mm x 5 mm x 1 mm). Three of the seven quartz tubes were used in the experiment and each contained different concentrations of CO and CO2 that were established by metering appropriate amounts of gases from a premixed gas bottle of He containing CO and CO2. Table B-1 shows the concentration levels of CO and CO2 established in the three quartz tubes, designated as T2, T3 and T4. Gas mixtures were analyzed both before and after flow through the furnace using a discharge ionization detector (DIDGC) made by PerkinElmer, Inc. Real time data acquisition was done by using computer and data-acquisition hardware from National Instruments, Inc.

Table B-1. Conditions of the test in the three tubes.

Tube #	Exposure time (hr)	CO (ppm)	CO2 (ppm)	CO/CO2 ratio	Log10(PO2)	Log10(ac)
T2	150-225-500	1090	1.0	1090	-20.178	-2.034
T3	150-225-500	540	0.90	600	-19.660	-2.598
T4	150-225-500	197	0.75	263	-18.901	-3.414

Before the exposure experiment at 1000°C was started, ultra high pure helium (99.9999% purity) was flowed through the sealed quartz tube in order to flush out any air present in the system. After about 6 hours of purging, the oxygen level was reduced to 1 ppm. The next step was to bake out the quartz tubes for removal of adsorbed air and moisture on its surface. For this purpose, the furnace was heated up to the exposure temperature of 1000°C while ultra high pure helium was passed through the tubes. After about 7 hrs of baking and after the oxygen level inside the quartz tube dropped below the detection limit, the desired CO and CO2 concentrations were established inside each of the three quartz tubes. The three test samples, which were initially placed in the cold zone of the quartz tube, were pushed into the hot zone of the furnace.

The first two corrosion coupons were moved out of the hot zone to the cold zone at 150 and 225 hrs of exposure and cooled until the third coupon was taken out at 500 hrs of exposure. Finally, all the three coupons were removed from the cold zone after 24 hours of cooling and weighed on the microbalance with a resolution of 0.01 mg. The microstructures of the exposed samples were analyzed using a SEM equipped with EDX.

B-1.3 Results and Discussion

Figures B-1, B-2, and B-3 are the plots of CO concentration at the inlet and outlet of tube 2 (T2), tube 3 (T3) and tube 4 (T4) as function of time, respectively. Two observations can be made from these plots: (1) the CO concentration at the outlet of a tube is greater than that at the inlet, indicating the production of CO, and (2) the difference between CO at the inlet and outlet of the tubes diminishes gradually after an exposure duration of about 300 hr. This behavior is much more pronounced in tube 4. Toward the end of the 500 hr exposure test, the CO concentrations at the outlet and inlet are close to each other, indicating that during the initial stages of exposure, a protective film of oxide has formed which resulted in reduced corrosion kinetics. A compact and protective oxide film on the surface of the alloy should reduce the oxidation rate and in the ideal case, parabolic oxidation kinetics is observed.

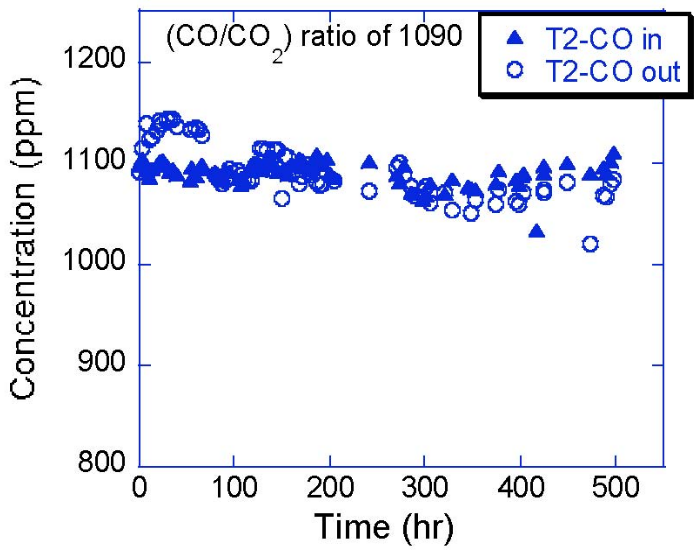


Figure B-1. Change in CO concentration in the tube 2 as a function of exposure duration. (CO/CO2) ratio of 1090

