INL/EXT-05-00758
Rev. 2

Next Generation Nuclear Plant Materials Research and Development Program Plan

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The INL is a U.S. Department of Energy National Laboratory
operated by Battelle Energy Alliance

Idaho National Laboratory

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EXECUTIVE SUMMARY

The U.S Department of Energy (DOE) has selected the Very High Temperature Reactor (VHTR) design for the Next Generation Nuclear Plant (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 about 10% of the heat) 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 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 is being initiated prior to the design effort to ensure that materials R&D activities are initiated early enough to support the design process and support the Project Integrator. 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.

- Irradiation testing and qualification of the reactor pressure vessel (RPV) materials (including post-irradiation examination of specimens). This data is required for NRC licensing and ASME Code Case development.

- Qualification and testing of the silicon carbide fiber/silicon carbide matrix composite materials needed for the NGNP. This effort is required because composites will need to be used for control rod cladding and guide tubes in the high-temperature environments of a 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 programs.

- Support of a program to address materials issues associated with the NGNP power conversion unit. Due to the various potential designs proposed, various materials issues need to be addressed.

- Support of a program to address the emissivity and other physical and mechanical properties of layers that either form by high-temperature environmental exposure or artificial engineered layers on the exterior surface of the NGNP RPV. Data is necessary for NRC licensing and design for off-normal conditions.

- Support of a program to study fabrication and transportation issues related to the NGNP RPV.

- Support of a program to study, design, test, and qualify NGNP internals, insulation, valves, bearings, seals, and other components. When the design is further defined, this work will be documented and focused in more detail.

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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

The U.S. Department of Energy (DOE) has selected the Very High Temperature Reactor (VHTR) design for the Next Generation Nuclear Plant (NGNP) Project. The NGNP reference concept is a helium-cooled, graphite-moderated, thermal neutron spectrum reactor with an outlet temperature in the range of 850 to 1000 °C and a 60-year operating lifetime. The reactor core is currently envisioned to be a prismatic graphite block type core. However, it is feasible to also consider a pebble-bed type of gas-cooled reactor. The final selection of a reference design will be made in the future. The plant size, reactor thermal power, and core configuration will be designed to ensure passive decay heat removal without fuel damage or radioactive material releases during accidents. The initial fuel cycle will be a once-through use of very high burn-up, low-enriched uranium.

The basic technology for the NGNP has been 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 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. The time consuming development of other materials technologies will be required to achieve practical component lifetimes for NGNP deployment if the reference design is maintained.

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.

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 Assumptions

The following assumptions are incorporated into this program plan and are used in estimating the scope, cost, and schedule for completing the materials R&D processes:

1. The materials R&D process will be directed and governed by the Energy Policy Act of 2005. The scope of this work will be adjusted to reflect the level of congressional appropriations.

2. The reactor design has not been formally selected. For the purposes of this document, the design is assumed to be a helium-cooled, prismatic, graphite block core design fueled with tri-isotopic (TRISO)-design fuel particles in carbon-based compacts or a pebble-bed reactor design.

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. In any case, 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) with a hydrogen demonstration unit sized to use at least ten percent of the plant output process heat and/or electricity.

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 with one or more DOE-sponsored commercial organizations.

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[a]. Complete bibliographic references appear in numerical order in Section 7. Throughout this document, reference notations appear in the normal numerical format.

1.2 Objectives

1.2.1 Generation IV NGNP Program

The objectives of the NGNP include:

1. Demonstrate a full-scale prototype VHTR by about 2021

2. Demonstrate high-temperature Brayton Cycle electric power production at full scale with a focus on economic performance

3. Demonstrate nuclear-assisted production of hydrogen (with about 10% of the heat) with a focus on economic performance

4. Demonstrate by test the exceptional safety capabilities of the advanced gas cooled reactors

5. Obtain an NRC License to construct and operate the NGNP and to provide a basis for future performance-based, risk-informed licensing

6. Support the development, testing, and prototyping of hydrogen infrastructures

The DOE has designated that the lead laboratory in the United States for nuclear energy technology development will be the INL.

1.2.2 NGNP Materials R&D Program

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 materials R&D program is being initiated prior to the design effort to ensure that materials R&D activities are initiated early enough to support the design and licensing process. The thermal, environmental, and service life conditions of the NGNP will make material selection and qualification a significant challenge for certain very high-temperature applications. 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. Irradiation testing and qualification of the high alloy reactor RPV materials (including post-irradiation examination [PIE] of specimens). This data is required for NRC licensing and ASME Code Case development. This work will include the establishment of a low flux irradiation facility to be used to support NGNP materials irradiations.

6. Qualification and testing of the silicon carbide fiber/silicon carbide matrix composite materials needed for the NGNP.

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

8. Support of a program to address materials issues associated with the NGNP power conversion unit (PCU).

9. Support of a program to address the emissivity and other physical and mechanical properties of layers that either form by high-temperature environmental exposure or are artificial engineered layers on the exterior surface of the NGNP RPV.

10. Support of a program to study fabrication and transportation issues related to the NGNP RPV. Materials issues associated with joining and inspecting heavy section forgings are covered in this task.

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

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

Planning guidance, particularly for workscope to be performed in government fiscal year 2005, was obtained from the following sources:

1. Budget guidance from the DOE

2. Discussions and interactions with the Generation IV International Forum (GIF) provisional VHTR Materials and Components Project Management Board (PMB)

3. Subtitle C−NGNP Project, Section 461 of the Energy Policy Act of 2005

4. Recommendations provided by the INL Materials Review Committee (MRC)

1.2.3 Summary of Subtitle C, Section 461 of the Energy Policy Act of 2005

The Energy Policy Act of 2005, Subtitle C─NGNP Project authorizes the DOE to establish the Next Generation Nuclear Power Plant Project. Based on this law, the Project shall consist of the research, development, design, construction, and operation of a prototype plant. The reactor plant will generate electricity and/or produce hydrogen.

The Project shall be managed by the Office of Nuclear Energy, Science, and Technology in the DOE. The Idaho National Laboratory (INL) is the lead National Laboratory for the Project and shall collaborate with other National Laboratories, institutions of higher education, other research institutes, industrial researchers, and international researchers to carry out the Project. The INL shall organize a consortium of appropriate industrial partners that will carry out cost-shared research, development, design, and construction activities, and operate research facilities, on behalf of the Project. Preference in determining the final structure of the consortium or any partnerships under this Project shall be one that retains United States technology leadership in the Project while maximizing cost sharing opportunities and minimizing federal funding responsibilities. The prototype nuclear reactor and associated plant shall be sited at the INL in Idaho. The Project shall use, if appropriate, reactor test capabilities at the INL. The Project may use, if appropriate, facilities at other National Laboratories.

The Project shall consist of the following major program elements:

1. High-temperature hydrogen production technology development and validation

2. Energy conversion technology development and validation

3. Nuclear fuel development, characterization, and qualification

4. Materials selection, development, testing, and qualification

5. Reactor and balance-of-plant design, engineering, safety analysis, and qualification.

The Project shall be conducted in two phases – an R&D phase and a construction phase. The R&D phase shall:

- Select and validate the appropriate technology for high-temperature hydrogen production technology development and validation

- Carry out enabling research, development, and demonstration activities on technologies and components for items 2 through 4 of the major program elements

- Determine whether it is appropriate to combine electricity generation and hydrogen production in a single prototype nuclear reactor and plant

- Carry out initial design activities for a prototype nuclear reactor and plan, including development of design methods and safety analytical methods, and studies for the reactor and balance-of-plant design.

The construction phase shall:

- Continue appropriate activities of the major program elements

- Develop, through a competitive process, a final design for the prototype nuclear reactor and plant

- Apply for licenses to construct and operate the prototype nuclear reactor from the Nuclear Regulatory Commission

- Construct and start up operations of the prototype nuclear and its associated hydrogen or electricity production facilities.

The Secretary of Energy shall seek international collaboration, participation, and financial contributions for the Project. The Secretary, through the INL, may contract for assistance from specialists or facilities from member countries of the GIF, the Russian Federation, or other international partners if the specialists or facilities provide access to cost-effective and relevant skills or test capabilities. The Project may involve demonstration of selected project objectives in a partner country. The Secretary shall ensure, through the INL, that the international activities of the Project are coordinated with the Generation IV International Form.

The Nuclear Regulatory Commission shall have licensing and regulatory authority for any reactor authorized under this subtitle. The Secretary shall seek the active participation of the Nuclear Regulatory Commission throughout the duration of the Project to:

1. Avoid design decisions that will compromise adequate safety margins in the design of the reactor or impair the accessibility of nuclear safety-related components of the prototype reactor for inspection and maintenance

2. Develop tools to facilitate inspection and maintenance needed for safety purposes

3. Develop risk-based criteria for any future commercial development of similar reactor architectures.

The first phase of the Project shall select the technology to be used for high-temperature hydrogen production and initial design parameters for the prototype nuclear plant no later than September 30, 2011. The Secretary, acting through the INL, shall fund up to 4 teams for not more than 2 years to develop detailed proposals for competitive evaluation and selection of single proposal for final design of the prototype nuclear reactor. The Secretary may structure the Project activities in the second phase to use the lead industrial partner of the competitively selected design in a systems integration role for the final design and construction of the Project. Target date to complete the Project is no later than September 30 2021. By this date construction and operation of the prototype nuclear reactor and associated energy or hydrogen facility shall be achieved.

1.3 Scope

The NGNP Materials R&D Program is responsible for performing R&D on NGNP materials in support of NGNP licensing and design activities based on the Energy Policy Act of 2005. The NGNP Materials R&D Program includes the following elements:

1. Developing a specific approach, program plan, and other project management tools for managing the R&D program elements

2. Developing a specific work package and program plan for the R&D activities to be performed for each government fiscal year

3. Reporting status and progress of the work based on committed deliverables and milestone to DOE

The materials R&D program will address the materials needs for the NGNP reactor concept selected, PCU, IHX system, and associated balance of plant. Materials for hydrogen production will be addressed by the DOE Office of Nuclear Energy's (NE) Nuclear Hydrogen Initiative and, hence, are not included in the NGNP Materials R&D Program. The Materials R&D Program Plan will be updated annually to reflect changes in the design requirements basis based on guidance from DOE.

This report is Revision 2 to the *NGNP Materials Selection and Qualification Program Plan*, November 7, 2003, INEEL/EXT-03-01128, Revision 0.

1.4 NGNP Reactor Materials Organization

1.4.1 Overall Organizational Structure

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) and the NGNP Program System Integration Manager (SIM) 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.

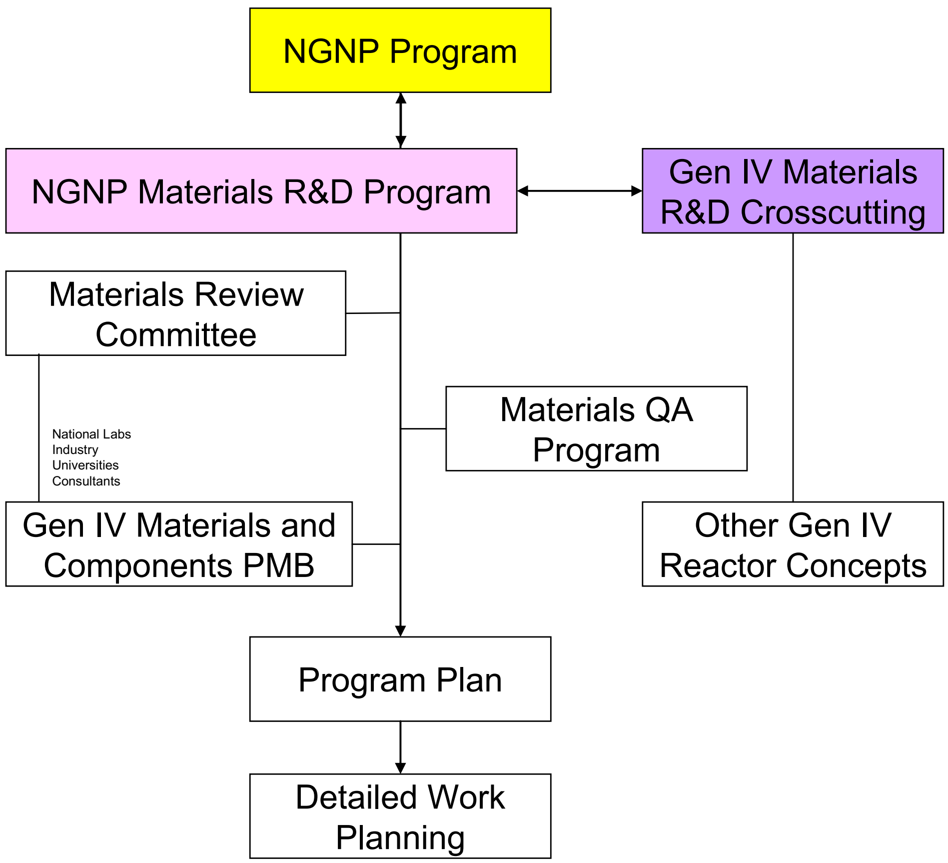

Figure 1. NGNP Materials Organization Structure.

1.4.2 NGNP Reactor Materials Review Committee

An NGNP MRC has been formed as a senior independent review body for the materials R&D program. The MRC is chaired by Russell Jones from Exponent, Inc. The MRC provides objective technical review of key selected materials documentation, including materials selection decisions, test program content, test results, etc. Information inquiries concerning MRC activities and meeting minutes can be obtained from george.hayner@inl.gov.

1.4.3 Generation IV Materials Crosscutting Interface

The Generation IV Reactors Materials Program within the Generation IV Initiative has responsibility for establishing and executing an integrated plan that addresses crosscutting, reactor-specific research needs in a coordinated and prioritized manner. The Generation IV Reactors Materials Cross-cutting and the NGNP Materials R&D Program are both part of the integrated Generation IV Materials Program. The NGNP Program is currently the highest priority reactor concept within the Generation IV Program. Consequently, the Generation IV Materials NTD and the NGNP Materials R&D Program Manager work closely to correctly define materials R&D program area priorities and detailed work scope to be performed.

The NTD of the Generation IV Reactor Materials Cross-cutting Program will ensure that cross-cutting materials R&D activities include:

- Supporting the NGNP Materials R&D Program activities with a minimum of duplication and overlap
- Supporting the NGNP product team efforts to ensure integration of product requirements into the R&D activities.

1.5 Program Interface with Design Organizations

The NGNP Materials R&D Program is designed to perform materials R&D activities on high-priority issues and support ASME Code and ASTM Standards activities that directly support the NGNP. The results obtained from the NGNP Materials R&D Program will be used to support the NGNP design and licensing efforts after these tasks are formally initiated.

2 PRELIMINARY DESIGN FRAMEWORK

This section briefly describes the preliminary design framework for the NGNP based on the current information available from potential reactor designs and from scoping studies performed at the INL, including environmental and design characteristics; discusses the QA requirements needed to obtain NRC licensing; and outlines the paths to obtain ASME codification and qualification of component materials.

Because no pre-conceptual design currently exists for the NGNP, the GT-MHR design and the PBMR design, developed by the GA and the PBMR Company, respectively, have been used to provide the starting point for the NGNP design for purposes of Section 2.2. GA and AREVA/Framatome are currently proposing a PMR designs and the PBMR Company is currently proposing PBR designs. The GT-MHR operational requirements were used to estimate operational requirements for the NGNP by adding estimated deltas to the GT-MHR operational requirements. Therefore, only generic temperatures, neutronics, and conditions or features are used for illustration in this program plan.

The environment expected for the NGNP will be very challenging for the structural materials. The sustained operating temperature may reach 1000 °C or higher in a helium atmosphere with a pressure of 7.5 MPa and flow velocities on the order of 40m/s. A pure helium atmosphere would not cause environmental degradation of high-temperature materials, but the helium could be contaminated with gaseous impurities such as CO, CO_2, CH_4, H_2, H_2O, and O_2. A reducing atmosphere, for instance, may be quite aggressive for conventional high-temperature alloys since they are typically designed for an oxidizing environment and designed to form a thin protective Cr or Al oxide layer to protect the alloy from attack. High-velocity flowing gases may also contain particulates from abrasion of the graphite or other materials in the system. A particulate-laden, high-velocity gas also raises the potential issue of particle erosion in some components.

To select materials for the NGNP reactor and predict their performance for a time period up to 60 years, it is necessary to identify the degradation mechanism(s) for different gas compositions and determine the kinetics of deterioration. An environmental testing program will determine the corrosion and oxidation performance of candidate alloys and the effect of environmental degradation on mechanical properties. While it might be feasible to predict reactions resulting in alteration of surface chemistry for the gas compositions of interest, the influence of high gas velocity and particle erosion are nearly impossible to predict without appropriate high-velocity testing.

2.1 Environmental Framework

2.1.1 Gas Environment

The expected contamination levels of the helium coolant must be ascertained so as to bound the helium test environments for determining the materials properties of the structural materials. Small amounts of impurities can contaminate the coolant from a variety of sources throughout the reactor system and quite small amounts of these contaminants can degrade the materials both by corrosion processes and by effects on mechanical properties. Carburization and decarburization are issues of particular interest. In the case of this system, the effects of O_2 may not be a significant problem because the large amount of graphite in the core at high temperature reacts with the O_2 to form CO or CO_2. From a corrosion viewpoint, it is assumed that the reactor internals, piping and IHX will operate in a helium environment, and the externals of the reactor, including the pressure vessel, will operate in air.

The interactions between structural materials in the helium atmospheres associated with gas-cooled reactors have been the subject of numerous investigations (see Kimball)[2]. The results of these studies conducted by various organizations in the United States, 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 internally or at the surface. Depending on their rate, these corrosion reactions can substantially affect long-term mechanical properties such as fracture toughness, fatigue, crack-growth rate, etc.

Typical, simulated, advanced High-Temperature Gas Reactor (HTGR) helium chemistries used in various previous test programs are shown in Table 1. Because of the low partial pressures of the impurities, the oxidation/carburization potentials at the metallic surface of a gas mixture are 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 should react with all the 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 1. Composition helium environments (advanced HTGR) used in past tests [3]

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 proposed in the various experimental sections that follow is 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. 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 will be performed at temperatures up to at least 50 °C above the proposed operating temperature for the various metallic components.

2.1.2 Irradiation

This section provides a background on irradiation effects and estimates of neutron fluxes and fluences. When a material is irradiated, virtually every property will change. This includes physical dimensions, as well as mechanical, electrical, magnetic, thermo-physical and other properties. The reason for this is that the existing crystal and defect structure is deconstructed and reconstructed on an atom-by-atom basis during irradiation. In a high-dose irradiation, each atom may be displaced from its lattice site numerous times. The standard measure of radiation dose in metallic materials is the displacement per atom (dpa). Conditions during irradiation, such as temperatures, dose, dose rate, and local materials composition, determine the property changes that will ultimately result. Three of the irradiation-induced changes of greatest concern are swelling, irradiation creep, and embrittlement.

Swelling is normally an isotropic volume expansion of an irradiated material. It occurs by the net absorption of interstitials at dislocations, with a corresponding net number of vacancies accumulating at cavities. It may reach tens of percent or more at high doses (e.g., tens to hundreds of dpa). In near anisotropic graphite, swelling can itself be anisotropic and is highly dependent upon texture of the graphitic microstructure and the macroscopic direction of a component with respect to the crystal texture.

Irradiation creep is a shape change in compliance with an applied stress, in excess of ordinary thermal creep. It occurs even at quite low temperatures, where thermal creep may be negligible. Dislocation-climb creep occurs by the asymmetrical partitioning of self-interstitials and vacancies to dislocations differently oriented to the stress axis. Climb-enabled glide creep occurs when a dislocation climbs and overcomes an obstacle, permitting it to glide. Creep may therefore result directly from net climb of particularly oriented dislocations, or indirectly from any climb that triggers glide in response to the applied stress.

Embrittlement occurs, broadly speaking, by two processes. In the first type of process, hardening of the material progresses by creation of many types of obstacles by radiation. This hardening reduces ductility. Most irradiation-induced hardening centers are so small they are beyond the ability to detect with transmission electron microscopy. However, atom probe field-ion microscopy has contributed significantly to the knowledge of the structure and properties of these ultra-fine hardening features. The second type of process is grain boundary weakening, caused by preferential diffusion of transmutation products, such as helium, or tramp elements, such as phosphorus, to the grain boundary.

Swelling, irradiation creep, and embrittlement have received a great deal of experimental and theoretical attention. As a result, a certain measure of understanding of these phenomena has been achieved, but investigation of these processes in the particular alloys, graphites, and structural composites being considered for NGNP applications will still be required under the particular conditions of interest. The activities needed to assess these changes are incorporated into appropriate sections of the qualification test plans.

2.1.3 High-Temperature Exposure

At high temperatures, thermally activated processes such as microstructural changes, plastic flow, and some types of fractures produce a number of time-dependent degradation mechanisms that must be recognized in the design and operation of high-temperature components.

In regard to microstructural changes, there are several concerns to the NGNP. First, the RPV will most likely be fabricated from ferritic/martensitic steel that derives its strength from a fine precipitate of carbides formed on highly dislocated lath martensite boundaries. With time, these precipitates will coarsen and the lath structure will reform into a fine-grain structure with much lower tensile and creep strength than the starting steel. The rate at which this aging process occurs is highly dependent on the elemental constituents in the carbide microstructure. A second time-dependent degradation mechanism that occurs in structural steels is that of intermetallic phase precipitation. In this process, coarse intermetallic phases precipitate and solid-solution strengtheners are removed from the matrix and impart brittleness to the grain boundaries. In stainless steels and nickel-base alloys that are potential candidates for select core internal components, piping, and other high-temperature components, some strengthening is often derived from stable carbides and fine dispersions of intermetallics that develop in-service. With time, these beneficial precipitates may coarsen or dissolve in preference to less desirable precipitate phases. Again, loss of strength and embrittlement are concerns. Work is needed in the NGNP Materials R&D Program to define the kinetics of the precipitation processes and predict the development of metastable, and eventually, the stable microstructures.

High-temperature yield strength and resistance to plastic flow are properties that are important in structural components. Good resistance to thermal transients, mechanical fatigue, ratcheting, and buckling depends on materials with good short-time strength properties. At the extreme temperatures expected in the NGNP components, the yield and flow properties of the structural materials are expected to be very rate sensitive and will be more sensitive to loading rates in the components. To address these issues, the materials testing program needs to produce information that can lead to improved analysis methods that accommodate greater rate dependency of short-time deformation and fracture. For very long service times there are additional concerns. The database on which allowable stresses are based is quite limited for several of the candidate materials, particularly at the upper temperature range that service in the NGNP will require. New deformation and fracture mechanisms may prevail at the long time and low stresses thought to represent steady-state operation of the NGNP. It is critical that predictive continuum damage mechanics models be developed on a sound metallurgical basis.

2.2 Design Characteristics

The discussion in this section provides some specific information based on the PMR and PBR conceptual designs, however, the actual conceptual design selected for the NGNP could be different from the information noted. Therefore, the information provided should be viewed as illustrative but not specific of the NGNP.

2.2.1 Component Material Life Prediction Modeling

The usable lifetime of materials in service are determined by the combination of initial quality, normal service conditions, and the cumulative effect of off-normal and/or accident conditions encountered. For essential components, an optimum materials selection will have a test data set that provides data on time dependent failure mechanisms such as corrosion and creep that bound the expected product lifetime. However, obtaining such a data set is generally impractical. The desired design life for most components for the NGNP have been set at a nominal 60 years and that is generally longer than most needed reactor system materials have been available.

Developing defensible design and regulatory arguments for the viability of materials beyond their existing test data set involves the combined use of test data and modeling that accurately predicts the effects of relevant time-dependent failure mechanisms. A standardized and structured approach to prediction of long-term behavior in materials has been documented in ASTM C-1174 [4].

This standard requires use of problem definition, testing, modeling, and model confirmation to predict long-term behavior. Testing typically involves attribute tests, characterization tests, accelerated effects tests, service condition tests, confirmation tests, and analog tests or analyses. This standard or a similar, approved standard shall be used to guide life prediction modeling activities.

2.2.2 Core Internals and Pressure Vessels

This section deals with the graphite and other internals inside the RPV. It should be noted that the materials of construction of many of the core internals has not been determined because the conceptual design for the NGNP has not been completed. In many locations in the text the word “internals” has been used. This is intended to mean either metallic or non-metallic components. In most cases it is known which components will be fabricated from graphite, however, in many cases it is not known whether a metallic component will be used in the design. Some good examples of this for the GT-MHR design are the upper core constraint, control rod guide tubes and the control rods. These components may be metallic or non-metallic and the composition will be determined as a function of the exposure temperature and neutron fluence based on the conceptual design.