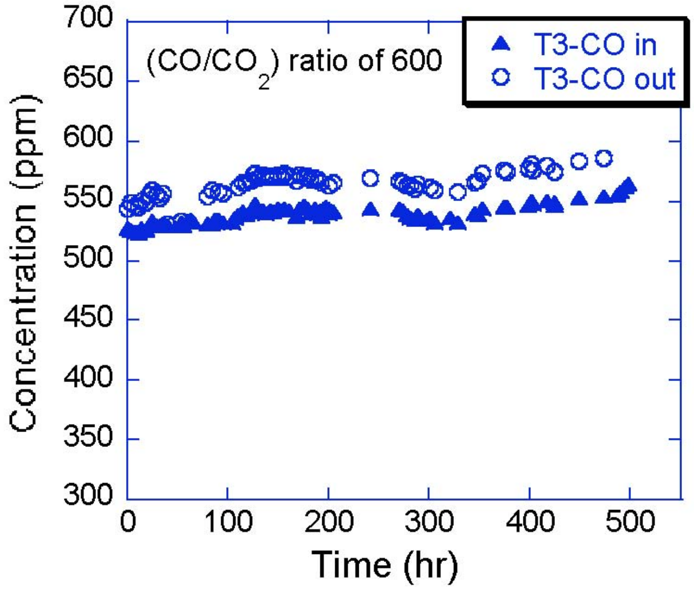


Figure B-2. Change in CO concentration in the tube 3 as a function of exposure duration. (CO/CO2) ratio of 600

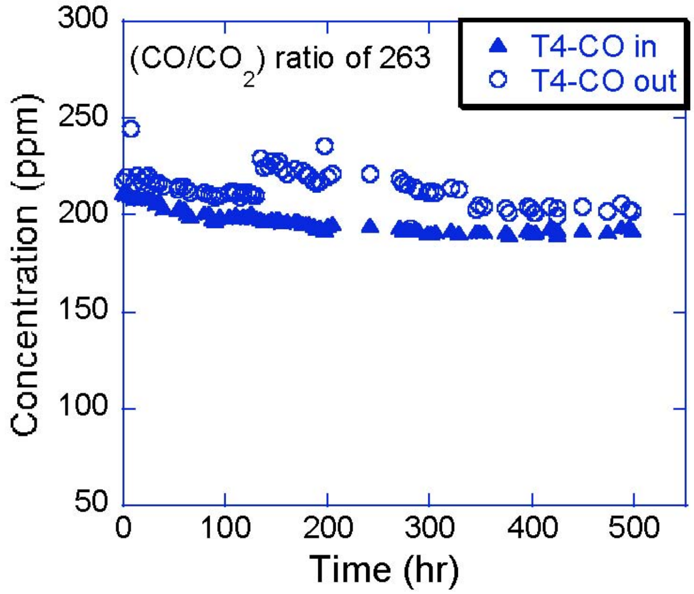


Figure B-3. Change in CO concentration in the tube 4 as a function of exposure duration. (CO/CO2) ratio of 263

Figure B-4 is a plot of the square of weight change of the samples vs. time. A linear behavior on this plot indicates parabolic kinetics and the slope gives the parabolic rate constant. As is evident from the plot, ideal parabolic behavior was not observed in any of the gas compositions for a 500 hr exposure duration test. Oxidation kinetics increases with increasing CO/CO2 ratio, as indicated in Table B-2.