2.2.2.1 Graphite Internals. The GT-MHR graphite components will be discussed initially and the PBMR components will be subsequently discussed. Descriptive information for the GT-MHR is available.[5]

The graphite core of the GT-MHR is a right circular cylinder composed of 102 columns each containing 10 blocks. A standard block is hexagonal in shape with a dimension of 0.36 m across the flat and height of 0.8 m. The cylinder is arranged in 11 circular rings. The inner reflector uses the first five rings; the active core uses rings 6, 7, and 8; the outer reflector composed of rings 9 and 10; and ring 11 is the permanent outer reflector. On top of the core column is a reflector block and a half height upper plenum block that caps the column. Below the core column is a bottom reflector block and two half-height insulation graphite blocks. Under each column is a graphite pedestal. The pedestals rest on two additional insulation blocks (graphite or ceramic), which in turn sit on the core support floor.

Each block has four dowel pins protruding from the top; subsequently, each block has four dowel pinholes in the bottom. These dowel pins lock the column together. The thermal expansion and flow induced motion in each block creates shear stresses on the pins and reactive stresses in the dowel pinholes.

The top and bottom insulator graphite blocks see fluences on the order of 9.1E15 n/cm^2 per year (E > 0.1 MeV) and negligible dpa. Normal operating temperatures are about 500 °C for the top blocks and about 1000 °C for the bottom blocks. The off-normal temperatures are about 1200 °C for the top blocks and 600 °C for the bottom blocks. This is due to a flow reversal in the core during off-normal conditions.

The core pedestal supports are graphite columns that support each hexagonal column in the core except for the permanent reflectors. Spaces between the pedestals create the lower plenum of the reactor where all the coolant channels flow. The fluence levels are higher in the plenum than in the insulator blocks, 3.7E17 n/cm^2 per year with negligible dpa. Coolant temperatures in the lower plenum reach about 1200 C at selected locations and an average temperature of about 1000 °C during normal operation with off-normal temperatures reaching 600 °C.

Upper plenum graphite blocks are half the length of regular blocks and cap off the graphite columns. The fluence, dpa, and temperatures are the same for the top insulator blocks.

Replaceable outer and inner reflector graphite blocks are placed on the inside and outside of the core ring. The inner reflector sees the highest temperatures and fluences. At the inside interface of the core ring, the fluence is 6.7E20 n/cm^2 per year (E > 0.1 MeV) with a dpa of 0.56 per year. The peak fluence in the outer reflector block is 1.8E20 n/cm^2 per year with a dpa of 0.16 per year. Temperatures in the outer reflector blocks are 750 °C for normal conditions and 1100 °C for off-normal conditions. Peak temperatures in the inner blocks are about 1000 °C during normal operation conditions and 1200 °C during off-normal conditions.

The last graphite internal components are the fuel blocks (see Figure 2). There are 210, 0.5 inch, fuel channels and 108 coolant channels in a standard fuel block. In each fuel channel, there are 15 fuel compacts, approximately 2 inches in length, pressed into the channel. When the fuel channels land on a dowel pin the numbers of compacts reduce to 14 in a channel. When a block contains a control rod the number of fuel and coolant channels is different. In total, there are 3,126 compacts in each standard element and 2,766 compacts in a control and RSSC element. Each compact is a carbonaceous dowel containing dispersed TRISO coated particle fuel.

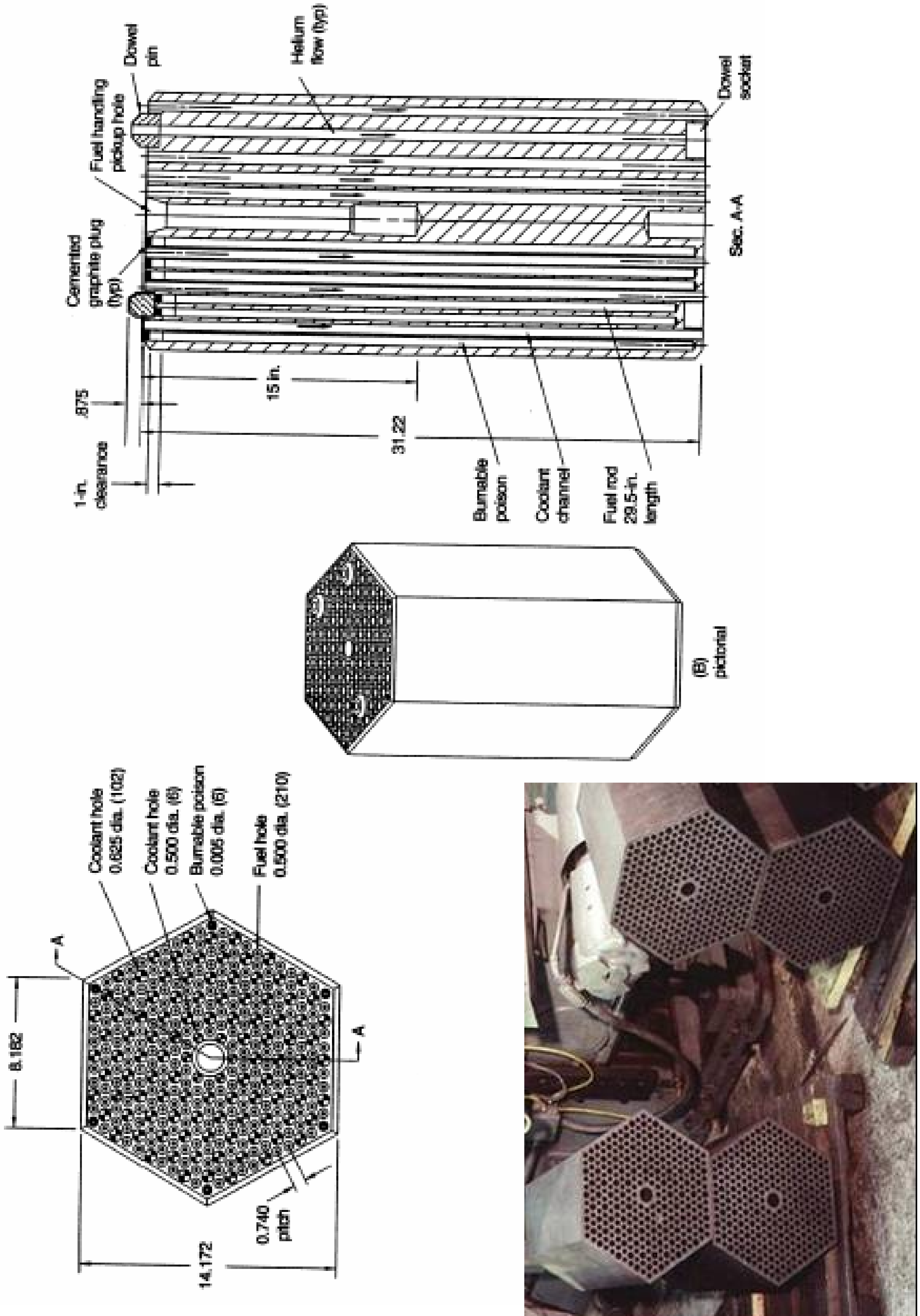

Figure 2. GT-MHR Fuel Blocks

The active core is composed of three rings of fuel blocks, which see the highest temperatures and fluences of all the graphite components. The fluence of the blocks is 9.9E20 n/cm2 per year with a dpa of 0.82 per year. Normal operating peak temperatures for the fuel blocks are approximately 1250 °C. Off-Normal peak temperatures for the blocks can be as high as 1600 °C.

The graphite internals of the PBMR are illustrated in Figure 3. The annular shaped reactor core, which is composed of a bed of fuel pebbles, is supported by the bottom reflector and is laterally restrained by the central reflector and side reflector. The central and side reflectors are constructed from stacks of large interlocking (keyed) graphite blocks. Figure 4 illustrates the central reflector of the PBMR and shows the interlocking and key-structure.

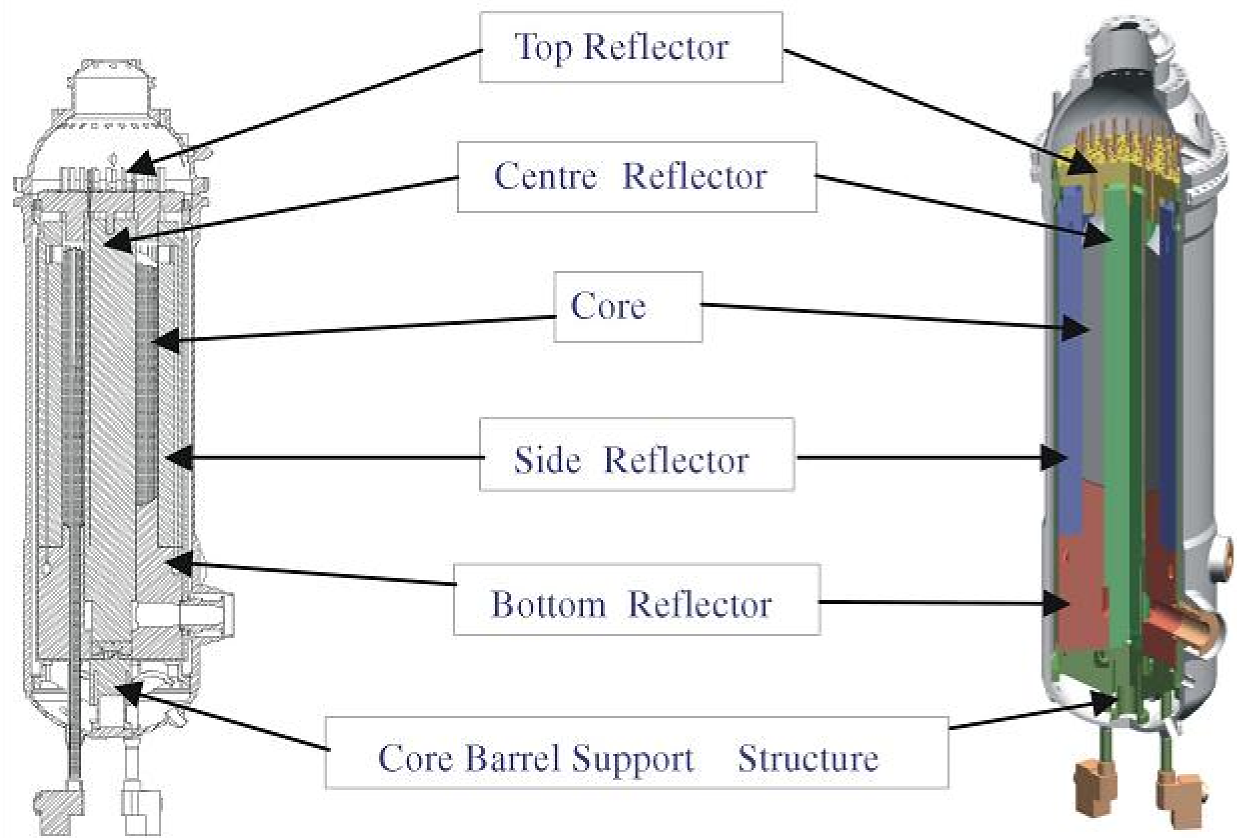

Figure 3. The graphite core internals of the PBMR

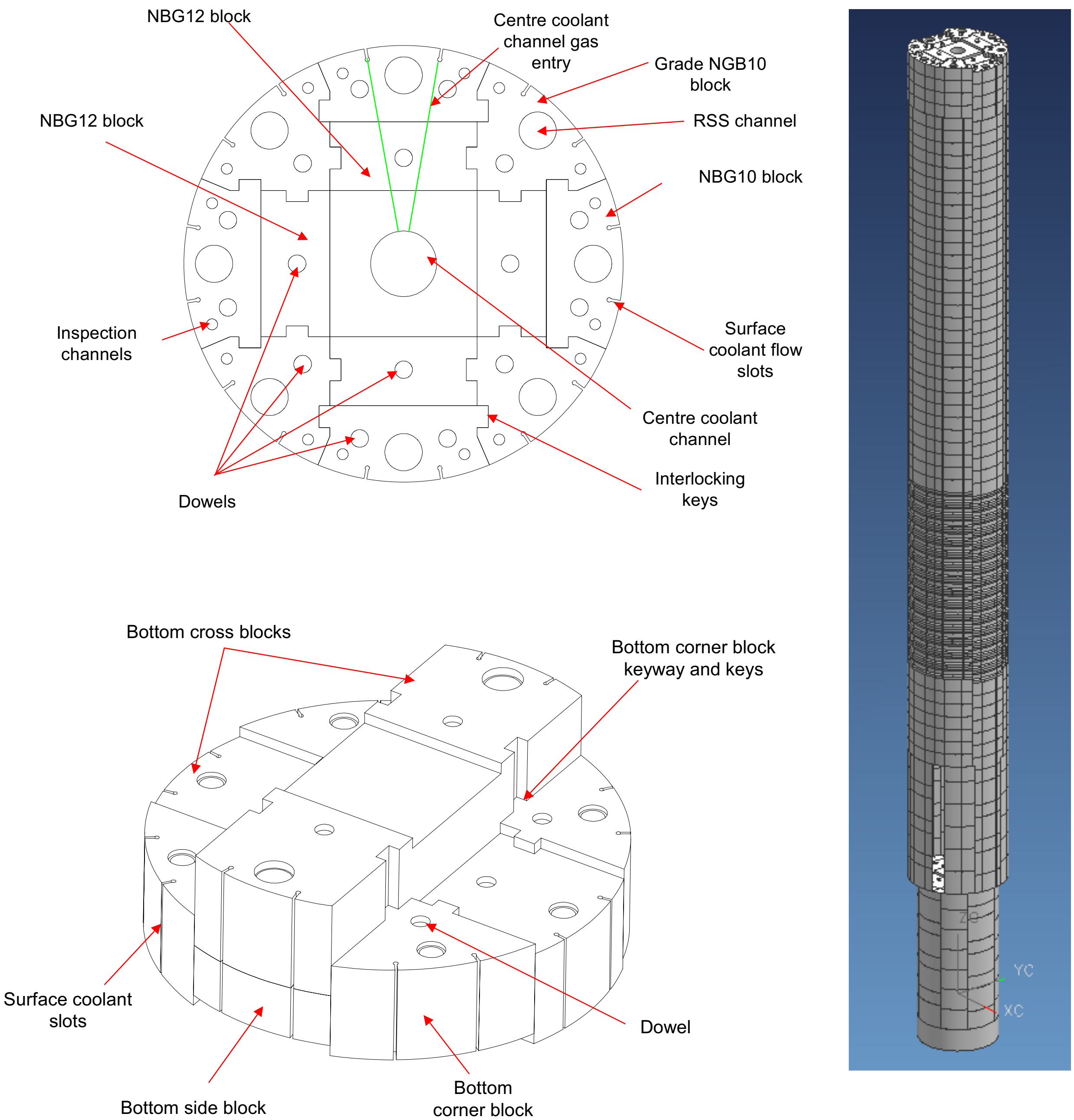

Figure 4. The central reflector graphite column of the PBMR

The currently designated graphite grades for the PBMR core internals are SGL NGB-10 and NGB-12. Both graphites are extruded, pitch coke graphites manufactured at SGL's Chedde facility in France. The pitch coke used is the same as that currently used for the production of the United Kingdom (UK) Advanced Gas Reactor (AGR) graphite fuel sleeves, and thus there is considerable production experience for this coke and graphite. Consideration is also being given to grade NGB-18, a vibrationally molded graphite.

The volume average thermal flux in the core is 7.90 x 10^{13} n/cm^2·s [E>1.86 eV]. The volume average fast flux, which is more relevant since it is fast neutrons that displace carbon atoms and cause the dimensional and property changes, is 3.26 x 10^{13} n/cm^2·s [E>0.1 MeV]. Typical lifetime fast fluences for the graphite core internals for a 35 effective full power year life are:

- Fuel Pebbles 2.65 x 10^{21} n/cm^2 [E>0.1MeV]
- Upper reflector edge (maximum) 0.21 x 10^{22} n/cm^2 [E>0.1MeV]
- Outer reflector side (maximum) 3.85 x 10^{22} n/cm^2 [E>0.1MeV]
- Inner reflector side (maximum 4.73 x 10^{22} n/cm^2 [E>0.1MeV]
- Lower reflector edge (maximum) 0.53 x 10^{22} n/cm^2 [E>0.1MeV]

The neutron fluence to the central and side reflector is clearly very significant, potentially necessitating their replacement during the life of the reactor. Consequently, the graphite blocks of the central reflector and the inner side reflector (Figures 4 and 5) are designed to be removable. The average fuel temperature in the PBMR varies axially through the PBMR core. The fuel temperature is ~500 °C at the top of the core where the coolant gas enters and increases to ~ 900 °C at the reactor mid plane.

The peak mean fuel temperature is ~1000 °C close to the bottom of the core. The PBMR fuel temperature is always less than 1160 °C. The peak graphite temperatures under normal operating conditions are also likely to be ~1000 °C. Consequently, those areas of the core (inner edge of the side reflector and the outer edge of the central reflector column) that experience high temperatures (>600 °C) and high neutron fluence (>3.0 x 10^{22} n/cm^2 [E>0.1 MeV]) will experience significant distortion due to the irradiation induced shrinkage reversal to growth. Temperature and fast neutron fluence gradients will cause differential stresses in the core, which will relax due to irradiation-induced creep of the graphite.

The PBMR core will also utilize carbon-carbon (C_f/C) composites. Anticipated applications include the core lateral restraints (Figure 5) and the hot gas outlet duct and interface components. Moreover, C_f/C composites will be utilized as metal replacements in selected interface components and for thermal expansion compensation of the core. The majority of these applications will be in low neutron fluence areas where the only affected property will be thermal conductivity. However, applications such as control rod cladding (if adopted) would experience greater fluences, and thus undergo dimensional and property changes. The GT-MHR is expected to use C_f/C composites in a similar manner as the PBMR. [5]

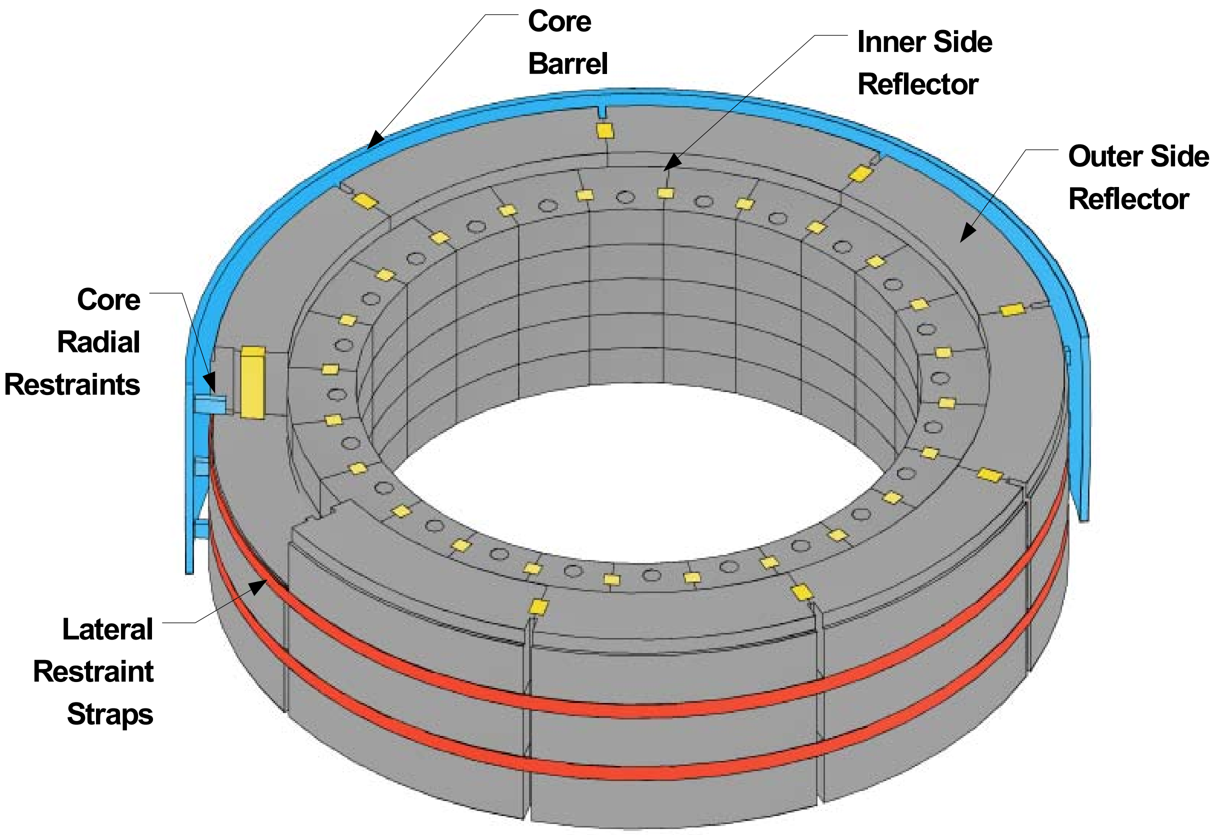

Figure 5. The inner and outer graphite side reflector of the PBMR

2.2.2.2 Internals and Pressure Vessels- NGNP Prismatic Design. Since there is no NGNP prismatic reactor design at present, the GT-MHR components will be used for illustrative purposes. Due to a lack of a conceptual design several different studies have produced different temperatures and core information. The source documents for these values have been referenced***.***

The three main vessels in the GT-MHR design, the RPV, cross vessel (CV), and secondary vessel (see Figure 6), represent the pressure boundary of the primary coolant. The GT-MHR uses a closed Brayton cycle to generate electricity where helium coolant flows out of the reactor directly through the main turbine. The helium exiting the main turbine is re-pressurized to the inlet operational conditions and pumped through the reactor. The NGNP Prismatic Reactor design operational inlet helium pressure and temperature for the reactor is less than 490 °C [6] at a pressure of 7.4 to 8.0 MPa. The inlet helium flows between the core barrel and the RPV maintaining the RPV at a cooler temperature than the core. Nominal operating temperature of the RPV wall is 470 °C. The helium exits the reactor core at temperatures less than 1000 °C at pressures of 7.33 to 7.93 MPa. The pressure drop across the core is ~70 kPa.

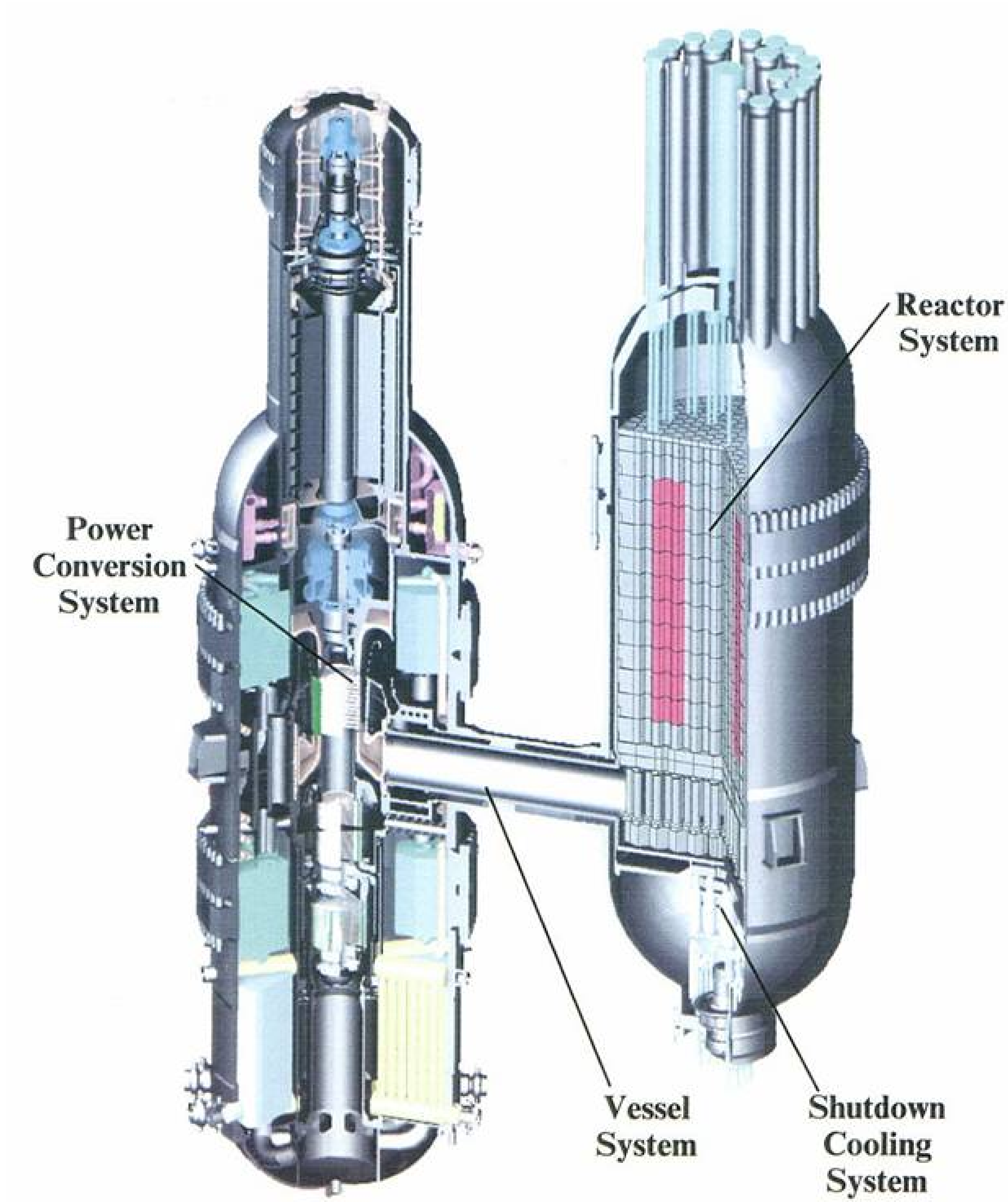

Figure 6. GT-MHR Full Section.