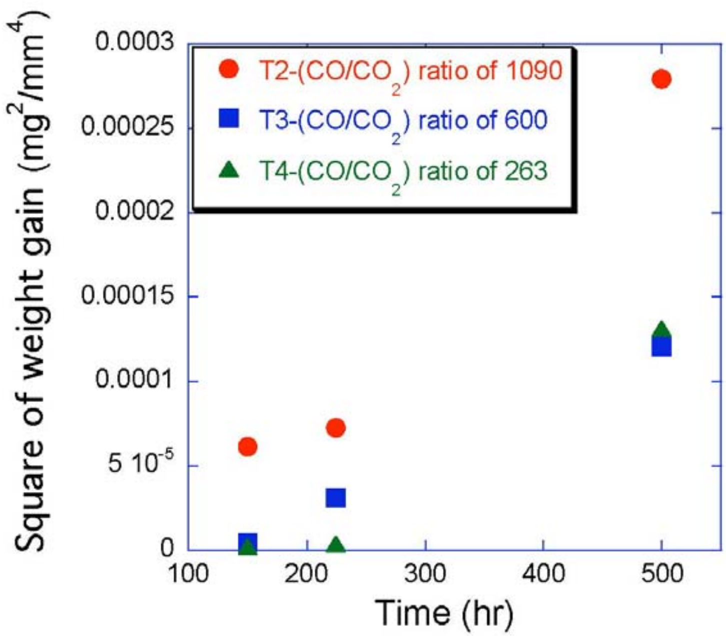


Figure B-4. Plot of square of weight change vs. time

Table B-2. Weight change of samples vs. time

Tube #	Exposure time (hr)	Initial weight (mg)	Final weight (mg)	Weight change (mg)	Total surface area (A) (mm2)	Weight change per unit area x10-3 (mg/mm2)	Measured oxide film thickness (µm)
	150	191.40	192.04	0.64	81.8	7.81	
T2	225	196.52	197.22	0.70	81.7	8.54	
	500	197.02	198.39	1.37	81.8	16.71	4.30
	150	192.40	192.57	0.17	82.5	2.07	
T3	225	195.33	195.79	0.46	81.4	5.61	
	500	197.86	198.76	0.90	82.0	10.98	3.36
	150	194.24	194.36	0.12	81.3	1.46	
T4	225	187.51	187.66	0.15	82.3	1.83	
	500	195.46	196.40	0.94	81.5	11.46	3.20

Figure B-5 is an x-ray map of the sample exposed to a CO/CO2 ratio of 600 for 500 h. As shown in the SEM image and its x-ray map, a surface oxide scale rich in chromium and oxygen has formed. Oxygen has also diffused further inward causing internal oxidation of Al to Al2O3.

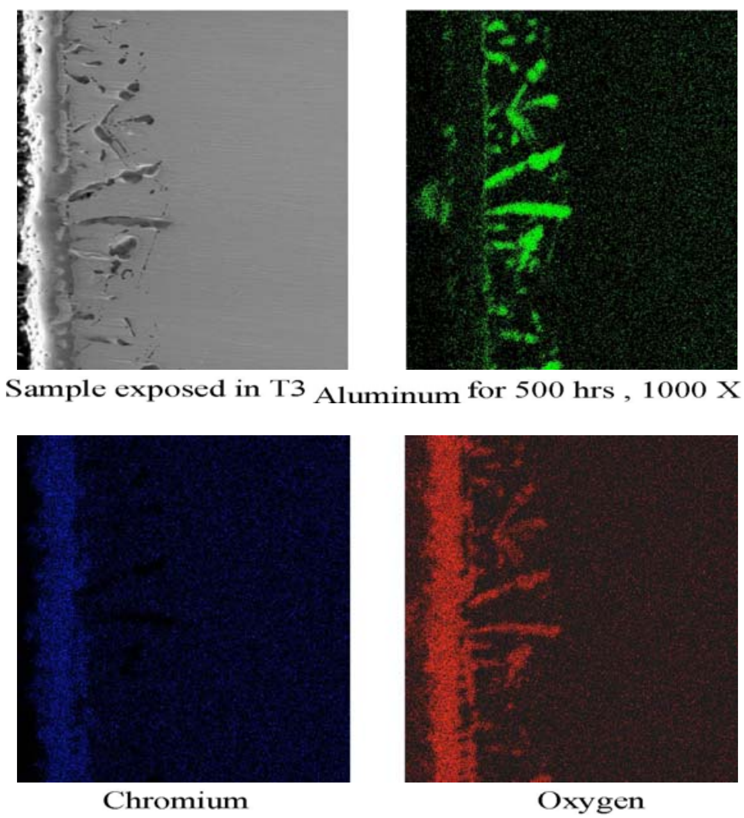


Figure B-5. X-ray map of sample exposed to a CO/CO2 ratio of 600 for 500 h at 1000°C.

Figures B-6, B-7, and B-8 are the SEM images of the test samples exposed for 500 h at 1000°C in CO/CO2 ratios of 1090, 600 and 263, respectively. A continuous chromium oxide layer is formed on all three samples. Finger-like internal precipitates of Al2O3 also form as the internal oxidation product of Al. Weight gain in a sample can occur in the form of external oxide scales as well as internal oxidation of Al to Al2O3. The thickness of the external oxide scale was measured to be 4.30 μm, 3.36 μm and 3.20 μm for the samples exposed to (CO/CO2) ratio of 1090, 600 and 263, respectively. The difference in the thickness of the oxide scales of the sample exposed to (CO/CO2) ratio of 1090 and 600 is 0.94 μm, which corresponds to a weight difference of 4.90 x 10^{-4} (gm/cm^2) of the sample, assuming that the oxide scale formed on the sample is pure Cr2O3 with a density of 5.21 (gm/cm^3). This weight gain is comparable to actual weight gain difference of 5.73 x 10^{-4} (gm/cm^2) between the samples exposed to (CO/CO2) ratio of 1090 and 600. Similarly, the difference in the thickness of oxide scale between the samples exposed to (CO/CO2) ratio of 1090 and 263 is 1.1 μm resulting into a calculated weight gain difference of 5.731 x10^{-4} (gm/cm^2). This calculated difference in weight gain between these two samples is close to the actual weight gain difference of 5.244 x10^{-4} (gm/cm^2) measured using the microbalance. The analysis shows that the additional weight gain by the sample exposed to a (CO/CO2) ratio of 1090 is mainly due to the formation of a thicker chromium oxide film.

Figure B-6. SEM Image of sample exposed to 500 hrs in tube 2, T2 (CO/CO_2 of 1090) at 1000C.

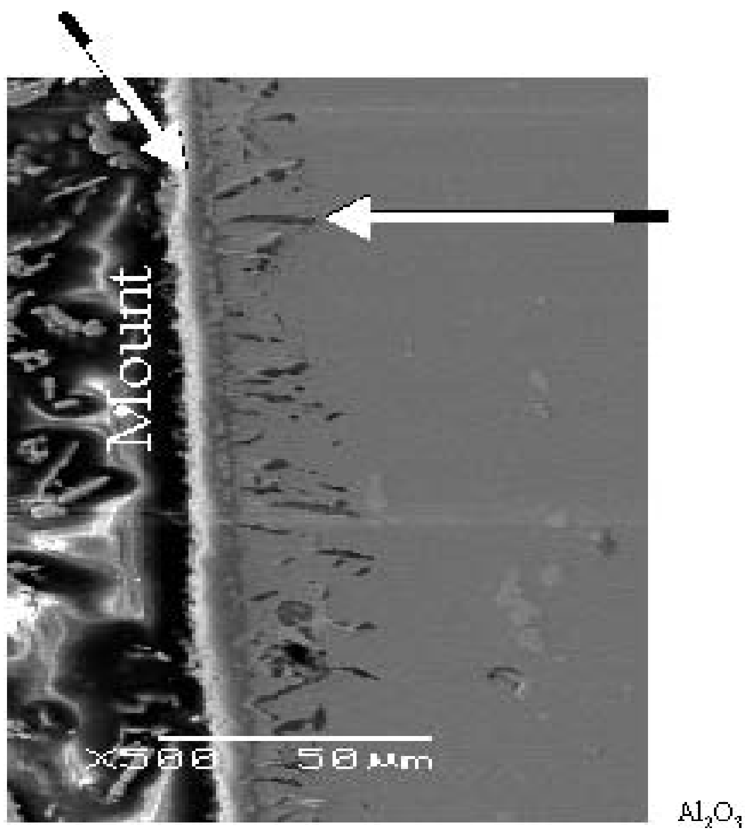


Figure B-7. SEM Image of sample exposed to 500 hrs in tube 3, T3 (CO/CO_2 of 600) at 1000C.