General Atomics recently presented a conference paper providing details of their H2-Modular Helium Reactor. The prismatic core has an inlet temperature of 590 °C and an outlet temperature 950 °C with a primary pressure of 7 MPa. Details were presented of a preliminary investigation into using bypass flow from the high-pressure helium compressor to supplement the inlet flow in reducing the temperature of reactor pressure vessel wall. Using 16.0 kg/s of 140 °C helium from the high-pressure compressor, the vessel wall temperature is decreased from 480 °C to 338 °C [7].

The components in the reactor internals system other than graphite that will experience significant radiation exposure are the core barrel, upper plenum shroud, core support floor, upper core restraint, and the shutdown cooling system (SCS; heat exchanger) shell and tubes. The design life of the non-replaceable core internals is 60 years. For some sub-components of those systems where temperatures are excessive, non-metallic materials may be specified. Relative to current light-water reactor (LWR) vessels and internals, the structures in the NGNP will be exposed to relatively low neutron doses. However, because of the significantly higher operating temperatures for the NGNP, the materials for most of the internal structures will not be the same as those for the LWRs for which a vast amount of experience is available. For the NGNP reactor internals, (depending on the specific component) normal operating temperatures may range from 600 to less than 1000 °C.

To determine the fluences (> 0.1 MeV) and displacements per atom (dpa) for each of the components, a Monte Carlo physics code evaluation (MCNP) [8] was performed on a prismatic core model. The model was limited to radial responses only at the mid-line of the core. The fluence for the core barrel was composed of dpa doses from iron, nickel, and chromium. The RPV wall fluence is assumed to be the same as the core barrel fluences. The MCNP model did not include the outer control rods or the borated steel pins in the permanent reflector; thus, the fluence and dpa for the RPV wall is an upper bound. The fluence for the RPV is 1E19[6] n/cm^2 fluence (> 0.1 MeV) and the dpa is 0.077 for 60 years.

2.2.2.3 Reactor Pressure Vessel - Prismatic Design. The operational goals set for the NGNP early in the program were a gas outlet temperature of 1000 °C, and a maximum 1600 °C core gas temperature as a result of an unmitigated accident condition. From the viewpoint of the pressure vessel materials effort, these goals must be viewed as optimums with the range of feasibility for normal gas outlet temperatures between 850 °C and 1000 °C.

The optimum conditions expected for the RPV for the current commercial reactor designs and their NGNP counterpart are shown in Table 2. RPV temperature reductions at least equivalent to the reduction in gas exit temperature would be expected for normal operation if gas exit temperatures below 1000 °C were implemented. Maximum accident RPV temperatures would still reach the Table 2 values for a short time unless active RPV cooling systems are included in the design.

Table 2. Comparison of Nominal Parameters for Prismatic and Pebble Bed Design.

RPV Parameter	GT-MHR [5]	GA-Prismatic [9]	Prismatic NGNP [6]	PBMR [10]	NGNP PBR [6]
Nominal Gas Outlet Temperature (°C)	850	950	1000	900	1000
Nominal Gas Inlet Temperature (°C)	491	590	490	500	490 [b]
RPV Normal Operating Temperature (°C)	495	350	470[c]	300	465 [d]
RPV Worst Case Accident Temperature (°C)	565	530	560 [6], [e]	450	560 [f]
Inlet Gas Pressure (MPa)	7.07	7.07	7.07	8.9	7
Outlet Gas Pressure (MPa)	7.02	7.02	7.02	8.6	6.5?
RPV External Diameter (meters)	8.2	8.2	8.2	7.02	7.06
RPV Nominal Wall Thickness (mm)	100-300	100-300	100-300	120-220	120-220
RPV Nominal Height (meters)	23.7	23.7	24	27	19
Maximum Radiation Fast Fluence in the RPV in the RPV over 60 years (n/cm^2)	3×10^{18} [11]		1×10^{19} [6], [g]	4.5×10^{19} [6]	3.0×10^{19}

Figure 7 identifies the components of the RPV. The estimated physical dimension of the RPV is a diameter of less than nine meters with wall thicknesses between 100 mm and 300 mm. The vessel itself can be made of welded sections of different thicknesses. The height of the vessel is less than 2 meters.

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b 490 °C Based on recent analysis still pending publication

c If the temperature is 490 °C on the inside then a temperature drop is assumed to reach approximately 470 °C

d 490 °C Based on recent analysis still pending publication

e Fig 72, Fig 76 shows 560 °C

f 490 °C Based on recent analysis still pending publication

g Core Barrel, Neutron Energy Group 2, (5.105E9 x 3600 x 24 x 365 x 60 = 1E19

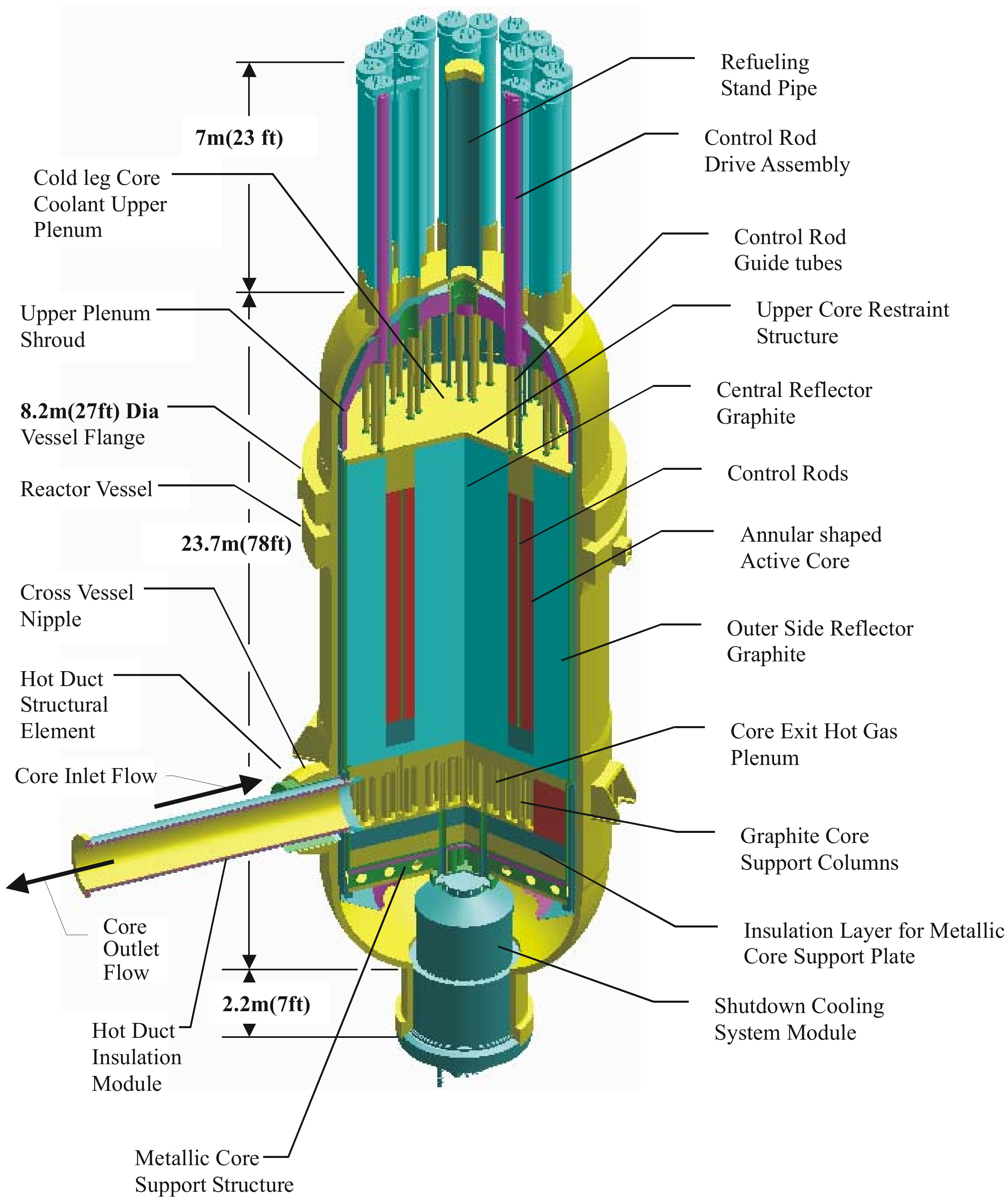

Figure 7. GT-MHR Reactor Vessel.

2.2.2.4 Cross Vessel – Prismatic Design. The CV is the pressure boundary for the exchange of helium between the RPV and secondary vessel (see Figure 8). The outside diameter of the vessel is expected to be on the order of 2.5 meters with a thickness of less than 100 mm. The CV is welded to the RPV and secondary vessel. To accommodate thermal expansion during operation, the secondary vessel is allowed to slide laterally away from the RPV. The helium flows out of the reactor at a temperature of about 1000 °C in a structural duct inside the CV and returns from the secondary vessel on the outside of the structural duct. The structural duct containing the 1000 °C helium is known as the hot duct. The hot duct uses ceramic insulation on the inside surface to reduce the service temperature to approximately the inlet helium temperature of 490 °C. The ceramic insulation is removable allowing inspection of the hot duct. The hot duct only sees the pressure differential of the core across its thickness. The hot duct is welded to the core barrel at the lower core plenum outlet and is connected to the secondary vessel by means of a metallic bellows. The hot duct is seal-welded to the metallic bellows. The bellows is mechanically connected to the turbine inlet shroud. The outer shell of the CV sees the temperature of the RPV at one end and gradually decreases to the lower temperature of the Power Conversion Vessel. As the CV expands pushing the secondary vessel away from the RPV, the mechanical bellows expands, thus maintaining the pressure boundary between exit and return helium.

The fluence and dpa seen by the CV and hot duct is the same as the RPV and core barrel where the attachment welds are made. The fluence and dpa in the remaining portion of the vessel will see a gradient over the length of the vessel decreasing to negligible values at the secondary vessel.

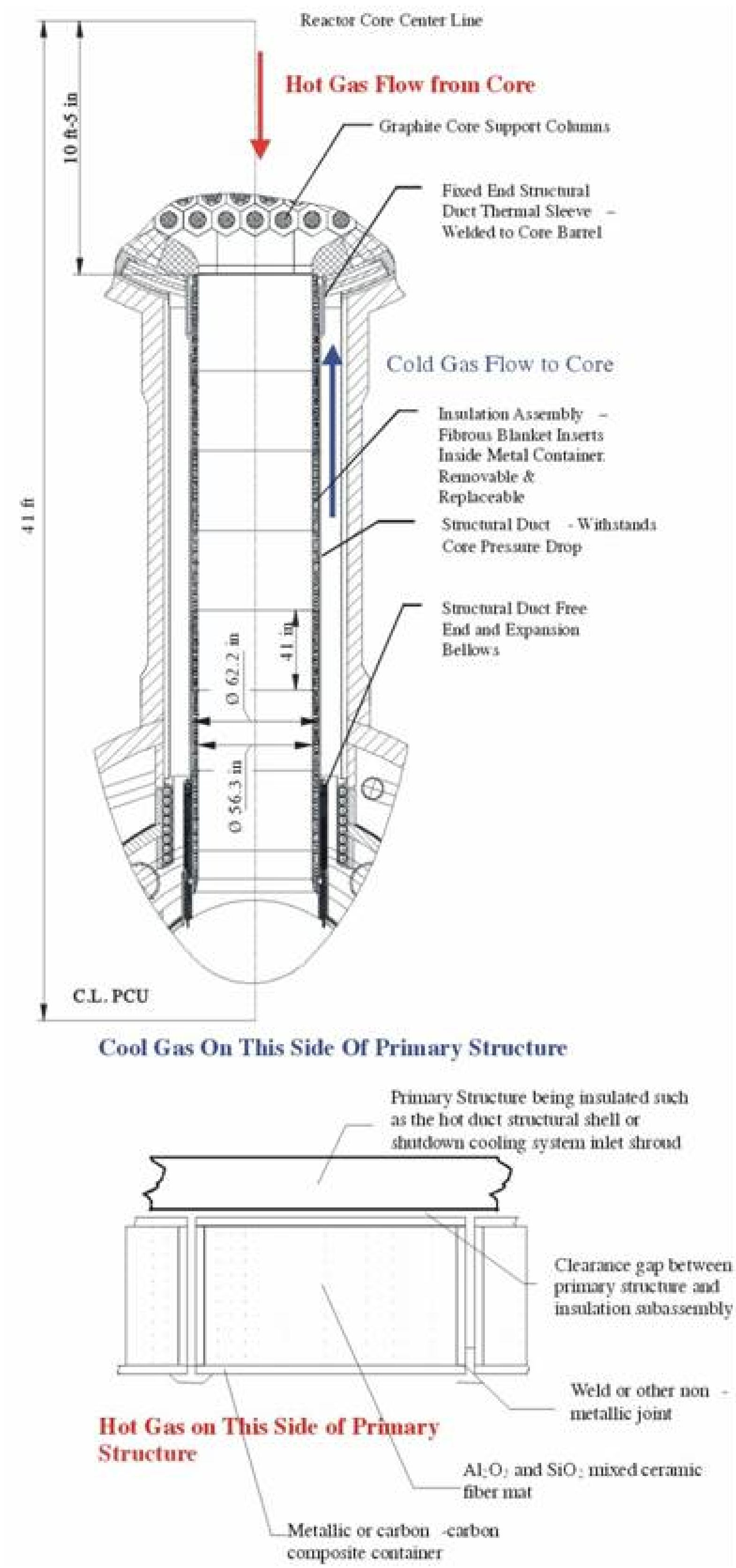

Figure 8. GT-MHR Cross Vessel.

2.2.2.5 Secondary Vessel - Prismatic Design. The secondary vessel (see Figure 9) houses the main turbine, generator, and associated turbo machinery and heat exchangers necessary to pump and reheat the helium to 490 °C at inlet pressures between 7.4 and 8.0 MPa. The vessel is on the order of 35 meters tall with outer diameters between 7 and 9 meters. The wall thickness is between 100 and 200 mm. The normal operating temperature for the vessel wall is 200 °C with an off-normal temperature of 300 °C. The design pressure of the vessel is between 5 and 6 MPa. The temperatures and pressure of the NGNP secondary vessel should not change much from the GT-MHR design. The fluence and dpa seen by the secondary vessel is negligible. The two components that will see the helium (<1000 °C) are the metallic bellows and a small portion of the turbine inlet shroud. The turbine inlet shroud is insulated to reduce the temperature seen by the structure except for a small portion that is un-insulated at the connection of the turbine opening. The remaining piping, turbo-machinery, and heat exchangers experience temperatures and pressures that do not challenge current ASME qualified materials. The main turbine, specifically the turbine blades, will be addressed later when more information is available.

2.2.2.6 Core Internals – Prismatic Design. The estimated lifetime of the core internals is 60 years. The fluence and dpa for these components, except for the control rods and guide tubes, are the same as described for the core barrel and RPV.

The core support floor is a structure of concentric rings welded together with radial beams originating from the center ring. The entire structure rests on supports forged into the lower head of the RPV. The floor supports the mass of the graphite core, core barrel, shroud, and upper core restraints. The structure is maintained at inlet helium temperatures by circulating helium from the inlet underneath the structure and insulating the structure from the core with ceramic or graphite blocks. The mass of the core is approximately 790 metric tons. The bounding dynamic load is the 0.3 g Safe Shutdown Earthquake.

The core barrel (Figure 10) is a metallic cylinder with a diameter of 6.8 to 7 meters, a height of ~14 meters and a thickness of 25 to 50 mm. The cylinder is welded to the core support floor. The core barrel physically restrains the graphite core during earthquakes and from radial thermal expansion during normal operations. The core barrel is centered and restrained in the RPV by keys that fit into corresponding keyways in the RPV. During operation, there is no space between the permanent reflector and the core barrel; the permanent reflector blocks remain in contact with the core barrel. The normal operating temperature of the core barrel is 600 °C. Temperature during off-normal conditions could reach as high as 700 °C for the core barrel [12]

The next internals component is the upper plenum shroud. This structure in the GT-MHR concept is metallic and sits on top of the core barrel and supports insulation on the inside surface. Inlet helium flows up the gap between the RPV and core barrel and through the slots in the shroud just above the joint with the core barrel and into the upper plenum cavity. The shroud forms an upper plenum cavity on top of the core and serves as a heat shield to the control rod and instrumentation drives during normal and off-normal operations. During normal conditions, the shroud is maintained at <500 °C by the circulating inlet helium. However, off-normal conditions could result in temperatures approaching 1200 °C in the upper shroud due to the flow reversal and metallic alloys will not survive under these conditions therefore, the issue of metallic or non metallic for this component will need to be considered in the future. The normal operating pressure in the upper plenum is between 7.4 to 8.0 MPa.

The upper core restraint is a structure fabricated of individual hexagonal boxes that have dowels on the bottom, which fit into the top of the upper plenum blocks. Each box is keyed to six of its neighbors, enforcing the lateral gap between graphite columns and allowing each graphite column to thermally expand in the axial direction independent of the adjacent column. Normal operating temperature for these structures is <600 °C, while off-normal temperatures could approach 1200 °C.

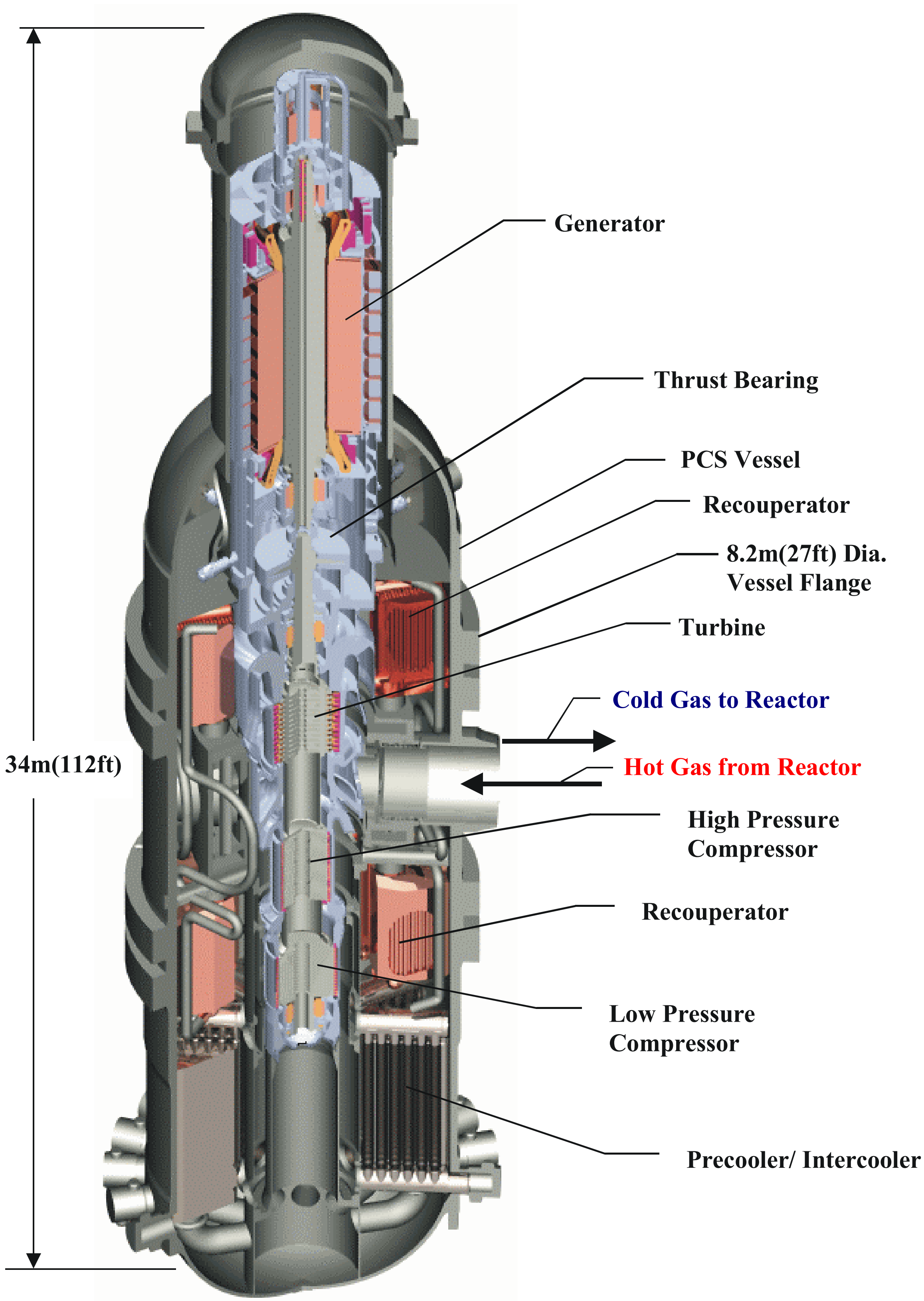

Figure 9. GT-MHR Power Conversion Unit.

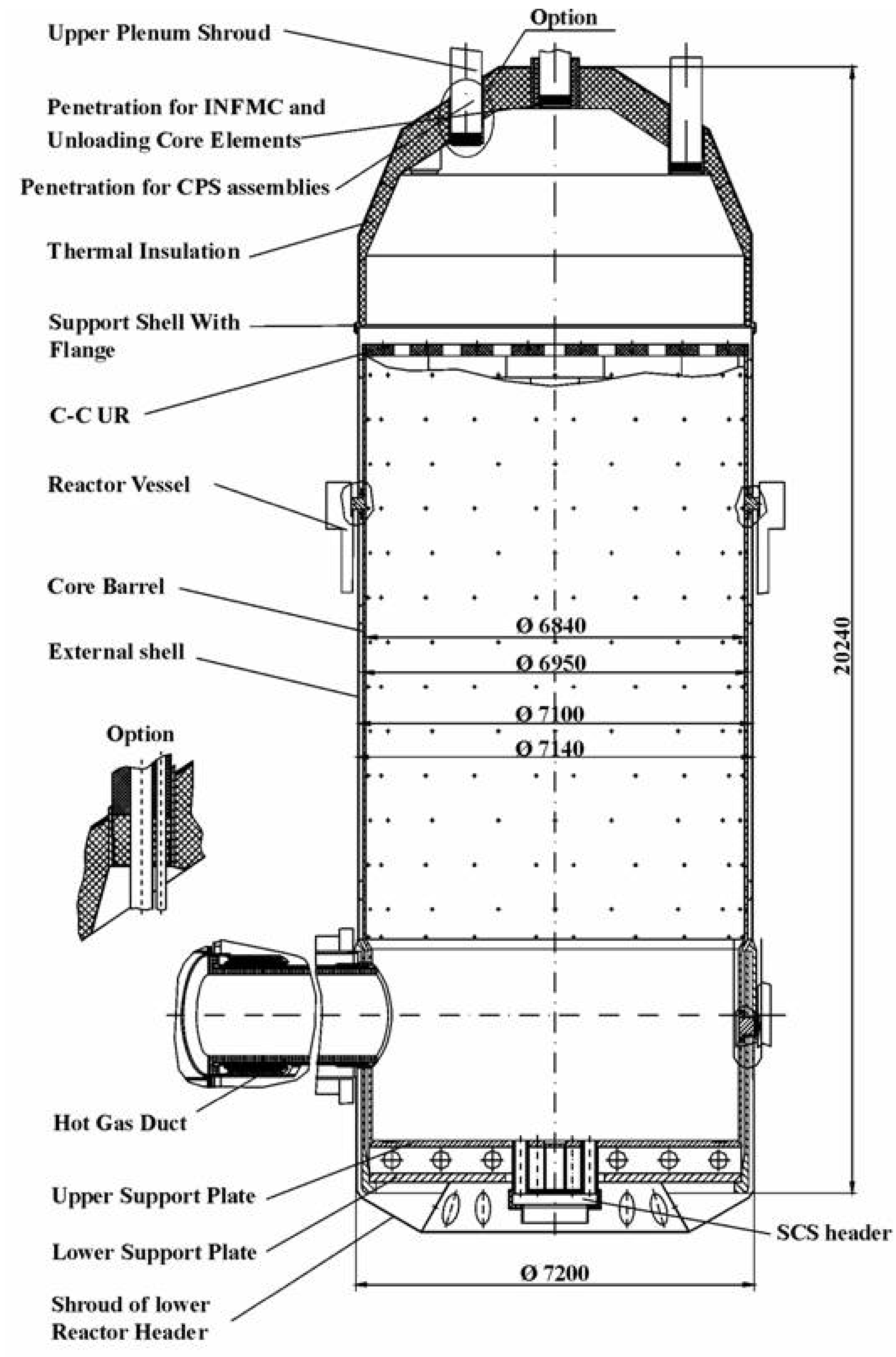

Figure 10. GT-MHR Core Barrel.

The shutdown cooling heat exchanger (Figure 11) is located in the bottom of the core and is used primarily to remove heat during refueling. The system can be used for normal and off-normal heat removal during shutdown. The upper portion is a helical tube heat exchanger in an environment of inlet helium at <500 °C mixed with <1000 °C flow from the lower core plenum. The tubes are between 12 and 19 mm thick. Water flows through the tubes at rates necessary to keep the water sub-cooled. During startup of the SCS, the helium temperature would increase to 1000 °C for a period of time and even to 1200 °C during startup of the cooling system in off-normal conditions. The operating pressure on the helium side is 7.4 to 8.0 MPa. The pressure on the waterside will vary but will be lower than the normal operating pressure of the reactor. The outside of the tube is 1000 °C and the inside will have pressurized water inside, that is the reason the tubes are ½ to ¾ inches in thickness.

The last internal components are the control rods and their guide tubes. In past prismatic gas reactor core designs, the control rods were metal tubes filled with B_4C graphite right cylinders with center annuli as shown in Figure 12. The rods transverse the upper plenum inside guide tubes from the upper inner head to just inside the core. The tubes are vented to the reactor pressure. The most important mechanical requirement of the control rods is that they cannot bow or deform during normal or off-normal conditions. The rods must remain straight to enable quick insertion at any time. The control rods that see the highest fluence and temperatures reside on the inside periphery of the core between the inner core and reflector. Normal operating temperatures reach 1050 °C with off-normal temperatures reaching as high as 1400 °C. Considering the high temperatures of these components, it is unlikely that metallic materials can be used solely and structural composites will likely be needed. These control rods see fluences of 6.7E20 n/cm^2 per year with dpa values of about 0.56 per year. These high fluences may limit the lifetime of the control rods to less than 40 years; therefore, the fluence and dpa are given on a per year basis. Since control rods may be changed out, the reactor lifetime (~60 years) is not limited by the life of the control rods.

Silicon-carbide/silicon-carbide (SiC_f/SiC) composites are being considered as a candidate material for the control rod sheath and guide tubes because metallic materials cannot withstand the level of neutron irradiation and high temperature of 1050 °C or higher found in the core. In addition, there is evidence that SiC_f/SiC composites show superior irradiation performance compared to other thermally stable composites such as C_f/C composites. Thus, SiC_f/SiC components have the potential to be lifetime components (no change-out required) within the expected high thermal and radiation environment of the NGNP core.

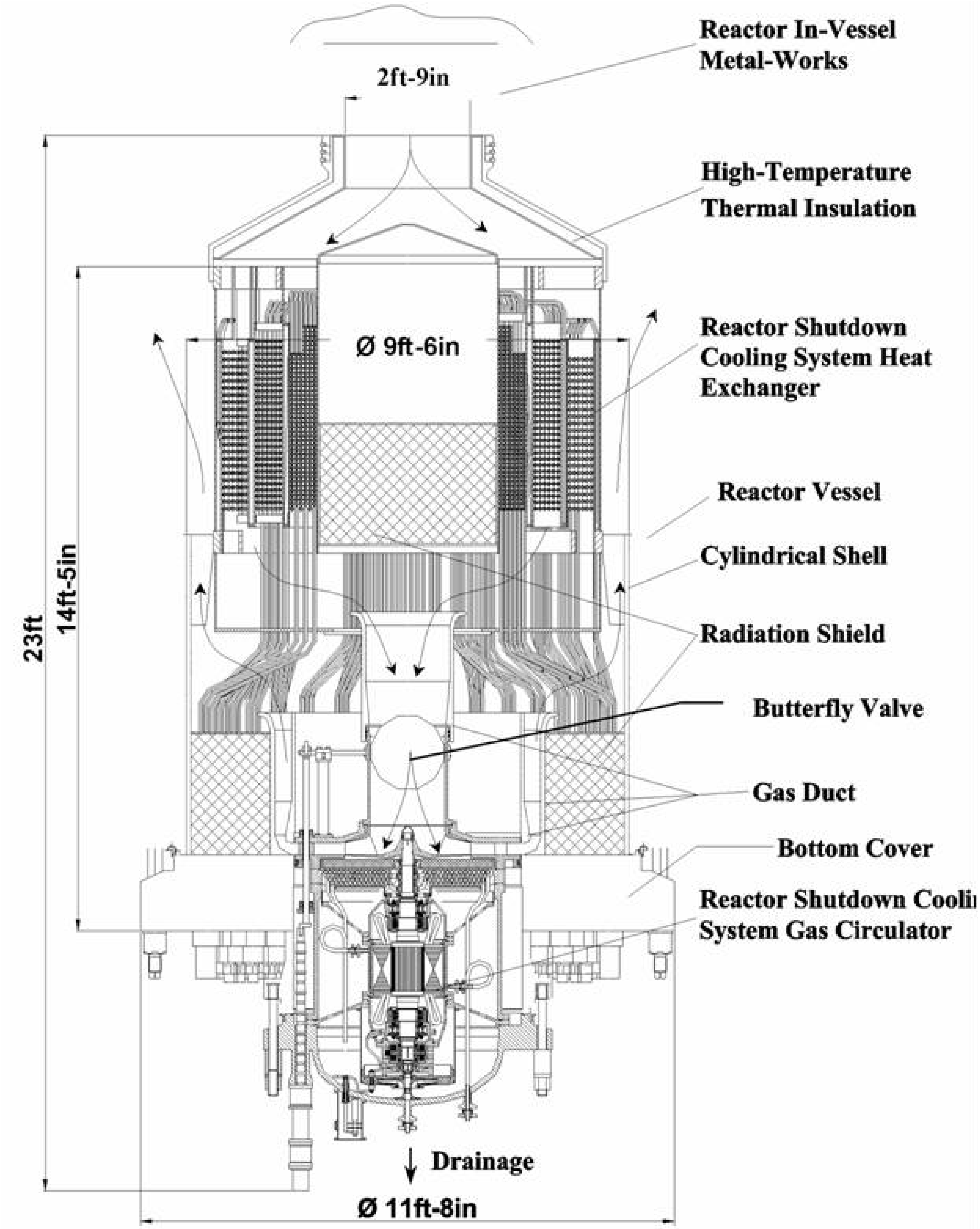

Figure 11. GT-MHR Reactor Shutdown Cooling System.

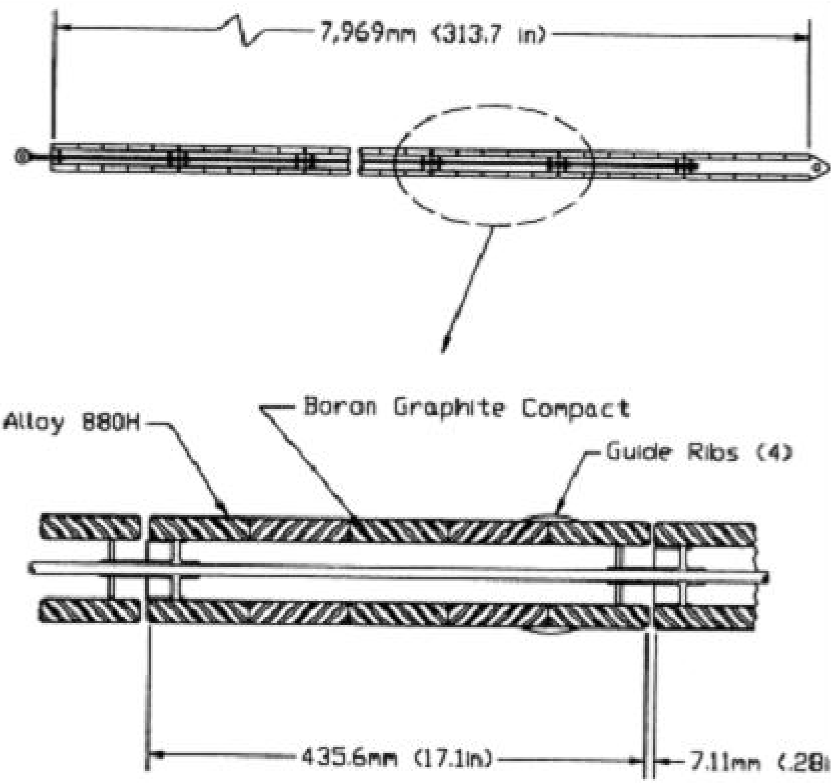

Figure 12. GT-MHR Control Rod Concept.

2.2.2.7 Metallic Internals and Pressure Vessels - Pebble Bed Design. There is no NGNP PBR design so the PBMR will be used for reference. A consortium is currently leading the development of this design.

The PBMR is a helium-cooled, graphite-moderated HTR, employing graphite fuel balls or pebbles, 6 cm in diameter, with TRISO ceramic particle fuel dispersed in the pebble. The ceramic fuel consists of a UO_2 kernel, (0.5mm) coated with layers of pyrolytic carbon and a silicon carbide layer for a total diameter of .92 mm. The helium gas from the reactor outlet is directly coupled to a gas turbine driven generator system forming a closed Brayton cycle. Recent design changes have incorporated a single shaft design where the high- and low-pressure compressor, the turbine, and the generator/reduction gear are driven by the same shaft. An overall view of the reactor and PCU are shown in the PBMR Module Building in Figure 13[13]. Figure 14 shows the components inside the PBMR pressure boundary, which include the reactor, the direct cycle power generation turbine, and high and low pressure turbo compressor components. The generator is outside for maintenance access.

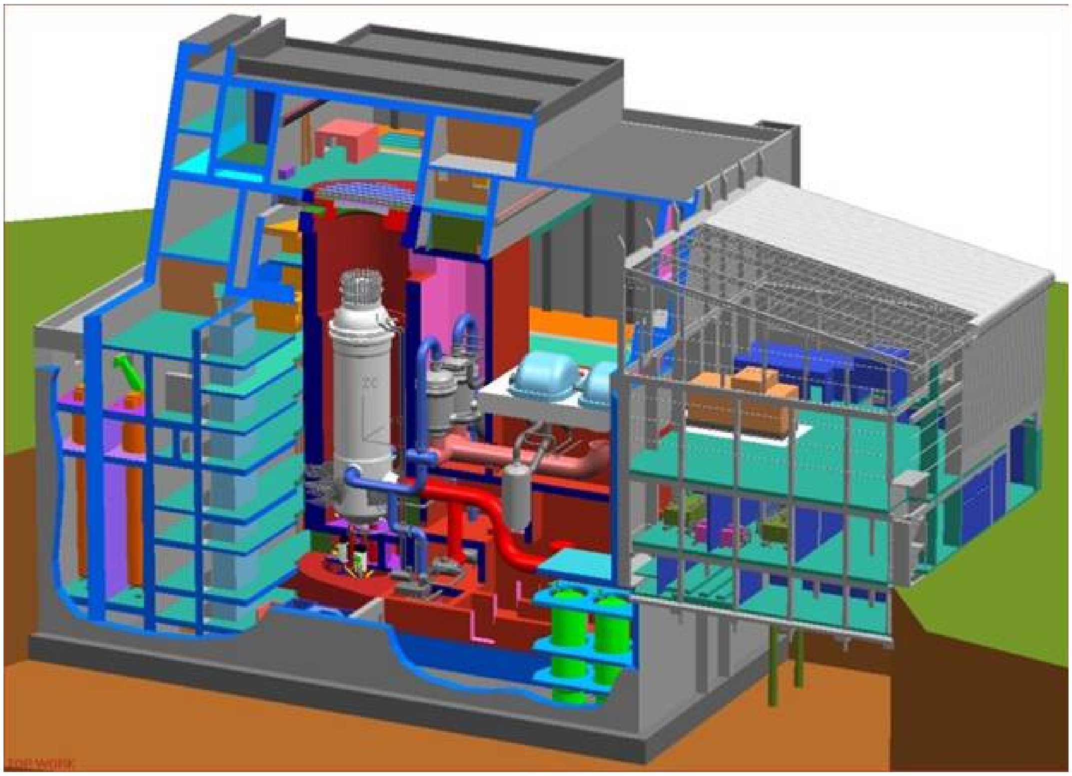

Figure 13. PBMR Single Module Building. [13]

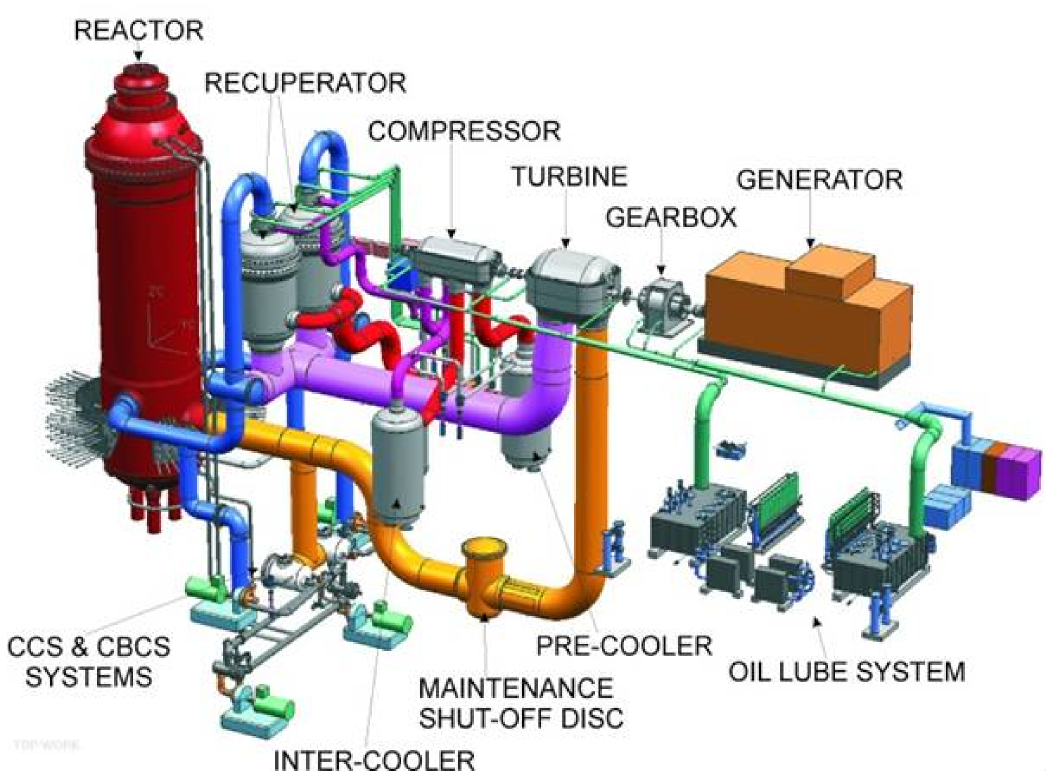

Figure 14. PBMR Pressure Boundary. [13]

Figure 15 illustrates the PBMR thermodynamic cycle. The helium exits the bottom of the reactor at a temperature of about 900 °C. The helium then expands in the High-Pressure Turbine that drives the High-Pressure Compressor (HPC). The helium then flows through the Low-Pressure Turbine that drives the Low-Pressure Compressor (LPC). The helium then expands in the Power Turbine, which drives the generator. The high-temperature helium then flows through the primary side of the recuperator where it transfers heat to the low temperature gas returning to the reactor. The helium that passed through the primary side of the recuperator is then cooled by means of a pre-cooler. The helium is then compressed by the LPC and cooled in the inter-cooler. The HPC then compresses the helium to 8.5 MPa. The cold, high-pressure helium stream then flows through the recuperator where it is pre-heated after which it returns to the top of the reactor.

The helium enters the RPV (Figure 16) at a temperature of about 500 °C through the cold gas inlet at a pressure of about 8.9 MPa. The inlet helium flows between the core barrel and the RPV maintaining the RPV at a cooler temperature than the core. Nominal operating temperature of the RPV wall is 380 °C. The helium moves downward between the hot fuel spheres in the core barrel (Figure 17).

The PBMR consists of a vertical steel RPV (Figure 16) 27 m high with an inside diameter of 6.2 m[14]. The pressure vessel material is ASME SA 508[15]/SA 533[16]. The pressure vessel is lined with a layer of graphite bricks. The core barrel surrounds and supports the graphite reflector (see Figure 5). This graphite layer serves as an outer reflector for the neutrons generated by the nuclear reaction and a passive heat transfer medium. The graphite brick lining is drilled with vertical holes to house the control elements. This graphite reflector encloses the core where the nuclear reaction takes place. Helium flows through the pebble bed and removes the heat generated by the nuclear reaction. Total height of core barrel is 22 m with an outside diameter of 5.85m[17].

The core barrel material is 316 stainless steel. The designs of the RPV and core barrel meet ASME Section III, Subsections NB and NG respectively. The Decompression Loss of Flow Accident (DLOF) maximum temperature for the core barrel is 621 °C and is covered under ASME Code Case N-201. The DLOF for the RPV is 450 °C and is covered under ASME Code Case N-499. The design life of this system is 40 years.

The fluences for the RPV, core barrel and reactor metallic internals are expected to be comparable to those discussed earlier for the GT-MHR.

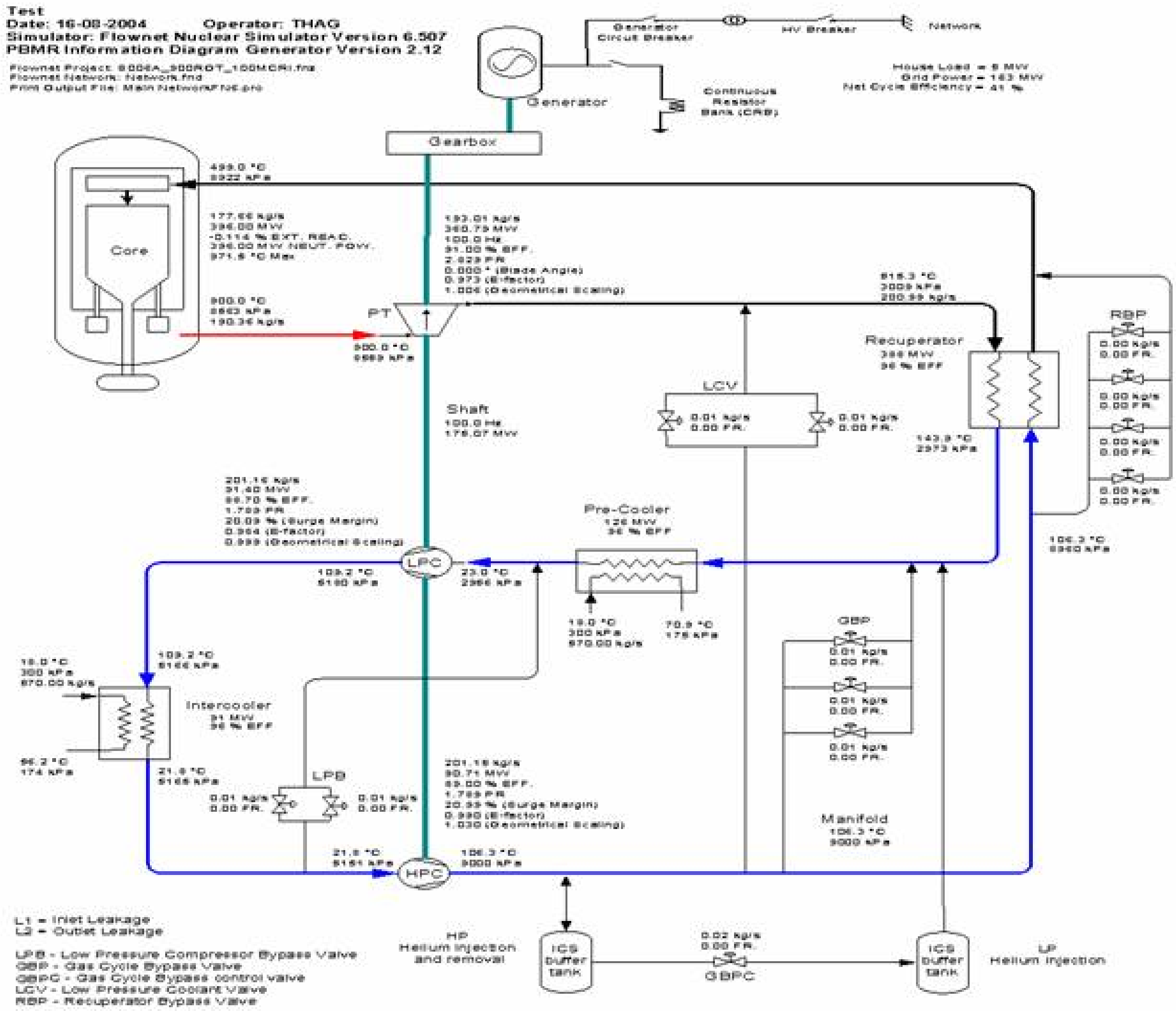

Figure 15. PBMR Thermodynamic Cycle. [12]

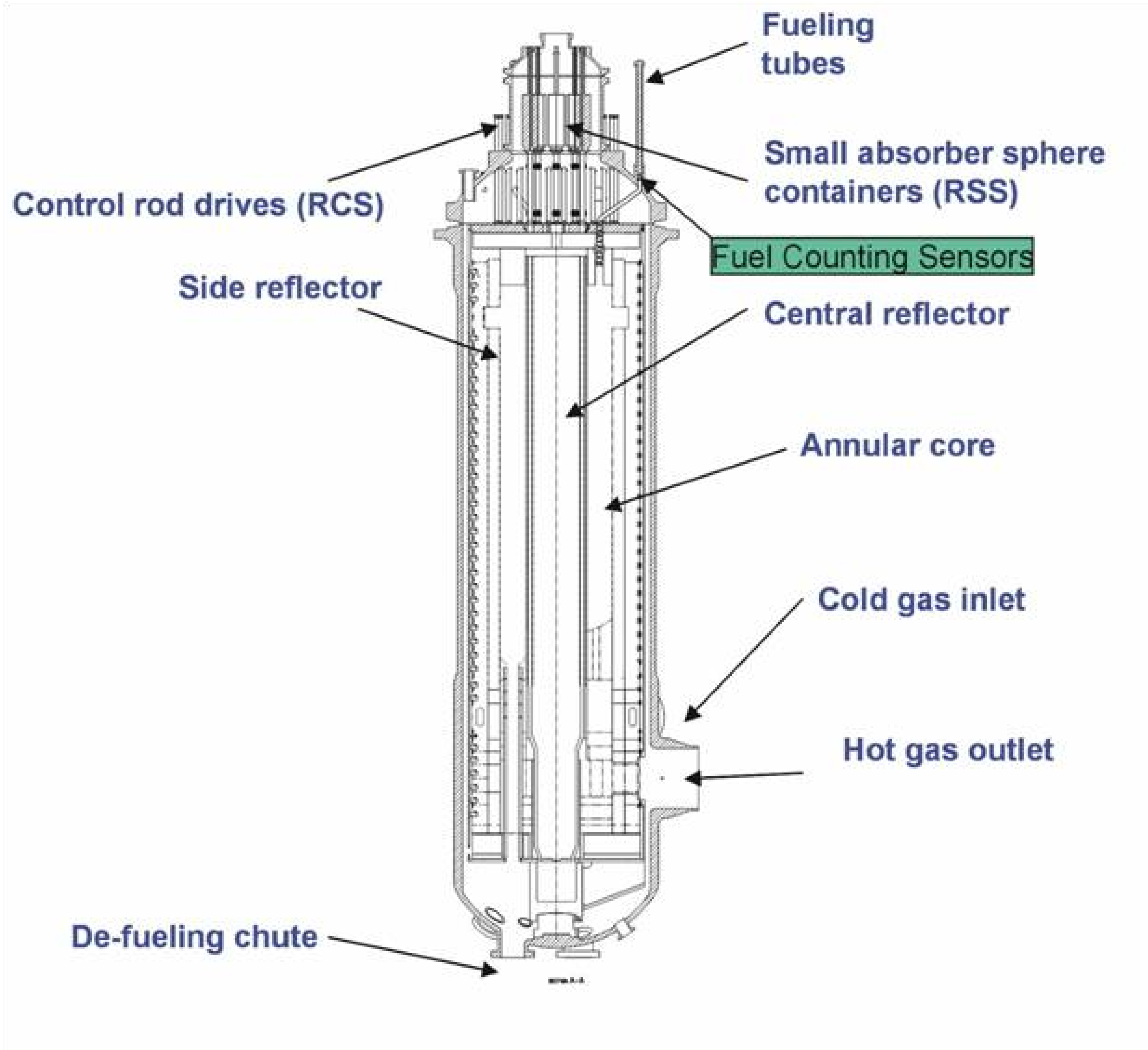

Figure 16. Reactor Unit Vessel Assembly.