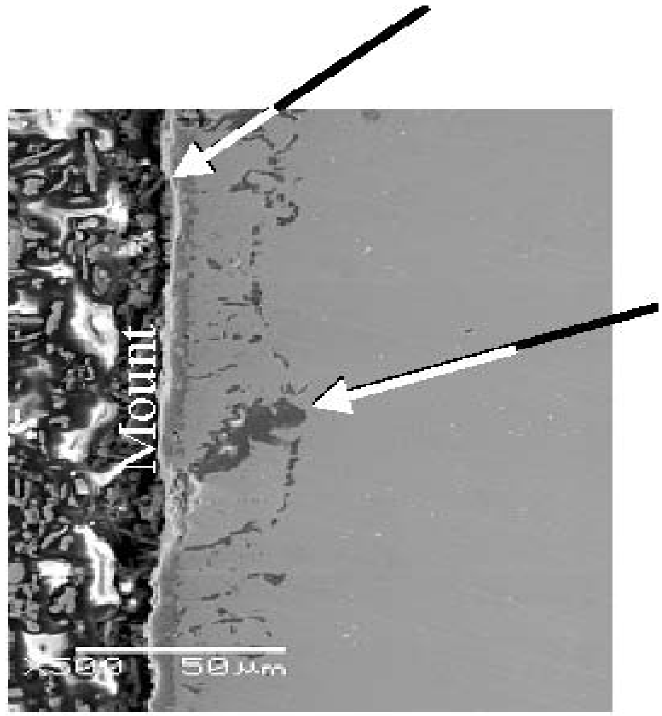
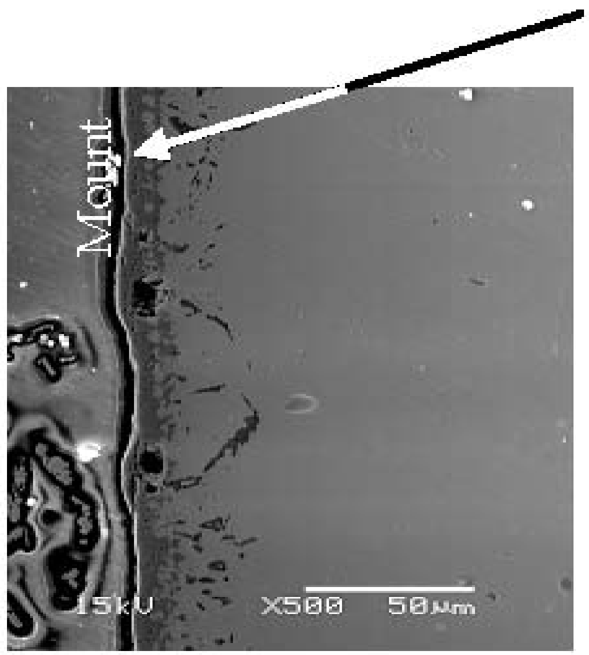


Figure B-8. SEM Image of sample exposed to 500 hrs in tube 4, T4 (CO/CO_2 of 263) at 1000C.

As observed by Bates [1], a particular feature of the surface oxide scale in Inconel 617 is the presence of matrix trapped within the oxide layer (Figure B-9). The maximum depth of internal oxidation is found to be 30 µm, 27 µm and 29 µm for the samples exposed to CO/CO2 ratios of 1090, 600 and 263 for 500 h, respectively, suggesting that the depth of internal oxidation is independent of the oxidation/carburization potential of the He gas. Internal oxidation of Al to Al2O3 in alloy 617 can be envisioned as occurring in three steps: (1) dissociation of O2 into atomic oxygen, (2) adsorption or dissolution of oxygen atoms on the surface of the sample, and (3) inward diffusion of oxygen atoms and reaction with Al to form Al2O3. For the Ni-Al and Ni-Cr system, the rate of internal oxidation was governed by the inward diffusion of dissolved oxygen, with negligible counter diffusion of aluminum or chromium[161]. The diffusion rate of oxygen depends on the diffusivity of oxygen in alloy 617, which is a function of the temperature and grain size of the given sample and should be independent of the oxidation potential of the environment. The observed independence of the depth of internal oxidation on the oxidation potential is consistent with this model.