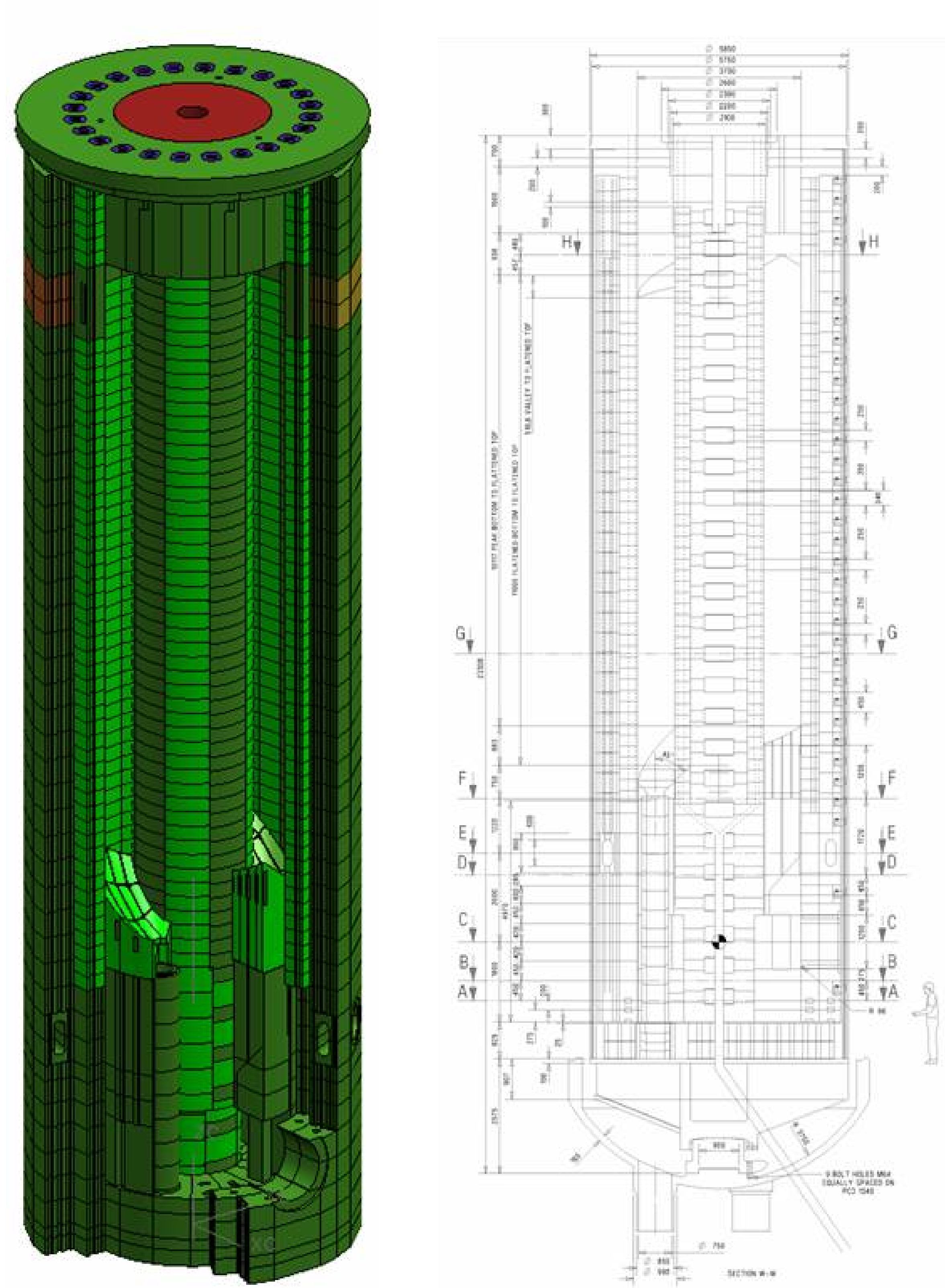

Figure 17. Core Structure Assembly. [14], [17]

2.2.2.8 Intermediate Heat Exchanger. A current NGNP requirement states that 10% of the heat from the primary loop must be able to be diverted to the production of hydrogen. The remaining 90% will be used to produce electricity. To accommodate this requirement, an IHX within a pressure vessel would be employed to divert heat from the primary side of the reactor to a hydrogen production plant. The heat exchanger needs to isolate hydrogen production plant equipment from the radioactive contaminants in the helium coolant and prevent any backflow of hydrogen or heat transfer fluids from the hydrogen plant back into the primary He loop. These cross contamination issues may be handled through equipment design, pressure differentials between the loops, or usage of secondary heat exchangers to ensure isolation. The primary circuit IHX may be employed in either a direct or indirect cycle application. The indirect cycle places the IHX and its pressure vessel directly between the reactor core and the PCU. The direct cycle diverts 10% of the reactor outlet gases to an IHX as a bypass around the gas turbine.

By definition, the IHX must handle the temperatures of the heated gases exiting the reactor core. The operational temperature of 850 °C (for the GT-MHR) is near the expected regulatory limits for the most heat resistant alloys available at this time. Accident situations may take the IHX beyond the realm of feasibility for a metallic material. For these reasons, depending on specific NGNP design, intermetallic or ceramic heat exchange components may have to be considered in the future. Although potentially feasible, such heat exchangers would require considerable development and extensive regulatory work to be licensed for reactor use in the primary system. In addition to the thermal requirements, the IHX must operate with some pressure differential and would need to be able to withstand the thermally induced stresses from expansion and contraction resulting from a loss of flow from either primary or secondary sides. The dimensional changes resulting from differential thermal expansion or contraction would be one of the major issues that would need to be overcome with use of intermetallic or ceramic heat exchange systems.

It is believed that due to the factors noted above, the IHX for the NGNP will be constructed of very high-temperature metallic materials and the structure will be designed to conform to ASME Code requirements. Therefore, the IHX will need to be controlled by the NGNP design to temperature, pressure and time exposure limits within the ASME Code capabilities of the structure. Therefore, systems will by necessity be in place that will allow the IHX to survive accident conditions.

2.2.2.8.1 Direct Cycle Application—The direct cycle application would require a compact heat exchanger sized for 10% of the reactor heat load to be placed inside or very close to the secondary vessel. The turbine inlet would have a small leg diverting primary coolant to the heat exchanger. The secondary side of the heat exchanger would contain coolant coming from the hydrogen production plant, probably from a secondary heat exchanger. The outlet of the primary side would re-enter the primary loop downstream of the turbine or in one of the turbine stages.

The primary side of the IHX (depending on design) could have 1000 °C helium flowing through it with an exit temperature of 900 °C or greater. The outlet of the secondary side of the IHX would be 950 °C to 975 °C with a secondary inlet side temperature of about 500 °C. To place a direct cycle IHX in the secondary vessel would require the surface area density to be greater than 1000 m^2/m^3. The pressure drop in each IHX leg must not exceed 2% of total pressure in each leg. The operating pressure in both legs is 6.7 to 7.1 MPa with the secondary pressure exceeding the primary pressure by 0.1MPa (if pressure is used to prevent primary gas flow into the secondary system). Depressurization of the secondary side while the primary remains hot and at pressure would create significant thermal stress within the IHX. Membrane stresses in the IHX will also be affected, but only as a function of the IHX design and the overall pressure differential. These off normal stress states may challenge the material properties at operating temperatures. The radiation fluence on the IHX is negligible inside or immediately adjacent to the secondary vessel. A pressurized core conduction cool-down event would push the average primary inlet temperature up to 1200°C for a short period. The full consequence of this temperature spike will have to be evaluated during design.

The primary advantages of the direct cycle IHX would be its relatively small size and the potential of incorporating the hardware within the secondary vessel. However, the direct cycle approach almost guarantees that a second heat exchanger will be needed to ensure isolation of primary system contamination and potentially allow change from He to a different operating fluid.

***2.2.2.8.2 Indirect Cycle*—**The indirect cycle application would require the IHX to be sized to handle the entire heat load of the reactor. The IHX is placed between the RPV and secondary vessel with structural ducts between the RPV and IHX and between the IHX and the main turbine/generator. The primary side of the IHX would see flow from the reactor and exit to the turbo-machinery pumps, intercoolers, pre-coolers and recuperator for conditioning the gas back to reactor inlet conditions. The IHX secondary side outlet helium would run the main turbine/generator. The secondary balance of plant would return the helium coolant to the secondary IHX inlet conditions using normal turbo machinery. The heat for the hydrogen plant would be drawn from the secondary outlet of the IHX upstream of the turbine. This configuration would isolate both the hydrogen plant and the main turbine/generator from the radioactive contamination in the primary leg. Such secondary vessel isolation is probably the biggest advantage of this approach.

The primary IHX inlet temperature could be up to 1000 °C and the secondary outlet temperature could be 950 °C-975 °C (depending on design). The pressure of the primary inlet would be nominally 7 MPa. The primary outlet and the secondary system pressures would need to be determined during design. The conditions for 95% efficiency would be equivalent to those discussed for the direct cycle. The fluence will be between that of the secondary vessel and RPV depending on the distance the IHX is placed from the RPV and the presence of shielding materials, though this could likely be reduced to an inconsequential level by engineering approaches, if warranted. If the secondary side experiences a loss of flow without scram, the entire primary side heat load is placed on the primary turbo-machinery, coolers, and recuperators to reduce the temperature and bring the pressure back up to inlet conditions. This IHX approach could also see the 1200°C spike in an off-normal condition.

Although an indirect cycle IHX would radically change the configuration of the systems external to the NGNP RPV, there are very good reasons to consider the approach. The major advantage of the indirect cycle IHX is that all the secondary vessel components are outside the primary loop. This difference would make the power conversion equipment easier to maintain and would limit the complexity of the equipment in the primary loop. Although difficult to quantify, this approach may also allow more efficient sizing of turbo machinery for processing the primary gas back to reactor conditions because both the primary and secondary systems would help handle the heat load in an accident situation.

***2.2.2.8.3 IHX Types*—**Three types of heat exchangers have been suggested for the IHX based on efficiency and potential feasibility: the printed circuit, the plate and fin, and intermetallics or ceramic open-cell heat foam. The more traditional, helical coiled tubes in a tube sheet design may also be feasible for the indirect cycle IHX. [18] Printed circuit heat exchangers (See Figure 18) rely on thermal diffusion welds between plates. The plate and fin type heat exchangers use high-temperature brazing to join the plates and seal the system. Intermetallics or ceramic heat exchangers hold potential for the NGNP, but the required unit would be several times larger than anything currently manufactured. Open cell graphite and intermetallic ceramic foam materials with exceptional thermal conductivities of up to 150 W/m K have been developed recently, though methodologies to use the thermal conductivity while retaining a pressure boundary must still be developed.

Each heat exchanger configuration has advantages and disadvantages to consider. Of the metallic systems, the printed circuit and plate-fin types allow the greatest surface area per unit volume of gas to minimize size. The tube type heat exchangers are more bulky and less efficient, but easier to build in a manner that can handle severe thermal transients. The pressure boundary for the printed circuit heat exchanger depends on the diffusion welds between every layer. The plate-fin pressure boundary depends on the high-temperature brazing material and the successful furnace brazing of an entire unit without any defects. In addition, the braze metal must take the stresses resulting from thermal transients. Intermetallic and ceramic heat exchangers may effectively eliminate the concerns of operating at high steady state temperatures but have major issues arising from their inherently brittle nature coupled with the need to handle thermal transients.

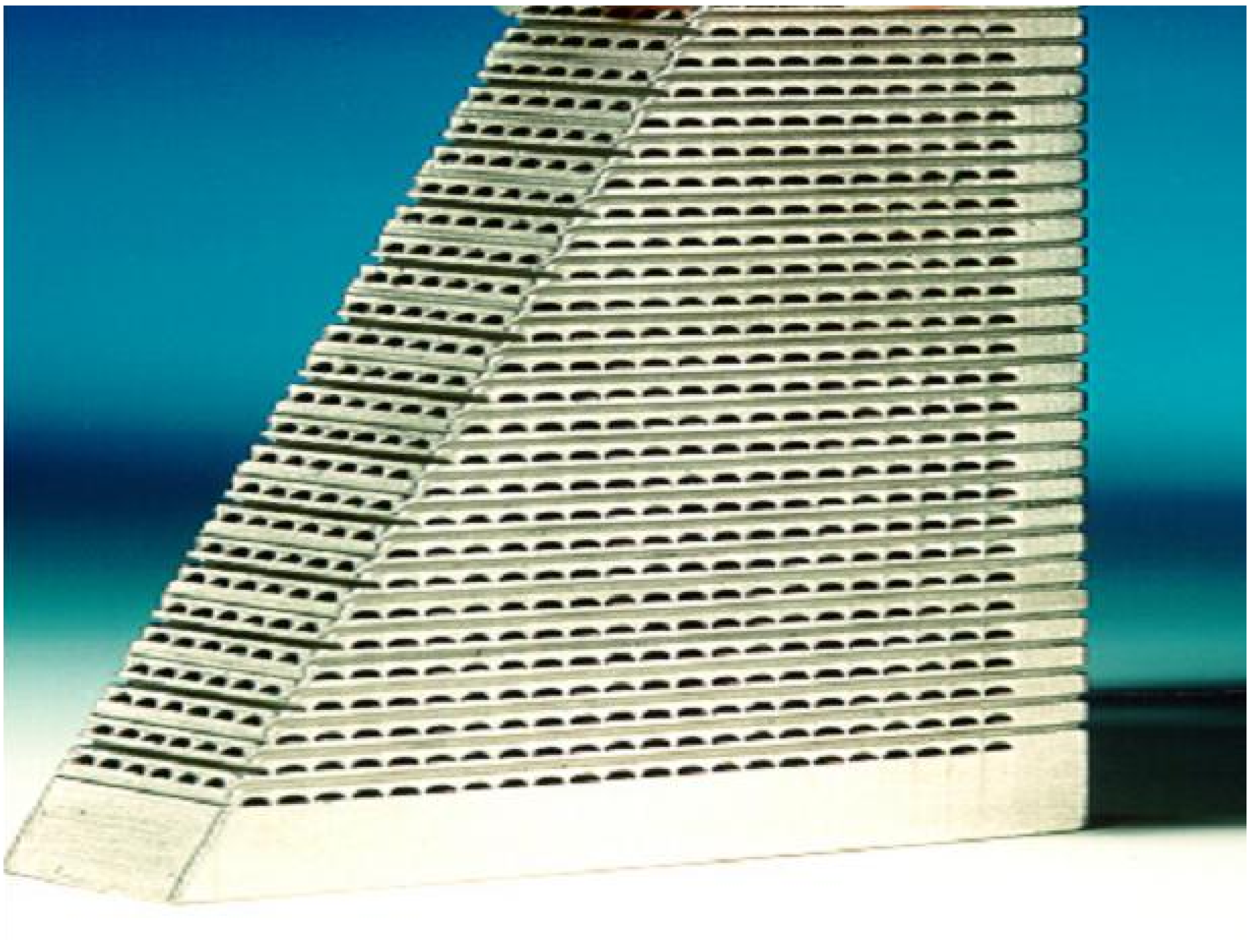

Figure 18. Printed circuit type heat exchanger.

2.3 Quality Assurance Requirements

All work performed to support the Technical Program for the NGNP Materials R&D Program will utilize the national consensus standard ASME NQA 1997, "QA Program Requirements for Nuclear Facilities Applications," and Subpart 4.2 of ASME NQA 2000, “Guidance on Graded Application of Quality Assurance (QA) for Nuclear-Related Research and Development,” for project-specific materials development activities.

The QA requirements for specific projects under the NGNP Materials R&D Program will be specified in project-specific Quality Plans and project-specific technical specifications. The project specific quality plans will include management controls commensurate with the project work scope and importance to the program. There are currently two project-specific quality plans – a plan at the Oak Ridge National Laboratory (ORNL) and a plan at INL; both of them are DOE National Laboratories. At the ORNL Laboratory the quality plan is entitled “Quality Assurance Plan for the NGNP Materials Program at Oak Ridge National Laboratory”, QAP-ORNL-NGNP-01, Rev. 1. The INL quality plan is entitled “Quality Program Plan for the INL NGNP Materials R&D Program # PLN-1792. The DOE Nuclear Energy Research Initiative university work activities may be managed under the umbrella of either the ORNL or INL Quality Plans mentioned above.

The material’s development effort has a broad scope and involves many different applications that fit within the scope of the ASME NQA-1 –1997 Standard or the ASME nuclear construction code. These unique application materials will require extensive R&D and specialized testing and qualification using various national or international methods and standards. This material development effort will require developing a common acceptable framework that will provide regulatory acceptable material with the appropriate certifications to the users.

Procurement actions required for development of materials shall be controlled as required to generate a solid qualification basis for potential future material procurements and installation and/or use related to pilot or production facilities.

There are key QA controls that are needed for these efforts and these shall be tailored and graded to provide the necessary level of assurance and documentation along with needed management controls to provide a material product that will be usable within the nuclear regulatory environment. Examples of these key QA controls are as follows:

Process Control- Experiments, tests, and material processing shall be performed to a written instruction outlining the required steps/actions. Results shall be documented in appropriate lab notebooks, travelers, run sheets or technical reports to provide processing information that is complete and available to the NGNP Program users.

Data Management-Define the methodology and industry being used to control, develop, qualify and store electronic information to be used in the materials qualification process. This would include existing information that may be derived from numerous sources over large spans of time or multiple individuals. This information may be electronic, hard copy technical papers and/or journal articles or existing material standards of material organizations.

Control of Purchased Materials, Items, and Services-These shall be considered if the quality of work results is dependant on the pedigree of materials, items or services and the assurance of conformance is documented. The specific name, manufacturer, chemical information, and model or serial number shall be documented for all materials and chemicals used in development processes that could relate directly to the quality of the product.

Software Management. The use of software in the materials development process shall be controlled by the individual users to assure accurate and reliable results that are obtained by established methods, qualified to known software management standard and documented, repeatable, and retrievable.

Management Reviews and Assessments- The use of peer/independent review activities shall be used during the initial stages of the NGNP Materials R&D Program and as the Program matures program management assessments shall also be performed to ensure the quality of the material program activities.

2.4 ASME Codification

Once appropriate materials have been designated for the NGNP use, it will be necessary to gain ASME Boiler and Pressure Vessel (B&PV) Code acceptance of those materials at the desired operating conditions. Since the ASME B&PV Code Section III establishes the rules for the construction of nuclear pressure vessels and many power plant components, ASME acceptance of the designated materials at the temperatures and environments expected during NGNP use is a significant programmatic step forward. The ASME B&PV Code is not the only national consensus standard that will have to be satisfied. However, the ASME B&PV Code is believed to be the most significant from the perspective of actually designing and constructing an NGNP demonstration plant.

It is important to note that once a material has been approved for use by the ASME Subcommittee on Materials under Section II, it is still necessary to interact with the other construction subcommittees, especially the Subcommittee on Nuclear Power, the Subcommittee on Design, and the Subcommittee on Pressure Vessels, to gain full ASME B&PV Code acceptance. Within the Subcommittee on Design, the Subgroup on Elevated Temperature Design works on issues affecting Section III Division I Subsection NH of the B&PV Code, which governs design of elevated-temperature Class 1 nuclear components.

2.4.1 ASME B&PV Code Background

The charter for the ASME B&PV Code committee is:

> To establish the rules of safety governing the design, fabrication, and inspection during construction of boilers and pressure vessels, and to interpret these rules when questions arise regarding their intent.

Many engineers consider the goal of the ASME B&PV Code is “to maintain the pressure boundary”. The ASME B&PV Committee has a large number of reporting subcommittees. The Subcommittee on Materials is responsible for Section II of the B&PV Code; the Subcommittee on Nuclear Power is responsible for Section III, Nuclear Power Components. The current B&PV Code has been in use for many decades and Section III has matured greatly over the last 30 years. However, it was written mainly for pressurized and boiling water reactors that typically have temperature limits below 427 °C. Section III, Division I, Subsection NH addresses temperature limits up to 816 °C but only for a limited number of materials. Due to the fact that the NGNP reactor is a helium-cooled reactor operating in excess of the temperature limits of 816 °C numerous areas of the ASME B&PV Code will need to be modified or expanded, including (but not limited to) new materials approved for use, new temperature limits for existing approved materials, new design considerations, new weld procedures, new equipment considerations, and more. Other sections of the ASME B&PV Code have temperature limits for certain materials in the 816 to 900 °C range but these temperature limits are still below what is desired for NGNP balance-of-plant and hydrogen generation uses. The current Section III Subsection NH criteria and material coverage originate largely from the liquid-metal reactor (LMR) program of the late 1960s, 1970s, and early 1980s. In the late 1960s the Atomic Energy Commission initiated a Materials and Structures Technology program and simultaneously asked the ASME B&PV Code Committee to charge an expanded subgroup on Elevated Temperature Design with developing the design rules that eventually provided the basis for Subsection NH. That subgroup was staffed largely with LMR program participants. A High-Temperature Structural Design Technology task within the nationwide Materials and Structures Technology program supported the development and experimental confirmation of design criteria to guard against creep, creep-fatigue, and ratcheting failures. The Mechanical Properties Design Data task provided the uniaxial data for design and quantification of the criteria. In companion efforts, the High-Temperature Structural Design Technology task provided recommended constitutive equations for the required inelastic design analyses, and the Design Data task provided the uniaxial stress-strain and creep data needed for designers to implement the equations. All of this work was based on experimental data from common heats of materials, so that the resulting design methods, criteria, and data were as consistent as possible. A recent Argonne National Laboratory report, prepared for the NRC provides a good overview of subsection NH and its associated cases and their shortcomings for HTGR components. [19]

2.4.2 Current ASME B&PV Code Material Acceptance Criteria

Section II of the ASME B&PV Code is considered a service subcommittee because it addresses materials approved for use by the construction subcommittees. Besides various specifications for ferrous, nonferrous, and welding materials, Section II also contains material properties such as Young's modulus, thermal conductivity, allowable stresses and stress intensities, etc. To achieve B&PV Code acceptance, specific material information must be provided to the appropriate subcommittees. Appendix 5 (contained in Part D of Section II of the ASME B&PV Code) lists the guidelines established for approval of new materials. The items in the Section II guidelines may require a significant effort to satisfy. The higher temperatures and operating environment for the NGNP may require even further efforts. Once the material is accepted it will then be used for construction approval in Section III, Subsection NH. The design rules and guidelines for Subsection NH are discussed in detail in Section 2.4.4 of this document.