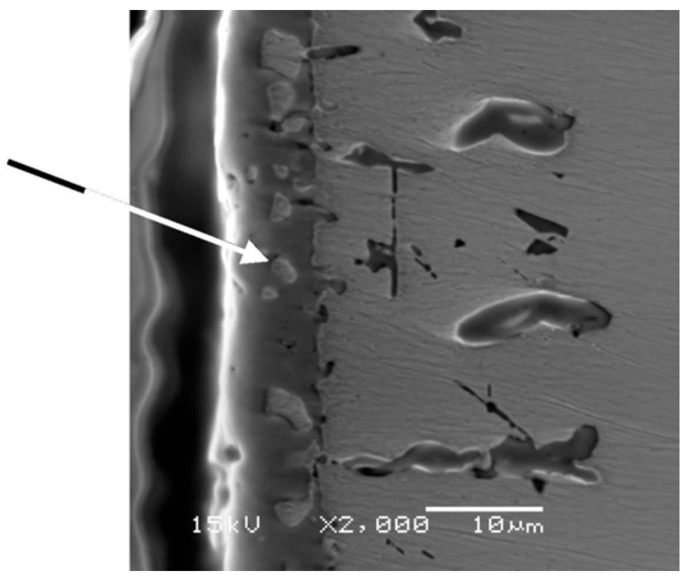


Figure B-9. SEM image of sample exposed in tube 4 (CO/CO_2 ratio of 263) for 500 hrs. Matrix particles trapped within the oxide layer are visible in the image.

B-1.4 Trip to Idaho National Lab

Deepak Kumar made a one-day trip to Idaho National Lab on July 11, 2006. The purpose of this visit was to exchange the ideas and experiences in assembling and performing exposure experiments in controlled-impurity He systems. In particular, the issues regarding controlling and continuously analyzing the moisture level during a test was discussed. We will like to thank Dr. Richard Wright for making this trip successful.

B-1.5 Summary

A 500 hr exposure test was conducted on alloy 617 samples in three different ratios of (CO/CO2) at 1000°C ranging between 263 and 1090. All the samples exposed showed a continuous surface layer of chromium oxide film with internal oxidation of Al to Al2O3. The difference in weight gain between samples is accounted for by the thickness of the chromium oxide external layer. The maximum depth of internal oxidation was found to be approximately the same for all three CO/CO2 ratios of 1090, 600 and 263 for 500 h. This independence of depth of internal oxidation on the oxidation potential of the environment is possible if the kinetics of internal oxidation is a diffusion-controlled.

B-2 Alloy Modification for Improved High Temperature Performance

One of the objectives of this program is to define new alloying approaches for development of materials for service at maximum temperatures of 1000˚C. Since long-term stability of microstructure is essential, nickel alloys that are nominally single phase are of interest. Refractory alloying elements, including Re, W, Ru and Ta are being investigated. Additions of Ta are of interest because they will influence the primary carbide composition and morphology as well as subsequent carbide reactions during elevated temperature exposures. Re, W and Ru will improve creep properties through their influence on interdiffusion.

B-2.1 Alloy Screening Studies

A major consideration in the screening studies is that any new alloys must ultimately be subject to fabrication via commercial hot working routes. Accordingly we have collaborated with Special Metals on the design of new alloys. The constraints of processing alloys into sheet form have been discussed in detail. It was established that there have been no prior investigations at Special Metals / Inco in the alloy composition domain of interest for this program. Thermocalc calculations were conducted for an initial set of alloys (J1 – J5, Table B-3). These alloys were melted homogenized and subjected to creep experiments in compression at 1000˚C and 20 MPa. Wrought 617 Alloy provided by Special Metals was also subjected to creep under the same conditions to establish a reference.

Table B-3. Nominal compositions of alloys for initial screening studies, (wt%).

Alloy	Ni	Cr	Co	Mo	Al	Fe	Mn	Si	Ti	C	B	W	Re
IN 617	45.78	24	15	8	1	3	1	1	0.6	0.1	0.006		
J1	84				6.5							9.5	
J2	72.95	12			3					0.05		12	
J3	82.95	6			3					0.05		8	
J4	76.95	6			3					0.05		8	6
J5	79.95	6			3					0.05		8	3
J6	55.05	15	12	2.7	2.3					0.05		5.3	7.6
J7	59.65	12	12		2.3					0.05		8	6
J8	55.65	12	12	3	2.3					0.05		12	3
J9	56.65	12	12	3	2.3					0.05		8	6
J10	60.65	6	12	5	2.3					0.05		8	6
J11	66.65	6	6	3	2.3					0.05		8	8

Figure B-10 shows the results of creep experiments. Note that the J1 alloy is stronger than 617, but that this is due to the presence of γ' precipitates at the test temperature. While these precipitates provide substantial strengthening, they would be expected to be morphologically unstable for long periods of exposure. We therefore have focused to a greater degree on alloys that do not contain these precipitates in the temperature range of 900˚C to 1000˚C; alloys J2 – J5 are single phase at temperature. Alloys J2 and J4 have creep properties somewhat inferior to 617, but it is important to note that the microstructure and higher order alloying additions are not optimized. Higher creep resistance in the experimental alloys is associated with high levels of W and Re, with a particularly strong effect of W. Based on these results (and results reported in previous quarterly and annual reports), a second set of 6 alloys (J6 – J11) are currently being melted and will be subjected to similar evaluations. The new set of alloys contain high levels of W and Re as well as Mo and Co additions. Alloy J6 has a composition that is equivalent to the matrix γ phase composition in the advanced single crystal superalloy René N5. Following the analysis of the new set of alloys, we will select compositions for scale up heats of 50lbs, to be fabricated at Special Metals.

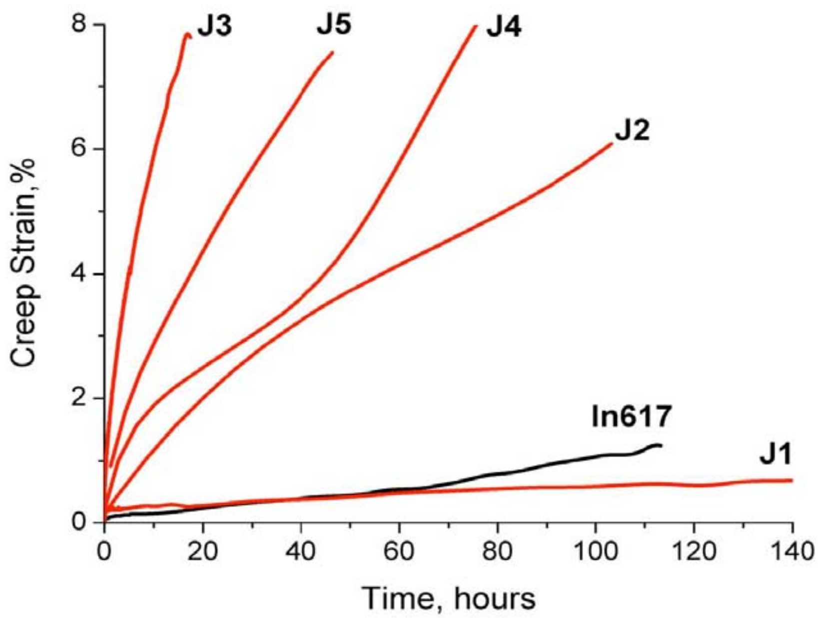


Figure B-10. Results of compression creep tests at 10000C and 20 MPa.

B-3 Plans for Next Quarter

Plans for next quarter include:

- Determination of minimum flow rate required to avoid starvation of the oxidation reaction.
- Exposure test of alloy 617 at two different temperatures 850 and 1000°C in (He-CO-CO2) environment with 6 different ratios of (CO/CO2)

Patents: None

Publications: None

Table B-4. Milestone Status for Year 2

Milestone/Task Description	Planned Completion Date	Actual Completion Date	Percent Complete
First test on effect of CO/CO2 ratio on oxidation	June 2006	June 2006	100%
Oxidant starvation test	August 2006		20%
1000°C exposure test at six CO/CO2 ratios	November 2006		0%
Determine processing conditions for optimum CSL-enhancement	December 2006		30%
850°C exposure test at six CO/CO2 ratios	February 2006		0%
Creep experiments on 617 – 3 environments at 950°C	February 2007		10%
Fabricate composition-modified alloys for screening studies	November 2006		60%
Fabricate 3-4 50lb heats of most promising composition-modified alloys	February 2007		0%

Table B-5. Budget Data (as of June 30, 2006)

Phase / Budget Period			Approved Spending Plan			Actual Spent to Date		
Year	From	To	DOE Amount	Cost Share	Total	DOE Amount	Cost Share	Total
Year 1	April 2005	April 2006	250,000	50,000	300,000	206,000	40,000	246,000
Year 2	April 2006	April 2007	250,000	0	250,000	46,000	0	46,000
Year 3	April 2007	April 2008						
Totals								

Appendix C:
Pebble Bed Modular Reactor (PBMR) Concept Description

Appendix C is a proprietary version of Section 3.3, *PBMR VHTR Concept Description*, and is not approved for public release. Contact Rafael Soto (rafael.soto@inl.gov) to inquire about obtaining a copy.

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