2.4.3 ASME B&PV Code Process

The preferred approach to gaining ASME B&PV Code acceptance of NGNP materials and construction details is to incorporate the materials into Section II and write applicable rules for NGNP construction, where necessary. Below the level of subcommittee there are subgroups, working groups and special working groups. The subgroup, working group, and special working group report to the subcommittee where the details of the rules governing construction are proposed, debated, and approved. Participation on behalf of the NGNP Materials R&D Program at these levels is essential. The participation task is discussed in Section 3.5.

Other ASME subcommittee groups may be identified once further details of the plant design become finalized. Due to the lack of current high-temperature, gas-cooled reactors, new groups (e.g., under Section XI of the ASME B&PV Code) may need to be added to the organizational structure.

Code Cases are an alternative option for gaining ASME B&PV Code acceptance. Code Cases are used when it is necessary to clarify the intent of existing Code requirements or, when the need is urgent, to provide rules for material or construction issues not covered by existing Code rules. Code Cases are also useful since one can quickly determine the B&PV Committee's acceptance of an idea and the limited scope of a Code Case may gain easier NRC acceptance. However, a potential downside of a Code Case is its limited (3-year) effective period, unless it is renewed.

A subcommittee will consider a Code Case (3-year effective period) permitting use of new material provided:

- Evidence can be provided that a request for material specification coverage has been made
- The material is commercially available and can be purchased per proposed specification
  - Inquirer shows reasonable demand
  - Specification form clearly described
  - Requirements of Appendix 5 satisfied.

2.4.4 ASME B&PV Code Section III Subsection NH

The design rules of subsection NH for Class 1 elevated-temperature components consist of:

1. Load-controlled (primary) stress limits (Section III, Div I-NH Appendix I),

2. Strain, deformation, and fatigue limits (Section III, Div I-NH Appendix T).

The load-controlled stress limits are in the form of time-dependent allowable stresses based on both short-time tensile test results and long-term creep test results. Allowable stress reduction factors for weldments are given, as are reduction factors to account for the degrading effects of prior service. Only elastic analysis results are required to satisfy the primary stress limits.

The second category of design rules – strain, deformation, and fatigue limits – are much more problematic. These rules deal with complex behavior, resulting from primary plus cyclic secondary and peak stresses. They are aimed at preventing failures due to excessive deformation, creep-fatigue damage, and inelastic buckling, and they generally require inelastic design analysis results for their satisfaction. The rules include strain accumulation limits, creep-fatigue criteria[h], buckling limits, and special limits for welds. The materials that are currently covered, allowable life times, and maximum allowable temperatures are limited in Subsection NH, as shown in Table 3. Only the temperature limits for Alloy 800H come close to those required for the NGNP vessels. Coverage for none of the materials is adequate for the very high-temperature NGNP components.

Aside from the fact that most preliminary candidate NGNP materials are not included in Subsection NH, there are several generic shortcomings that will require resolution. First, the maximum temperatures permitted will have to be significantly increased. Second, allowable time-dependent stresses will have to be extended beyond the current 300,000 h maximum to 600,000 h. Third, environmental effects (impure helium) need to be incorporated into the failure criteria, particularly creep-fatigue.

2.4.5 Confirmatory Testing of Methodology

Time-dependent structural tests provide data that either validates the high-temperature design methodology (HTDM) or leads to changes in inelastic design analysis guidelines or Code rules. The role of structural tests will be even more important for the NGNP materials because of the lack of long-term service experience. The need for very-high-temperature, time-dependent tests of structural models was

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[h] As currently formulated in Subsection NH, the creep-fatigue rules are based on a linear damage accumulation rule, an interaction diagram to account for the synergistic effects (and for environmental effects in the case of ferritic steels), and multiaxial strength theories for both fatigue and creep rupture.

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.

It should be emphasized that the structural tests to be performed in this Materials Program 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 thermal and mechanical loadings. The tests are contrived to explore key features or problem areas of the methodology. Past examples include beams, plates; thick-walled cylinders subjected to thermal gradients, capped cylindrical shells, and nozzle attachments. Cylinders and plates with notch-like discontinuities and with axial or circumferential welds were included. The latter two types of tests will be particularly important to NGNP because of the two major NRC concerns of weldments and discontinuities.

While not strictly a part of the design methodology, the safety assessments required for licensing depend on much of the same materials and structures database. A particular need is for a flaw assessment procedure capable of reliably predicting crack-induced failures as well as the size and growth of the resulting opening in the pressure boundary. High-temperature flaw assessment guides have been developed in France, Japan, and the United Kingdom, and work on elements of a procedure is currently underway in the United States under Pressure Vessel Research Council sponsorship. An overall proven procedure, which will require inelastic analysis of flawed components, characterization of sub-critical creep and fatigue crack growth, and a structural failure criterion, does not exist however. These will be developed for the NGNP materials.

Experience has shown that once detailed design assessments are undertaken, shortcomings and issues with the design methodology and criteria will arise, requiring additional R&D for their resolution. In addition, the licensing process will likely result in the identification of further R&D requirements, as it did in the case of the Clinch River Breeder Reactor Project (CRBRP). Thus, it is anticipated that the HTDM project will continue throughout the design effort to resolve the shortcomings, issues, and regulatory concerns.

Table 3. Current Subsection NH materials and Maximum Allowable Times and Temperatures [a&c]

	Temperature (°C) b	
Material	**Primary stress limits and ratcheting rules**	**Fatigue curves**
304 stainless steel	816	704
316 stainless steel	816	704
2 1/4 Cr – 1 Mo steel	593b	593
Alloy 800 H	760	760
Modified 9 Cr – 1Mo steel (Grade 91)	593b	538

a. Allowable stresses extend to 300,000 h (34 years) unless otherwise noted.
b. Temperatures up to 649 °C are allowed for not more than 1000 h.
c. Alloy 718 is allowed up to a maximum temperature of 550 °C.

Four current ASME B&PV Code cases and a draft Code case are relevant to the HTDM project.

1. Case N-499 was developed for HTGRs. It permits Class 1 components fabricated from SA-533[16], Grade B steel to exceed the normal 371 °C low-temperature design limit for short periods for Levels B, C, and D events. A similar case might be developed for the NGNP vessel material under off-normal conditions.

2. Case N-201 provides rules for construction of core support structures made of ferritic steels, austenitic stainless steels, and high-nickel alloys, and having metal temperatures not exceeding those in Section II, Part D. This Case, with modifications, might be useful for the metallic core internals of NGNP. The basis for the Case is the same high-temperature structural design methodology as that on which Subsection NH is based.

3. Code case N-253 provides rules for Class 2 and 3 components for elevated temperature service. Unless exemption rules are met, the case essentially defaults to the criteria of Subsection NH.

4. Code case N-290, which covers expansion joints in Class 1 liquid-metal piping, can serve as a starting point for criteria and design methods for the NGNP bellows.

5. A draft Code case developed in the 1980s for design of Inconel 617 to 982 °C is directly pertinent to NGNP [20]. The original request for the case came from DOE and General Electric. The specific gas-cooled reactor component of primary interest was a steam-methane reformer, which was to be part of the reactor primary pressure boundary. Materials of potential interest included nickel alloys 800H, X, and Alloy 617. Alloy 617 was chosen for the case because it was a leading choice of designers, and a reasonable database of material properties existed. The case was developed by an ad hoc group of the Subgroup on Elevated Temperatures Design (SG-ETD). The case was subsequently approved by SG-ETD and submitted to its parent group, the Subcommittee on Design, for approval. However, further action on the case was suspended when the DOE project was canceled. The case is of value to NGNP because it can serve as a springboard for establishing NGNP Code rules. It was the result of a five-year effort of experienced high-temperature materials and structures engineers, as well as gas-cooled reactor project participants. It also had the participation and input of researchers from the Japanese Atomic Energy Research Institute (JAERI) and the Institute for Chemical Technology (KFA) in Germany. The draft case, while having the same framework as Subsection NH, has several unique features that are ramifications of the very-high-temperature material behavior. This behavior includes (1) the lack of clear distinction between time-independent and time-dependent behavior, (2) the high dependence of flow stress on strain rate, (3) softening with time, temperature and strain. Therefore, the design rules of Subsection NH that are based on the separation of time- and rate-independent response, or on strain-hardening idealizations of material behavior required careful reconsideration in the case. For example, the case specifies that inelastic design analyses for temperatures above 649 °C must be based on unified constitutive equations, which do not distinguish between time-independent plasticity and time-dependent creep.[i] The draft case also recognizes that significant environmental effects on Alloy 617 could exist, and it recognizes that extended exposure at elevated temperature may cause a significant reduction in fracture toughness of Alloy 617, thus introducing an additional failure mode – brittle fracture – to be considered. Finally, because of the uncertainties in data extrapolation and the lack of experience in designing to such high temperatures, where allowable stresses are very low, the draft case is limited to design lives of just 100,000 h or less.

Table 4 lists the ASME Code status and design allowable values for the candidate materials being considered for the reactor vessel and IHX and high-temperature components. As can be seen, there will be a lot of standards and code work required to have materials ready for the ASME B&PV Code Section III design process. For a list of potential candidate materials being considered for all internals, as well as the

[i] This is also the case for the high-alloy ferritics (e.g., 9Cr – 1Mo steel) at the upper end of their useful temperature range.

other high-temperature components likely to be constructed from metallic alloys, see Section 3.1.2, Table 6.

Table 4. ASME Code Status and Design Allowable Values.

Material/UNS	ASME Code Status	Code Maximum Temperature (°F)	Product Form	Stress Value (ksi)
800H/NO8810 (also ferritic steels, 304, 316)	Code Case N-201-4, (Class CS Components in Elevated Service, Section III, Div. I)	1400 °F (760 °C)		
602CA/NO6025	Code Case 2359-1 (Section I and Section VIII, Div.1)	1800 °F (982 °C)	Forgings, bar, plate sheet strip, welded and seamless pipe and tubing	0.32 (Section VIII only)
617/NO60617	Code Case 1982-1, (Section VIII, Div.1)	1800 °F (982 °C)		0.73 (creep-fatigue, thermal ratcheting and environmental effects must be considered)
617/NO60617	Code Case 1956-7, (Section VIII, Div.1) Annulled.	1650 °F (899 °C)	Plate, rod, bar, forgings and seamless tube	1.7 (creep-fatigue, thermal ratcheting and environmental effects must be considered)
617/NO60617	Draft Code Case, not completed, (Section III, Subsection NB, Class 1 Components, Subsection NH, Class 1 Components in Elevated Temperature Service)	1800 °F (982 °C)		
X/NO6002	Section II, Part D, Table 1B, (Section I, Section III, Class 2 and 3, Section VIII, Div. 1)	1650 °F (899 °C) (not all product forms—fittings, pipe, tube, plate, rod)	Forgings, bar, plate sheet strip, welded and seamless pipe, tubing, and fittings	1.2
XR-Mitsubishi Materials	Not currently in Code	Unknown	Unknown	Unknown
230/NO6230	Code Case 2384 (Section VIII, Div. 1)	1800 °F (982 °C)	Strip, plate, bar, welded pipe and tube, seamless pipe, tube, and fittings, forgings	0.45
HR-120/NO8120	Code Case 2315, (Section VIII, Div. 1)	1650 °F (899 °C)	Strip, plate, bar, welded pipe and tube, seamless pipe, tube, and fittings, forgings	1.4
9Cr-1MoVNb, Grade 91	Section III, Class 1	593 °C	Plate	4.3

3 TECHNICAL PROGRAM

The technical program is described in this section first by discussing the candidate materials for the components (Section 3.1), and then a breakdown of each of the specific projects (Sections 3.2 through 3.16).

3.1 Component Candidate Materials

A variety of options have been identified for potential use of materials in the NGNP reactor and balance of plant components. These options originated through an initial look at the materials issues for a very high-temperature reactor in January 2003[1] and a much larger, focused NGNP materials options identification activity that included meetings at INEEL and ORNL in July 2003. The information shown in this section is a summary of the options identified as a result of these activities and any others that have been identified since the July meeting.

3.1.1 Reactor Core Graphite, Reflector, and Supports

Graphite will be the major structural component and nuclear moderator in the NGNP core. The graphites used previously in the high-temperature gas reactor programs in the United States, H-451, are no longer in production and thus replacement graphites must be found. Hence, it will be necessary to qualify new grades of graphite for use in the NGNP. Fortunately, likely potential candidates currently exist, including fine-grained isotropic, molded or isostatically pressed, high-strength graphite suitable for core support structures, fuel elements and replaceable reactor components, as well as near isotropic, extruded, nuclear graphite suitable for the above-mentioned structures and for the large permanent reflector components. These candidates would meet the requirements of the draft ASTM materials specification for the Nuclear Grade Graphite.

The fine-grained isotropic, molded or isostatically pressed, high-strength graphite suitable for core support structure includes Carbone USA grade 2020 and Toyo Tanso grade IG-110. Toyo Tanso grade IG-110 was used in the Japanese HTTR for fuel blocks and in the Chinese HTR-10 PBR. These fine-grained materials are suitable for the fuel elements and replaceable reactor components.

Graphite is a complex material whose structure and properties reflect the raw materials used in its manufacture, the processing techniques, and the thermal history of the material. Our understanding of neutron irradiation damage in graphite is well developed. However, fundamental models relating structure at the micro- and macrostructural level to the irradiation behavior are less well developed.

Graphite comprises a composite structure manufactured from a filler coke and pitch binder. Nuclear graphites are usually manufactured from isotropic cokes (petroleum or coal-tar derived) and are formed in a manner to make them near-isotropic or isotropic materials. After baking (carbonization), the artifact is typically impregnated with a petroleum pitch and re-baked to densify the part. Impregnation and re-bake may occur several times to attain the required density. Graphitization typically occurs at temperatures >2500 °C. Additional halogen purification may be required. Typical manufacturing times are 6-9 months.

The forming and densification processes impart property variations within the billet. The properties will be somewhat different in the forming direction compared to the plane perpendicular to the forming direction. Moreover, a density gradient will exist from billet edge to center. These variations must be quantified for the selected grades of graphite. In addition, variations in properties will arise from billet to billet within a batch, and between production lots. Finished graphite is machined to the complex geometries required for the reactor components (fuel elements, reflector blocks, core support post, etc.).

Early in the program, it will be necessary to review and document the existing data, from all available sources, on the properties of these new grades of graphite. Irradiation data from ongoing experiments in the Petten Reactor (European Union program) will be of great value. A complete properties database on the new (available) candidate grades of graphite must be developed to support the design of graphite core components. Data is required for the physical, mechanical (including radiation-induced creep) and oxidation properties of graphites. Moreover, the data must be statistically sound and consider in-billet, between billets, and lot-to-lot variations of properties. The data will be needed to update and benchmark existing design models for graphite performance. Since the available near-isotropic, extruded graphites are somewhat similar to the prior grade H-451, design models for H-451 can be incrementally adjusted for the currently available graphites as new data becomes available. This review will provide data that will be input into the preliminary selection process.

As part of the preliminary selection process, a radiation effects database must be developed for the currently available graphite materials. As mentioned above, there is the potential to leverage data from European Union activities in the area of irradiation experiments on PBMR graphites (Petten Reactor irradiation experiments are currently being initiated). However, it is anticipated that a substantial number of additional graphite irradiation tests will be needed to complete the database. Since NGNP graphite service temperatures are anticipated to be as much as 200 °C greater than that in the Fort St. Vrain with H-451 graphite, additional data are required for all properties at these higher temperatures, including radiation damage effects. Therefore, in order to be qualified for the NGNP, existing graphite behavior models need to be modified based on sound materials physics and then validated/verified against new data for the currently available graphite grades. Property data must support the service conditions, including effects of higher temperature, helium gas (plus air and water), and neutron irradiation effects. Irradiation creep data for the candidate graphites must also be obtained.

New near isotropic, extruded, nuclear graphites have been developed in the United States and Europe for the South African PBMR. The new, currently available graphites are GrafTek (UCAR) grade PCEA—a petroleum coke graphite, and SGL Grade NBG-10, NGB-17 and NGB-18—a pitch coke graphite based on UAGR fuel sleeve graphite. These graphites may be candidates for the fuel elements and replaceable reactor components.

Graphites suitable for the large permanent reflector components are currently in production (e.g., SGL grade HLM or GrafTek [UCAR] grade PGX). Some data are available for these graphite grades. Grade PGX was used for the permanent reflector of the Japanese HTTR, also PGX and HLM were used in Fort St. Vrain for the core support and permanent reflectors respectively. Fine-grain, high strength, graphites are available from POCO Graphite, Inc. However, the available billet sizes are small and very expensive, thus not suited for NGNP core applications.

Near-isotropic, extruded, nuclear graphites (e.g., grade H-451 manufactured by SGL Carbon) were developed in the 1970s for large helium cooled reactors such as the Fort St. Vrain reactor. However, grade H-451 graphite has not been manufactured in the United States for more than 25 years.

There is a substantial database for U.S. standard grade qualified for nuclear service Grade H-451, including data for the effects of neutron irradiation on the properties, statistical variation of properties, oxidation behavior, etc. This body of data was considered sufficient to license the Fort St. Vrain reactor. Moreover, graphite behavior models were developed for Grade H-451 graphite. Fine-grained isotropic, molded or isostatically pressed, high strength graphites suitable for core support structure (e.g., Carbone USA grade 2020 or Toyo Tanso grade IG-110) are available today. These fine-grained materials are suitable for the fuel elements and replaceable reactor components, but they cost about three or four times more than fine-grain, near-isotropic, extruded graphite.

3.1.2 Reactor Internals

The components addressed in this subsection may be classified as core supports and core internals. They include:

- Core barrel
- Inside shroud
- Core support floor
- Upper core restraint
- Shutdown Cooling System (SCS) shell
- SCS tubes.

The following sections will develop additional data used to make materials choices for intermediate and high-temperature materials for the NGNP reactor internals and balance of plant components. Table 5 lists possible choices from an initial material scoping document[1] and additional information developed during reviews and discussions between materials experts at several meetings.

Depending on the specific component, the normal operating temperatures will range from 400 to 1000 °C. However, the maximum temperature estimated for accident conditions ranges could be higher from one component to another. In regard to loading, these components are not pressure boundary components, except for the SCS tubes. In some cases, however, the weight loads can be quite significant. This is the case for the core barrel. The fatigue, thermal-fatigue, seismic, and other loadings that could produce damage are largely unidentified at this time. Compatibility with the coolant gas is a requirement for core internals. In addition, radiation and thermal aging effects on properties are important considerations in material selection. Fabrication and joining must be considered. Finally, the materials must be ASME Code approved for the design conditions. Metallic core support structures must conform to ASME Sect. III, Div. 1, Subsection NG. Other core internals may conform to different rules. It is not clear whether the SCS tubes will be considered to be Class 1 or Class 2 components. At this point in time, it is best to assume that the materials of construction, regardless of the applicable subsection, will be limited to those listed in ASME Section II, Part D, Tables 2A, 2B, and 4. These tables cover temperatures to 370 °C for ferritic alloys and 425 °C for austenitic alloys. Subsection NH of Section III permits construction to higher temperatures for a limited number of materials. These are 2 1/4Cr-1Mo steel (Class 1), 304H stainless steel, 316H stainless steel, and Alloy 800H.

Table 5. Conditions Affecting Materials Selection for Intermediate and High-Temperature NGNP Components.

Condition / Cmpts	Loading	Environment Issues	Radiation Issues	Aging Issues	Joining Issues	Manufacturing Issues	Prime Materials
SCS Tube	Thermal Stress, LCF/HCF	Helium, Pressurized water, SCC,	Not significant	Some	Some	None	316FR Alloy 800H
Core Barrel	Core Weight	Helium	Negligible <0.005DPA	Some	Some	None	Alloy 800H 316FR
Core Support Floor	Core Weight	Helium	Negligible <0.005DPA	Some	Some	None	Alloy 800H 316FR
SCS Shell	Own Weight	Helium, Off Normal Helium	Negligible <0.005DPA	None, if CC composite	N/A, if CC composite	Major, if CC composite	CC composite Alloy 617 Alloy 230 Alloy HR-120 Alloy X, XR
Inside Shroud	Own Weight	Helium, Off Normal Helium	Negligible, Evaluate effect of Co	None, if CC composite	N/A, if CC composite	Major, if CC composite	CC composite Alloy 617 Alloy 230 Alloy HR-120 Alloy X, XR
Upper Core Restraint	Own Weight	Helium, Off Normal Helium	Negligible, Evaluate effect of Co	None, if CC composite	N/A, if CC composite	Major, if CC composite	CC composite Alloy 617 Alloy 230 Alloy HR-120 Alloy X, XR
IHX Indirect	Thermal Transients	Helium	None	Some	Some	Major	Alloy 617 Alloy 230 Alloy HR-120 Alloy X, XR
Hydrogen HX	7 MPa, Cycles	Helium, Heat transfer fluid	None	Some	Some	Some	Alloy 617 Alloy 230 Alloy HR-120 Alloy X, XR
Hot Duct	Own Weight	Helium	None	Some	Some	Some	Alloy 800H 316FR
Bellows	Fatigue	Helium	None	Some	Some	Major	Alloy 800H 316FR
He Circulator	Fatigue, Creep Fat.	Helium	None	Some	Some	Some	316FR Alloy 800H
Primary to Secondary Piping	7 MPa	Helium, Heat transfer fluid	None	Some	Some	Major	Alloy 617 Alloy 230 Alloy HR-120 Alloy X, XR
Recuperator			None	Some	Some	Some	347 SS 316FR

Potential candidate materials for the internals, as well as the other high-temperature components likely to be constructed from metallic alloys, are listed in Table 6. These materials include alloys for which significant databases exist and new state-of-the-art alloys which are being developed for other high-temperature applications. ASME Code status and design allowable values for the subset of candidate materials being considered for the reactor vessel and intermediate and high-temperature components are listed in Table 4.

For very-high-temperature components (>760 °C), the most likely material candidates are:

- Variants or restricted chemistry versions of Alloy 617
- Variants of Alloy 800H
- Alloy X and XR
- Alloy 602CA
- Alloy 230

Alloy 617, Alloy X and Alloy XR were developed for earlier, gas-cooled reactor projects. Alloy 617 has the significant advantage in the United States of having gone through ASME Code deliberations that culminated in the draft Code case, and the body of experts that developed the case simultaneously identified what must be done before the Code case could be approved. Alloy 800H is in Subsection NH, and would be the leading candidate for the intermediate temperature range of 600-760 °C. Alloy X and XR have a significant database and body of experience in Japan. Alloy 602CA is a relatively new high-temperature alloy that has been approved for Section VIII, Division I construction to 1800 °F [2, 21] Alloy 230 has good high-temperature and environmental resistance properties and is approved for Section VIII, Division I Construction to 1650 °F[8].

However, the upper limit of these materials is judged to be 1000 °C. Any component that could experience excursions above 1000 °C would need greater very high temperature strength and corrosion resistance capabilities. C_f/C composites would then be the leading choices for materials available in the near future for service that might experience temperature excursions up to 1200 °C. For similar high-temperature service at some later point in the future, oxide dispersion strengthened (ODS) alloys could be an alternative. SiC_f/SiC composites are also being considered for certain applications.

Table 6. Potential Candidate Materials Selection for Intermediate and High-Temperature Metallic NGNP Components

Nominal Composition	UNS No.	Common Name	Existing Data Max Temp (°C)	Helium Experience
Ni-16Cr-3Fe-4.5Al-Y		Haynes 214	1040	
63Ni-25Cr-9.5Fe-2.1Al	N06025	VDM 602CA	1200	
Ni-25Cr-20Co-Cb-Ti-Al		Inconel 740	815	
60Ni-22Cr-9Mo-3.5Cb	N06625	Inconel 625		
53Ni-22Cr-14W-Co-Fe-Mo	N06230	Haynes 230	1100	
Ni-22Cr-9Mo-18Fe	N06002	Hastelloy X	1000	Yes
Ni-22Cr-9Mo-18Fe		Hastelloy XR	1000	Yes
46Ni-27Cr-23Fe-2.75Si	N06095	Nicrofer 45		
45Ni-22Cr-12Co-9Mo	N06617	Inconel 617	1100	Yes
Ni-23Cr-6W		Inconel 618E	1000	
Ni-33Fe-25Cr	N08120	HR-120	930	
35Ni-19Cr-1 1/4Si	N08330	RA330		
33Ni-42Fe-21Cr	N08810	Incoloy 800	1100	Yes
33Ni-42Fe-21Cr	N08811	800HT	1100	
21Ni-30Fe-22Cr-18Co-3Mo-3W	R30566	Haynes 556	1040	
18Cr-8Ni	S30409	304H SS	870	Yes
16Cr-12Ni-2Mo	S31609	316H SS	870	Yes
16Cr-12Ni-2Mo		316FR	700	
18Cr-10Ni-Cb	S34709	347H SS	870	
18Cr-10Ni-Cb		347HFG	760	
18Cr-9Ni-3Cu-Cb-N		Super 304	1000	
15Cr-15Ni-6MnCb-Mo-V	S21500	Esshete 1250	900	
20Cr-25Ni-Cb		NF 709	1000	
23Cr-11.5Ni-N-B-Ce		NAR-AH-4	1000	
Ni-20Cr-Al-Ti-Y2O3	NO7754	Inconel MA 754	1093	
Ni-30Cr-Al-Ti-Y2O3		Inconel MA 754	1093	
Fe-20Cr-4.5Al-Y2O3	S67956	Incoloy MA956	1100	

For service in the temperature range of 600 to 760 °C, alloy 800H appears to be a leading candidate. A restricted chemistry version of 800H, namely alloy 800HT, is considered, as well. Alternative alloys to 800H exist, but they have little experience in nuclear systems at temperatures above 600 °C.

For temperatures at 600 °C and below, a wide choice of materials is available. Those alloys contained in ASME Sect. III, Subsection NH are leading candidates. Also, 9Cr-1Mo-V steel is in the final stages of acceptance into Subsection NH. An alternative low carbon 316 stainless steel (316FR) is considered to be a strong candidate since the steel could achieve Code approval with less effort than other alternatives.

Compatibility with the helium coolant and irradiation resistance of the potential candidate materials needs to be addressed. The experience base that exists must be evaluated for the different alloys regarding temperatures, fluences, and environments and/or expectations based upon what type of data or models must be determined.

3.1.3 Primary Coolant Pressure Boundary System

Possible primary coolant pressure boundary Systems envisioned for the NGNP are illustrated in Section 2. It will comprise a large RPV containing the core and internals (a second vessel containing an IHX and circulator or a PCU) and a pressure-containing CV joining the two vessels. Because of the wide range of material thicknesses in the Primary coolant pressure boundary system, it will be constructed in a segmented configuration. Although the specific design is not yet available, such a configuration will play a role in the materials selection as it relates to fabrication issues, effects of loading variables such as cycling, etc. The three 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 7.

The advanced materials tentatively selected for further investigation for the gas-cooled Primary coolant pressure boundary system service are ferritic/martensitic steels, alloyed primarily with chromium and molybdenum. The preliminary four classes of steels are:

- 9Cr-1MoVNb
- 7-9Cr2WV
- 3Cr-3WV
- 12Cr-1MoWV.

Table 7. Primary Coolant Pressure Boundary System operating conditions affecting candidate material selection for the NGNP based on GT-MHR design

Component	Normal VHTR System Operating Conditions: Temp. [ºC] Pressure [MPa]	Neutron Fluence, E>0.1 MeV (dpa)	Abnormal Conditions	Estimated Component Size
Reactor Pressure Vessel (RPV)* [6], j	300-500 ºC k [7.4-8.0 MPa]	1x1019 n/cm-2 per 60 years (0.077 dpa)	≈560 ºC at 1 atm for 200 hours	Diameter: >9m, Thickness: 100-300mm, Height: >24m
Cross Vessel (CV) [6], j	300-500 ºC [7.4-8.0 MPa]	1x1019 n/cm-2 per 60 years (0.077 dpa)	300 to 560 ºC for 200 h [7.4-8.0 MPa]	Diameter: >2.5m, Thickness: >100mm Length: 4-5m
Secondary Vessel [22]	300 ºC [5.0-6.0 MPa]	Negligible 3x1014 n/cm-2 per 60 years	300 ºC [5.0-6.0 MPa]	Diameter: 7-9m, Thickness: 100-200mm Height: .35m
Closure Bolting [6], j	550 ºC	1x1019 n/cm-2 per 60 years (0.077 dpa)	≈560 ºC at 1 atm.	

The currently estimated maximum normal operating temperature of 300-500 ℃ for the RPV and CV is in the creep range for any ferritic or ferritic-martensitic steel currently in any part of the ASME B&PV Code, while the maximum abnormal (off-normal accident) temperature of 560 ℃ for 200 hours is approaching the limit in the Code and provides an even greater challenge. For the ferritic steel option, there are four classes of advanced, higher alloy ferritic-martensitic steels that have been identified as potential candidate alloys, while the 2 1/4Cr-1Mo alloy is listed especially for the lower temperature design. These five alloy classes are listed in the order recommended as priority for consideration as the structural material for the primary coolant system components for the NGNP. Additionally, the class of austenitic stainless steels is listed as well, as a fallback option, but an option that retains the potential for operation at the desired temperatures, especially considering the abnormal temperatures under accident conditions, albeit at a significantly higher capital cost.

1. Class of 9Cr-1MoVNb

    a. This class of materials has the most industrially mature high strength database. For example, the 9Cr-1Mo-V (Grade 91) alloy is ASME Code approved to 649 ℃ for Section III, Classes 2 and 3 components and is in the final stages of approval for inclusion in Subsection NH for Class 1 applications.

    b. There are, of course, limits to Code applicability involving time at temperature, thickness of forgings, etc.

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[j] Core Barrel, dpa {(7.584E-7+1.548E-6+1.217E-6) x 365 x 60 = .077dpa}

[k] A cooling system design and insulation will affect this range.

c. Within this class of alloys, it seems prudent to consider variants such as 9Cr-1MoWV (Grade 911), (Grade 92), etc., because available research data show significantly improved high-temperature strength for those alloys relative to Grade 91.

2. Class of 7-9Cr2WV

   a. Various alloys of this class are currently being developed under the Fusion Materials Program.

   b. There is a smaller database than for the 1st class mentioned above, but some of these alloys offer the possibility of better high strength properties.

   c. Examples of specific alloys within this group include F82H (7.5Cr2WV), JLF1 and EUROFER (9Cr2WV).

   d. A potential advantage of these alloys is the fact that they have also been developed to have reduced activation under neutron irradiation with resultant advantages for decommissioning.

3. Class of 3Cr-3WV

   a. This class of alloys offers good high strength properties, but is one of the newer alloys under development and, as a result, has a very limited database. In relatively modest section sizes evaluated to date, the yield strength of the specific 3Cr3WV alloy under development at ORNL is about twice that of the SA508[15] Grade 3 forging steel used for current LWR RPVs.

   b. Because of its lower alloying content, it offers the potential for substantially lower cost than those more highly alloyed steels in the two classes discussed above. However, because of its lower alloying content, environmental effects at high temperatures may be limiting.

   c. There are indications that this alloy offers the possibility of no need for a post-weld heat treatment.

   d. One other alloy in this class is a 2.75Cr-1MoV variant under development in Russia.

4. Class of 12Cr-1MoWV

   a. The alloy designated HT9 is an older existing alloy within this class of materials.

   b. The HT9 alloy has a broad database available, but is has poorer properties than, e.g., 9Cr-1MoVNb.

   c. There are some more recent 12Cr variants that offer improved properties relative to the HT9. For example, the HCM 12A alloy has a good database and is currently approved by ASME Code Case 2180 to 649 °C for application in Sections I and VIII. Additionally, a Japanese alloy designated SAVE12 appears to have good high-temperature strength, but the available database needs to be reviewed.

5. Fallback for lower temperature operation: 2.25Cr-1Mo

    a. Of course, there is an extensive database for this alloy, including data in different operating environments such as helium.

    b. Another advantage is the extensive industrial experience with this alloy in many different applications around the world.

    c. However, its high-temperature strength is significantly lower than the alloy classes discussed above and, as such, is only applicable for substantially lower vessel temperature, such as in the case of the HTTR at JAERI.

6. Class of austenitic stainless steels (Types 304, 316, etc.)

    a. There is an extensive database for many of these alloys, including some data in helium with various impurity contents.

    b. There is extensive industrial experience with this class of steels in many different applications, including in irradiation environments.

    c. The tensile strengths of these alloys are much inferior to the ferritic-martensitic steels, but their strength properties do not degrade as rapidly at high temperatures. However, at temperatures in the range of 650 °C, their maximum allowable stresses are not necessarily superior to some ferritic-martensitic steels. The primary reason for inclusion of the class of stainless steels here is their metallurgical stability at the higher temperatures currently anticipated for the abnormal conditions.

    d. In general, stainless steels have superior oxidation and corrosion resistance in many media, but they are not immune to severe degradation in some common environments.

The pressure vessels must consistently resist the internal gas pressure of the primary system without damage and significant deterioration resulting from the operating temperature and radiation environment. Pressure vessel steels that may potentially be used for the NGNP are primarily dictated by their capability to operate at elevated temperature without structural changes or creep damage. Acceptable alloys must also be fully ASME B&PV Code Section III approved because they constitute the primary pressure boundary for the reactor. Because some reactor systems incorporate special cooling systems for the RPV, there is a significant range in potential materials that might be used in the RPV application. The PBMR usage of SA508[15]/SA533[16] steels is predicated on their ability to design and operate a cooling system for the RPV. The GT-MHR approach assumes that no such cooling system is available, which results in designation of more expensive, higher temperature alloys such as modified 9Cr-1Mo for the RPV. This approach is, however, under review [7].

Also of concern in accident situations are metallurgical changes to the alloy itself that may occur as a function of time, temperature, and radiation. Grain boundary precipitation reactions and grain coarsening are probably of greatest concern as they reduce the mechanical, fatigue, creep, and impact properties of the steel and there are no reasonable methods to recover from the changes while the vessel is in place.

The physical size of the RPV results in a number of other materials considerations that cannot be ignored. Only a few companies in the world (none in the United States) could manufacture ring forgings of appropriate size and quality for the NGNP using even standard pressure vessel alloys. The high-temperature alloys, such as modified 9Cr-1Mo, would require significantly higher forging temperatures and pressures that may be beyond any existing capacity. Thick section welding is also a major consideration for the RPV where cross section thicknesses up to 300 mm are involved and weld filler metals providing properties equivalent to the base metal may be difficult or impossible to achieve with the alloy content of most specialized high-temperature steels without subsequent heat treatments.

The reactor CV has metallurgical concerns similar to the RPV with similar requirements to sustain elevated temperatures in an accident situation. Although the CV is not large enough to result in major manufacturability issues, it must be made from a material that is metallurgically compatible with both the RPV and secondary vessel such that it can be welded to both and provide a fully qualified pressure boundary with the stresses and thermal expansion/contraction issues involved during reactor transients and shutdown.

Recent conceptual design for the NGNP RPV provides for an operating temperature similar to that of the GT-MHR. For both cases, however, there are uncertainties regarding the abnormal temperatures and times, loads, load-time history, time-temperature-load histories, and the temperature and neutron flux gradients through the RPV walls. The current estimate for temperature gradient through the RPV wall is about 50 °C. Additionally, the current estimate for fatigue cycles is for about 150 cycles plus hydrogen cycles for a total of about 600 small cycles. It is recognized that the normal operating temperatures for the primary coolant pressure boundary system are dependent on the capabilities of the materials of construction. Thus, an iterative approach will be required to eventually match the limiting material capabilities and the design operating conditions.

Potential candidate alloys for the secondary pressure vessel could include those for the RPV and CV, but there are lower cost options available because of the lower operating temperatures. Even under abnormal conditions, the secondary pressure vessel will be subjected to temperatures about the same as those currently used for commercial LWR vessels. The secondary pressure vessel should not be a major development issue from the viewpoint of thermal or pressure requirements but will be the most challenging from the viewpoint of its physical size.

Thus, the current LWR pressure vessel materials, SA508 [15] grade 3 class 1 forgings or SA533 [16] grade B class 1 plates are potential candidates, as is the 2 1/4Cr-1 Mo alloy, dependent on material compatibility issues. It is noted that the CV is welded to the secondary vessel and the welded joint with dissimilar materials must be a consideration.

Potential candidate alloys for high-temperature closure bolting are Alloy 718 and Types 304 and 316 stainless steels. Although Alloy 718 has superior strength, it is currently approved up to 550 °C in ASME Section III, Subsection NH. It is believed that this limit is based on the standard double aging heat treatment normally used for this alloy. A more recently developed “direct-aged” heat treatment may increase the approved ASME temperature limit noted. The two types of stainless steels, however, have allowable stress intensities for bolting up to 704 °C. An evaluation of the database for the Alloy 718 will be conducted to assess the data needed, if any, for increasing the allowable temperature to that required for the NGNP. Also, the estimated irradiation exposure for closure bolting will be assessed to evaluate the need for inclusion of bolting in the irradiation program.

3.1.4 Control Rod and Composite Structures

A number of structural composites were identified for use in control rods and other composite structural applications in the reactor. The components and potential materials are shown in Table 8. A C_f/C composite material comprises a carbon or graphite matrix that has been reinforced with carbon or graphite fibers.

Table 8. Potential Structural Composite Applications.

	Graphite	**C_f-C**	**SiC_f-SiC**
Hot Duct		X	X
Core Support Pedestal	X		
Fuel Blocks	X		
Replaceable Outer/Inner Reflector Blocks	X		
Top/Bottom Insulation Blocks	X		
Upper Plenum Block	X		
Floor Block	X	X	X
Upper Core Restraint & Upper Plenum Shroud (Structural Liner & Insulation)		X	X
Control Rods and Guides		X	X

Composites of either carbon fiber/carbon (C_f/C) or silicon carbide fiber/silcon carbide (SiC_f/SiC) could be potentially used to fabricate several different components. Future qualification tests will be required to delineate which of the composites are the best choices for a given component based upon the response of the composite to exposure conditions expected within the reactor. For simplicity, C_f/SiC composites were not included in the table, but were considered to be an intermediate between C_f/C and SiC_f/SiC composites. The C_f/SiC composites will be lower in cost than SiC_f/SiC composites, but might exhibit cracking problems due to the use of dissimilar materials. The C_f/SiC composites were classified as a subcategory of SiC_f/SiC and would require the same qualification tests as SiC_f/SiC.

A preliminary list of selection factors for the previously identified types of ceramic composites is shown in Table 9. The use of C_f/C composites appears to be desirable for many applications within the reactor because of their strength retention at high temperatures. For example, C_f/C is a top candidate for the control rod sheath or guide tubes for a prismatic NGNP because metallic materials cannot withstand the level of neutron irradiation and high temperature of 1050 °C or higher found in the core.

Ceramic composites made from silicon carbide fibers and silicon carbide matrices (SiC_f/SiC) are promising for nuclear applications because of the excellent radiation resistance of the β phase of SiC and their excellent high-temperature fracture, creep, corrosion and thermal shock resistance. In addition, there is some evidence that SiC_f/SiC composites have the potential to be lifetime components (no change-out required) within the high radiation environment within the core. Unfortunately, these SiC_f/SiC composites have not been as well characterized as C_f/C composites, so there is more uncertainty in the applicability. Therefore, it will be necessary to carefully evaluate both C_f/C and SiC_f/SiC for the control rod material.

Table 9. Materials Pro/Con Analysis.

Pros	Cons
SiC_f/SiC Composites	
Good oxidation resistance	Higher cost than C_f/C
Stronger than C_f/C	Many have Boron coated interfaces
Greater radiation damage resistance than C_f/C	Free silicon (not desired)
Less change-out, lasts longer	Lack of manufacturing/infrastructure
	Qualification – different weaves require a new qualification. ASME specification issue.
C_f/C Composites (Note: Replacement for super alloys. Could be used for guide tubes [~10 feet long, telescope feature], the Upper Core Restraint structure, and other internals where temperatures are too high for metallics)	
Good material for accident situation.	More Radiation damage/shrinkage than SiC_f/SiC.
Flaking is less likely than SiC-SiC.	Qualification – different weaves require a new qualification. ASME specification issue.
Eliminates metal from the core.	Lack of design criteria.
Good Residual properties (e.g., strength). Strength and fracture resistance is greater than graphite.	
C_f/SiC Composites	
Higher thermal conductivity than SiC_f/SiC	Possible Radiation damage
Higher strength	Qualification – different weaves require a new qualification. ASME specification issue.
Higher moderating power	

3.1.5 Intermediate Heat Exchanger and Piping

For the Indirect Power Generation Cycle, the reactor coolant outlet temperature could be as high as 1000 °C. Materials for the IHX will need to come from the list in Table 6. The reactor coolant system pressure is 7 MPa, but the difference from the primary to secondary circuit may be small (0.1 MPa) if He is used and the IHX will be contained within a pressure vessel. The leading IHX design for this cycle is a compact counter-flow configuration that involves channels passing through diffusion-bonded metallic plates. Transient thermal loadings could be a problem but the details needed to identify the materials performance requirements will depend on the design that will be selected. Environmentally induced degradation of the metals from impurities in the helium or flow induced erosion is a concern. Aging effects are a concern for very long-time thermal exposure since embrittlement could affect the performance of the IHX during thermal transients. Welding/brazing and fabrication issues exist that will depend on the IHX design details. Again, the leading potential candidates for service up to 1000 °C are Alloy 617, Alloy 230, and Alloy XR. Other nickel-base alloys such as Alloy 740, and Alloy 602CA will be considered. There is a possibility that the compact IHX could be fabricated from a C_f/C composite.

Alternate IHX designs such as tube-and-shell introduce concerns that can only be addressed when more is known about the performance requirements. The operating temperature and environment for the indirect power generation cycle are not likely to change. Rather, the loading conditions will require a database that is extended to a broader range of design criteria than the reference compact IHX configuration. Except for the fact that the tube and shell IHX would be helium to helium, the design and associated materials issues might be similar to the heat exchangers already evaluated in the German and Japanese gas-cooled programs.

The hydrogen plant HX may operate at 850 to 950 °C and will experience an operating pressure as high as 7 MPa. Details of the HX are unknown. The operating pressure and corrosion potential of heat transfer fluid to the hydrogen plant are unknown and these will have influence on the choice of materials for the HX. The HX designs will likely result in only about 10% of the power being diverted to the hydrogen plant and the remaining 90% to the turbine. It seems likely that thermal stresses and expansion loads will be a concern in the HX if it is a tube and shell design. A compact unit similar to the IHX will also present problems with respect to fabrication and inspection. A design methodology is needed for this relatively complex structure and characterization of materials will be an essential element of this technology. Again, the materials of construction are subject to environmental and aging-induced degradation. In addition to corrosion effects of the impurities in the helium, there may are concerns about corrosion effects or even mass transfer in HX should molten salts be used for heat transfer in the hydrogen plant. These issues cannot be adequately assessed until a more mature design of the IHX evolves.

3.1.5.1 Piping. The primary-to-secondary piping may operate at 950 °C to 1000 °C and will experience an operating pressure as high as 7 MPa. Creep-type conditions will prevail. Further, thermal stresses and expansion loads are always a concern in such piping systems. Again, the materials in all components are subject to environmental and aging-induced degradation.

3.1.5.2 Intermediate Heat Exchanger Pressure Vessel. The IHX pressure vessel is one of the least well-defined components in the NGNP system. To minimize the technical issues surrounding the IHX itself, the inclusion of an external pressure vessel is expected to be necessary. This vessel would operate at a lower temperature than the IHX internals and maintain a positive pressure close to the reactor primary system to ensure that the IHX does not experience a large pressure differential that would dramatically reduce its lifetime. Pressure vessel materials choices would be limited to the materials in Table 6 above for an uninsulated vessel, but even preliminary selection will not be feasible until further pre-conceptual design information is developed to provide requirements for such a vessel. If the vessel were internally insulated to substantially reduce the wall temperature, then the materials included in Section 3.1.3 on RPVs could also be considered.

3.1.5.3 Hot Duct Liner and Insulations. NGNP insulation will include both structural ceramics of low thermal conductivity (typically designed to be stressed in compression, since ceramics exhibit high compressive yield strengths) and low-density ceramics (e.g., foams or fibers) that will provide excellent thermal insulation. There are many design concepts available to achieve insulation. For example, a meter of graphite (K_{th}> 10 W/m-K) thickness plus 0.2 meter of C_f/C composite blocks is sufficient to insulate the lower metallic core support structure from the core outlet gas. However, where room is limited to a few inches of insulation thickness to do the same job, a more efficient form of insulation is required. A suitable insulation system, where significant structural support is not required, is to sandwich Al_2O_3 - SiO_2 mixed ceramic fiber mats (K_{th}<0.1 W/m-K) between metallic cover plates that are attached to the primary structure. Figure 19 illustrates the basic principle of this type of insulation as applied to the hot gas duct of the GT-MHR design.

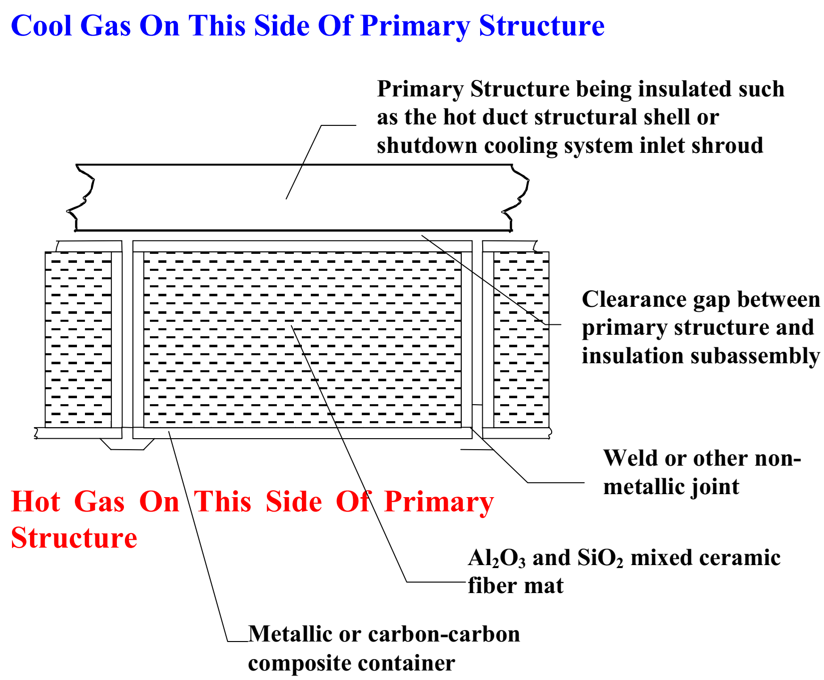


Figure 19. Thermal insulation system for the GT-MHR (part of Figure 8).

Structural ceramics (excluding ceramic matrix composites) probably will be monolithic (fabricated as single large pieces), being of high density and creep resistance[23] (with low levels of chemical impurities, moderate grain sizes and low applied stresses). These large pieces might be engineered as interlocking blocks to provide lateral structural stability. Examples of candidate ceramics compositions for the blocks are high-purity alumina[24] and stoichiometric mullite[25]. In one reactor design (prismatic), an example is the "Floor Blocks", where the blocks support the considerable weight of reactor core materials above them. Typical operating parameters for components suitable for monolithic ceramics are given in Table 10.

Table 10. Operating conditions for monolithic thermal insulators

	Operating Temp. (°C)	Maximum Temp. (°C)	Lifetime Neutron Dose (dpa/30yr)
Ceramic Floor Block	600	600	0.001
Top Insulator Block	700	1100	0.0003
Bottom Insulator Block	1050	600	0.0003

Ceramic insulating materials could be in the form of bricks, easily poured powders or "castable cements" containing voids or hollow spheres, sheets (e.g., Grafoil)[26], or fibers (loose fibers, woven fibers, or fiber "blankets"). Fiber blankets would provide ease of installation, durability, and low thermal conductivity. Grafoil™ is a sheet form of graphite that has low thermal conductivity in the direction perpendicular to the sheet and can be used as high-temperature gaskets. Examples of fibers include Kaowool[27] (representative of a family of mullite compositions in many different purities and physical forms), alumina, zirconia, and carbon (loose fibers or fiber blankets). Significant thermal data is available for alumina-enhanced thermal barrier Al_2O_3 – SiO_2 rigid fiber tiles produced for space shuttle applications. An example of a low-density carbon fiber blanket is one manufactured by Calcarb[28].

Because of their low thermal conductivity, high-temperature fibrous insulations are prime candidates to be used throughout the reactor system and the PCU. Typical applications would be within the Hot Duct (tubular-shaped structural line plus insulation), upper plenum shroud, SCS helium inlet plenum, and turbo-compressor. The Hot Duct canisters would be in direct contact with the hottest gas conditions in the reactor. Thus, the materials chosen for these canisters will need to withstand temperatures in the 1000 °C to 1200 °C range. This is a very high service temperature for super-alloy metals; therefore, non-metallic materials such as C_f/C composites may be required for the canisters. The sandwich design would apply to the Hot Duct canister with the outer shells being constructed of metal or a ceramic matrix composite (see Figure 8). The insulation within the canisters is required to retain its physical characteristics during normal operating and conduction-cool-down-accident conditions up to 1200 °C.

An alternative to the sandwich design is one where ceramic fiber "blankets" of various configurations can be attached to cooler outside structures using refractory pins and washers. This design could possibly eliminate containment canisters, but would leave the outside of the fiber blankets directly exposed to the full velocity of the cooling gas. Refractory ceramic coatings (e.g., SiC) would probably have to be used on the exposed surfaces of the fibrous structures in order to minimize erosion or "powdering" that would produce entrained (probably erosive) particles within the cooling gas.

Operating conditions for fibrous insulation include low neutron fluence (<0.01 dpa) and gamma flux, and high temperatures. Mechanical loads on the thermal insulation result from differential thermal expansion, acoustic vibration, seismic vibration, fluid flow friction, and system pressure changes of up to 100kPa. As an example, a pressure drop difference of between 70-100 kPa is anticipated between the hot and cold side of the Hot Duct. The insulation (or insulation/canister package) must withstand these forces over an extended period of time.

The maximum temperature rating is typically 1260 °C for the highest rated Al_2O_3 - SiO_2 mixed-ceramic fiber mat insulation. By reducing the fraction of silica in the fibers, or through the addition of ZrO_2, insulating mats can achieve continuous and maximum operating temperatures of 1300 and 1400 °C, respectively. High purity alumina fiber mat can be used at operating temperatures above 1500 °C. The carbon fiber insulation materials can operate at temperatures considerably higher than 1500 °C in inert atmospheres and should have minimal chemical compatibility problems in the NGNP internal environment of helium and graphite (if fibers were encapsulated in a C_f /C canister or shell). Therefore, carbon fiber insulation should be considered as a serious candidate.

3.1.6 Power Conversion Turbine and Generator

The NGNP PCU involves the turbine, turbine inlet shroud, the generator and recuperators. The recuperators are covered in the next section. Considerable materials work is involved in both the turbine and the generator components and existing component manufacturers are an excellent source of the needed materials information. As such, much of the turbine and generator materials efforts will be performed via subcontracts to existing manufacturers. However early efforts should be conducted in-house to identify the materials preferred by various manufacturers and to assess the performance potential of these materials under operating conditions representative of the NGNP. The high-temperature bellows and turbine inlet shroud will be the primary focus of the NGNP materials efforts. These components operate continuously at close to 1000 °C. Off-normal (accident) temperatures for these components are about the same as their maximum operating temperatures. The potential materials to be selected for these components will need to come from the listing of high-temperature materials listed in Table 6.

The turbine inlet shroud accepts the coolant exiting the hot duct and directs it to the turbine inlet. It is insulated to minimize thermal gradients and heat loss across the shroud wall. There is, however, a stiffening element or collar between the shroud and the turbine that is not insulated and experiences the maximum system operating temperature, 1000 °C. This non-insulated collar can be exchanged at each period of planned turbine maintenance (nominally every 7 years). The boundary/container material for the shroud insulation must also withstand the 1000 °C helium temperature. Prime potential candidates for the non-insulated turbine inlet shroud collar are Alloy 617 and the cast Ni-base alloys.

3.1.7 Power Conversion Recuperators

The recuperator is a modular counter-flow helium-to-helium heat exchanger; its most likely design has corrugated-plate heat exchange surfaces. Helium inlet temperature will be 490 °C and desired design life is 60 years. Prime candidate materials for this application are the 300 Series stainless steels listed in Table 6.

The power conversion recuperator operating temperatures are relatively low, with a 490 °C inlet gas temperature from the turbine exhaust and a less than 200 °C outlet temperature. Recuperator technology for the temperatures and pressures of operation is relatively mature. For gas turbine applications, tube-on-plate and primary surface units are often fabricated from fine-grained 300 series stainless steels. Type 347 stainless steel is typical. Recuperators in which the corrugated plate surfaces are sealed by brazing have suffered from thermal fatigue when pushed to higher temperatures, but the NGNP operating conditions will not subject the recuperator to severe cycling. A pressure differential across the membrane wall is expected, so some consideration of creep will be needed to prevent closure of the low-pressure gas passages in the corrugations. Since relatively thin sections will be present, environmental effects must be considered. Also, long-time exposure of 300 series stainless steels at these high temperatures often leads to sigma phase embrittlement and carburization.

3.1.8 Valves, Bearings, and Seals

A few valves may be required in the primary or secondary piping systems for this plant, and a flapper valve is used in the SCS. Bearing surfaces exist between the RPV and the core barrel. Seals may be required in a variety of locations. However, insufficient information relating to the specific requirements and issues relating to valves, bearings, and seals is available at this time to initiate a selection activity. It is expected that a materials R&D program covering these areas will be added in later revisions to the plan.

3.2 Materials Qualification Testing Program

The following project list follows the current priority order being followed by the NGNP Materials R&D Program. These projects are highly integrated with each other and the work is currently performed primarily at the INL and ORNL. The Deliverables and Milestones are listed in Section 4.

The content of the projects listed is specifically designed to envelop the high-priority and long-lead materials R&D information anticipated to be required regardless of the NGNP system design chosen. Subsequent detailed project content will be developed in keeping with the results of the work described herein:

- Graphite Development
- HTDM
- Code Committee Support
- Environmental Testing and Thermal Aging
- Irradiation Facility

- Composites
- Database and Handbook
- Power Conversion Turbine and Generator
- RPV
- Emissivity
- Metallic Core Internals
- IHX Fabrication
- Hot Duct Insulation Testing
- Valves, Bearing and Seals
- Administration.

3.3 Graphite Development Project

The work described in the remainder of Section 3 gives a summary of the results obtained for each task in FY-05 or plans to perform the task in the future. The work to be performed for each task in FY-06 is also provided. The format used for each task is the format used in the work packages used at the INL and the ORNL for each task. Please note that the format used in the work package to identify the task changed from FY-05 to FY-06, however, the follow-on work to be performed in FY-06 is given immediately after the FY-05 work performed and is therefore self explanatory. Essentially all work in FY-05 and the work to be performed in FY-06 has been or will be performed at the INL or the ORNL due primarily to program funding constraints. The only exception to this is a small amount of work that is ongoing in the composites area that is performed at the Battelle Pacific Northwest National Laboratory (PNNL). Essentially all work performed has been or will be documented in the form of reports which are referenced in the description provided.

3.3.1 Task 1A (INL and ORNL): Graphite Selection Strategy

FY-05 Activities

The grade of nuclear graphite (H-451) previously used in the United States is no longer available. New graphites have been developed and are currently being considered as candidates for the NGNP. Irradiation data from ongoing experiments in Petten Reactor (European program) will be of great value. A complete properties database on the new, available candidate grades of graphite will be developed to support the design of graphite core components for those graphites selected to be used in the NGNP. Data are required for the physical, mechanical (including radiation induced creep) and oxidation properties of graphites. Moreover, the data must be statistically sound and take account of in-billet, between billets, and lot-to-lot variations of properties. The data will be needed to support the ongoing development of the risk-informed ASME graphite design codes. The previous Fort St. Vrain design used deterministic performance models for H-451, while the NGNP will use new graphite grades and risk-informed performance models. The risk-informed approach is currently being used by the NRC for consideration of new license applications. A radiation effects database must be developed for the currently available, graphite materials and this requires a substantial graphite irradiation program. There is the potential to leverage data from European Union activities in the area of irradiation experiments on PBMR graphites (Petten Reactor irradiation experiments are currently being initiated). Prior to selection of a specific NGNP design, properties data will be obtained for the currently available graphites to support design activities for both the GT-MHR and the PBMR. Candidate graphite materials are presented in Table 11.

Table 11. NGNP Materials Program: graphite grades, vendors, and available processing information

Graphite Grade	Source	Country of Origin	Process Details
NBG-17	SGL Carbon	Germany/France	Pitch coke, vibrationally molded, medium grain
NBG-18	SGL Carbon	Germany/France	Pitch coke, vibrationally molded, medium grain
H-451 (Reference Grade)	SGL Carbon	USA	Petroleum coke, extruded, medium grain
PCEA	GrafTech International	USA	Petroleum coke, extruded, medium grain
IG-110 (Reference Grade)	Toyo Tanso	Japan	Petroleum coke, isostatically molded, fine grain
IG-430	Toyo Tanso	Japan	Petroleum coke, isostatically molded, fine grain
Highly Oriented Pyrolytic Graphite (HOPG)	Advanced Ceramics	USA	Gas phase deposition, high temperature annealed
A3 Matrix Graphite/Carbon	ORNL	USA	Blend of natural flake & manufactured graphite powders, phenolic resin bonded, hot pressed & carbonized
HLM	SGL Carbon	USA	Petroleum coke, extruded, medium grain
PGX	GrafTech International	USA	Petroleum coke, molded, medium grain
PPEA	GrafTech International	USA	Pitch coke, extruded, medium grain
NBG-25	SGL Carbon	Germany/France	Isostatically molded, fine grained
2020	Carbone of America	USA	Isostatically molded, fine grained
PCIB	GrafTech International	USA	Petroleum coke, Isostatically molded, fine grained
BAN	GrafTech International	USA	Petroleum (needle) coke, secondary/green coke process, extruded, medium grain
NBG-10	SGL Carbon	France	Pitch coke, extruded, medium grain

Nuclear graphite codes and standards development is required in support of the NGNP. ASTM standard test methods are required for determining key physical and mechanical properties, for example, the critical strain energy release rate (K_{Ic}), the crystalinity of the graphite (by X-ray diffraction), coefficient of thermal expansion, and the air oxidation rate. ASME design codes must be developed for the graphite core supports structures and carbon-carbon composite structures to be used in the NGNP. Activities in support of the graphite design code have already been initiated by a task group under the auspices of ASME Section III. Carbon-carbon composites are required for NGNP components such as control rod structural elements, upper vessel insulation support structure, and insulation shroud covers.

The graphite selection strategy has been developed by interaction and discussion with the GIF. The GIF has developed a graphite program 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. Discussions with other graphite vendors will be conducted in the future. Graftech has yet to make a decision to commercially produce BAN. All other grades of graphite are available commercial graphite grades.

It was agreed that one or more billets of the following graphite grades be purchased for inclusion in the graphite program. The graphites to be purchased are categorized into major, minor, and experimental grades. The graphites in these grades are given in Tables 12, 13 and 14.

Billets of major grades will be purchased when funding becomes available for subsequent preirradiation characterization. Both SGL and Graftech have pledged a sufficient quantity of the graphites PCEA, NBG-17 and -18 for irradiation testing at no cost.

Table 12. 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

Billets of the following minor grades will be purchased based on available funding for characterization.

Table 13. 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

The following experimental graphite grades are given in Table 14 and will be included in irradiation experiments only with no precharacterization planned.

Table 14. Experimental Grade Graphite.

Grade	Manufacturer	Comments
BAN	Graftech International	Extruded, candidate for high dose regions of VHTR concepts
HOPG	Advanced Ceramics	Vapor deposited and compression annealed. A model material for the graphite single crystal
A3-matrix	Oak Ridge National Laboratory	Hot pressed graphite/carbon used for the matrix of AGFR fuel compacts

FY-06 Activities

The work in this task was completed in FY-05 and further work is not planned in FY-06.

3.3.2 Task 1B (INL and ORNL): Procurement of graphite for irradiation and testing[29]

FY-05 Activities

Discussions have been held with the graphite vendors regarding acquisition of materials for inclusion in the graphite program. Where relatively modest amounts of material are required for the AGC-1 irradiation program the materials will be supplied free of charge. Samples (typically a 200 lb block cut from production billet) of the following graphites have been received to date or sufficient material was available at ORNL:

- SGL Grade NBG-10
- SGL Grade NBG-18
- GrafTech PCEA
- IG-110 & H-451 (these grades are available at ORNL from prior DOE-NE research programs).

A similar size block of SGL's grade NBG-17 will be shipped to ORNL in late calendar year 2005 from a production batch that is currently being manufactured. Similarly, sufficient quantity of the minor graphite grades HLM, PGX, PPEA, and PCIB will be provided by graphite vendors shortly. The experimental grades listed above (A3, BAN) will also be provided by graphite vendors or government programs or purchased. Samples of A3 will be obtained from the Advanced Fuel Cycle Initiative (AFCI) program and will be produced at ORNL. BAN graphite will be provided by GrafTech from the Parma Technical Center, USA. Single Crystal Graphite (Highly Oriented Pyrolytic Graphite, HOPG) will be procured from GE Advanced Ceramics at an estimated cost of $2,000 to $3,000.

Where larger quantities of material are required for characterization to determine the statistical variation of properties due to texture, density gradients, etc., full size billets of the grades must be purchased. It is estimated that the material cost will range from $5.50 to $6.70 per pound yielding a per billet cost of $7,500 to $10,000 depending on the size of the graphite billet.

Recently, a 250 mm thick slab from a full size production billet of PGX graphite (914 mm diameter) was purchased. This material will support ASTM standards development round-robin testing in the areas of graphite oxidation and fracture toughness. The purchase price of the PGX was ~$5,000.

Due to the high cost of full size billet procurement, it is anticipated that the number of graphites to be subjected to full characterization will be limited to grades PCEA, IG-430, and NBG-17/NBG-18. Purchase of the requisite number of billets is estimated to cost ~$50,000. If purchase of billets of PGX and HLM is required (the use of these latter grades is dependant upon specific NGNP design) additional funding will be required. Since billets of PGX and HLM are significantly larger than those of the major grades their cost is greater.

FY-06 Activities

Procurement activities of graphite billets are limited to available funding. Currently, funding for this activity is not in the base FY-06 budget.

3.3.3 Task 1C (INL and ORNL): Graphite Irradiation Creep Capsule Design and Planning

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). 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 designed 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 are planned for irradiations at 600 and 1200 °C to provide design data over the anticipated graphite in-reactor operating temperature for a PMR design. These experiments do not cover all conditions required to investigate the graphite to be used in the PBR design due to lack of funding in the program.

The AGC-1 irradiation capsule contains six pneumatic pistons that apply a controlled stress to the graphite creep samples accommodated in six peripheral channels of the capsule. Two stress levels will be utilized in AGC-1, 13.8 MPa (2 ksi), and 20.7 MPa (3 ksi). These stress levels were chosen based on: (1) historic norms (2 and 3 ksi were used in the OC series of irradiation creep experiments performed at ORNL in the 1970's and 80's) and (2) detailed discussions with reactor vendors via the ASME graphite core design project team. In addition, each of the six peripheral channels in AGC-1 contains companion unstressed graphite specimens. In addition to the unstressed creep control samples each peripheral channel contains a number of smaller, so called "piggyback" samples of VHTR relevant graphites. These piggyback specimens do not provide irradiation creep data, but do provide valuable physical properties data.

The center channel of capsule AGC-1 additionally accommodates a large number of piggyback samples as well as silicon carbide (SiC) temperature monitors whose purpose is to provide a post irradiation check on the irradiation temperature.

Creep data will be obtained for six major graphite grades, i.e., both stressed and unstressed samples of these grades are included in the capsule. The major grades are H-451 and IG-110, both of which are included as reference graphites, and four new grades, PCEA, NBG-17, NBG-18, and IG-430. In addition, AGC-1 contains ten minor grades of graphite. These minor grades are not located in the stressed section of the capsule, and thus no creep data will be generated for them. However, as discussed subsequently, they will yield significant amounts of design data. The minor grades include candidates for lower dose locations in the prismatic reactor designs, such as permanent reflectors and core supports components, such as grades HLM, PGX, PCIB, NBG-25, and 2020. Three additional grades of graphite are included in AGC-1 because of their interest to the NGNP program, grades NBG-10, PPEA, and BAN. Grade NBG-10 is an extruded grade and its behavior is of interest in comparison with the vibrationally molded grades NBG-17 and -18. Grade PPEA provides a comparison of the performance of identical pitch-coke and petroleum-coke graphites. BAN graphite is an experimental grade that is expected to exhibit superior irradiation behavior. A3 fuel matrix (a graphite filler carbonized resin binder material) is included to yield dimensional change and physical property data for the AFCI research program. Finally, samples of HOPG are included to provide vital data on the crystal dimensional change rates, and hence the parameter XT. The graphites to be included in AGC-1 are given in Table 15 along with information on their potential application in an NGNP.

Table 15. 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
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

Graphite	Reactor Vendor	Proposed Use	Capsule Location	Remarks
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

FY-06 Activities

In FY-06 it is planned that the gas system that will be used in conjunction with the AGC-1 experiment will be fabricated and installed in the ATR; the AGC-1 capsule will be fabricated and the design review completed; the final AGC-1 experiment design and test plan will be completed; and the specimens that will be used in the AGC-1 experiment will be fabricated and inspected.

It is planned that the AGC-1 experiment will be installed in the ATR in the first quarter of FY-07. A final AGC-1 design and test plan is required because of the likelihood of the installation of new experiments in irradiation facilities adjacent to the South Flux Trap in the ATR prior to the first quarter of FY-07. It is anticipated that these experiments will affect the temperature and nuctronic calculations that are reflected in the current experiment design and test plan. Therefore, a final experiment design and test plan will be issued prior to the installation of the AGC-1 experiment in the ATR. The final test plan will account for the changes that are made to the environment in the ATR adjacent to the South Flux Trap. ORNL Activity 101 and INL Activities 101, 102, 103, and 106 will support this FY-06 work.

3.3.4 Task 1C (ORNL): HFIR Rabbit Capsule Post Irradiation Examination

FY-05 Activities

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 United States and Europe. NBG-10 is one of several graphites included in the DOE VHTR program.

Thirty-six NBG-10 graphite flexure bars have been successfully irradiated in a series of 18 High Flux Isotope Reactor (HFIR) PTT capsules at ORNL. The capsule design temperatures were 300, 500 and 800°C. The peak doses attained were 4.93, 6.67, and 6.69 x 10^{25} n/m^2 [E>0.1 MeV] at ~300, ~500, and ~800°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 (~300°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. This information is detailed in *Initial Post Irradiation Examination Data Report for SGL NGB-10 Nuclear Grade Graphite*, ORNL/TM-2005/518 [30]. While the data reported here 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.

<u>FY-06 Activities</u>

It is planned that the PIE examination of NBG-10 HFIR bend-bar samples including measurement of the temperature dependency of the thermal conductivity, SEM examination of selected specimens and interrogation of the temperature monitors will be completed in FY-06. ORNL Activity 104 will support this FY-06 work.

3.3.5 Task 1D (INL and 1F (ORNL): Graphite Model Development for Predicting Irradiation Effects

<u>FY-05 Activities</u>

The data acquired from AGC-1 and other planned irradiation capsules will be used in the numerical modeling of the physical behavior of the graphite. The modeling starts with taking into account the irradiated behavior of the physical properties of graphite. This takes the form of building mathematical expressions as a function of temperature and fluence for the property. The next step is to use the irradiated properties mathematical expressions in a numerical model of the graphite core using methods such as finite elements. The finite element model will be used to determine the effect of temperature and fluence on the structural response of the graphite core during its lifetime. This work is discussed in *Physically Based Models of the Behavior of Nuclear Graphite under Neutron Irradiation*, ORNL/TM-2005/509 [31].

The operating environment of a high temperature gas cooled reactor places significant demands on the graphite moderator. The properties of the graphite are markedly affected by neutron irradiation and these effects must be allowed for in the design of core components and structures. Since it is not possible to experimentally determine these effects for all possible combinations of operating temperature and neutron dose, sound physical models for the effects of neutron damage on key graphite properties need to be developed. The models thus allow prediction of property changes in the absence of experimental data. The models need to be validated against experimental data over the widest possible envelope of doses and temperatures. Key design properties that must be modeled are dimensional change, irradiation creep, thermal conductivity, thermal expansion, strength, and failure probability.

The process for the development of models and the physical basis for the modeling process were developed in FY-05. This included the procurement of a SUN workstation, discussions with the National Science and Aerospace Administration (NASA) with regard to the interfacing of the CARES probabilistic analysis software package to the FEMLAB modeling software and the documentation of the physical basis of the modeling process.

<u>FY-06 Activities</u>

Model development activities, ORNL Activity 107 and INL Activity 109, will be continued in FY-06. This will include the development of a software module that will couple the NASA developed CARES software with the INL FEMLAB finite element analysis program, development and evaluation of example problems and the integration of irradiated and unirradiated HFIR bend-bar data into the graphite modeling software.

3.3.6 Task 1E (ORNL) Modify ASTM C-1421 Standard for Fracture Toughness Testing of Graphite

FY-05 Activities

ASTM committee D02.F has been working on a new test method for determining the Critical Stress Intensity Factor (K_{Ic}) 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 2004 meeting. The committee recommended some changes in the proposed method be made and a second ruggedness study was conducted in early 2005. The results of the second ruggedness test where 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 are currently being 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) is currently being shipped from SGL (USA) to ORNL and specimens will be machined subsequently. It is anticipated that the draft test method and round robin specimens will be distributed to participants in August 2005. Twelve labs in six different countries wish to participate in the fracture toughness round robin. The establishment of a standard method is a prerequisite for generation of graphite fracture toughness data needed to support the ASME design code development discussed in Section 3.5 (ATSM and ASME Code Support). The draft ASTM standard test method that was developed was reviewed and approved by ASTM Committee DO2-F at their meeting in June 2005. The status of this task was documented as *Development of a Fracture Toughness Testing Standard for Nuclear-Grade Graphite Materials*, INL/EXT-05-00487 [32].

FY-06 Activities

It is anticipated that the round robin testing associated with the draft standard that was developed will be completed in FY-06. Currently, only INL Activity 308 supports this FY-06 work.

3.3.7 Task 1G (ORNL): High-temperature Graphite Irradiation Experiments

FY-05 Activities

A preliminary design and experimental plan has been prepared for the high temperature graphite irradiation experiments. The irradiation capsules will be HFIR target capsules (spline design), each containing 64 specimens of nominal dimensions 12 mm outside diameter, 3 mm inside diameter, and 6 mm thickness. The exact outside diameter will vary specimen to specimen to allow the gas gap to be set and thus determine the irradiation temperature. The capsules will each contain three temperature zones: 900 °C, 1200 °C, and 1500 °C. The capsule spline materials will be graphite, molybdenum, or tungsten, in the 900, 1200, and 1500 °C zones, respectively. The capsules will be filled with argon gas. Flux wires and SiC temperature monitors will be included in the 900 °C sections of the capsules.

Four graphites will be included in the experiments, all of which are candidates for the NGNP-VHTR, namely, PCEA (GrafTech), NBG-17 and NBG-18 (SGL), and IG-430 (Toyo Tanso). Additionally, a few samples of H-451 will be included in the 900 °C sections to provide reference data. Two capsules, designated HTV-1 and HTV-2, will be irradiated for one and three cycles, respectively yielding data over the dose range 0.7 to 4.8 DPA. The primary data to be obtained from these experiments include dimensional changes, Young's modulus and ring strength changes, and room temperature thermal conductivity degradation.

The summary report for this task was issued late in September and therefore could not be reference in this report.

FY-06 Activities

It is planned that the test plan and design for the high temperature capsule irradiation in HFIR will be completed in FY-06 and capsule fabrication will be initiated. ORNL Activity 110 will support this FY-06 work.

3.3.8 Activities 113-115 (ORNL), PIE on graphite METS Capsules from HFIF.

FY-06 Activities

In 2004, "new" graphites (relevant to the VHTR) were irradiated in the Fusion Energy Mapping Elevated Temperature Swelling (METS) capsules in HFIR. The experiments, which comprised three capsules each segmented into 10 temperature zones (in the range 600-1500 C) and to peak doses of ~ 2,6, & 10 dpa, respectively, have now completed irradiation. The capsules hold samples of graphites NBG-10 (SGL) and PCEA (GrafTech) as well as some reference samples of H-451. A total of ~ 60 samples (5 mm diameter x 4 mm thickness) will be examined. The dimensional changes behavior and the degradation of thermal conductivity will be obtained in the Post Irradiation Examination of the samples.

This data will be of great use in guiding the detailed design of the HFIR 1200 °C capsules (FY-06 activities) which will contain several additional VHTR candidate graphites. Moreover, the data will be directly relevant to the design of the high temperature (1200 °C) ATR creep experiments to be conducted at INL in the future.

3.3.9 Activity 116 (ORNL), International Nuclear Graphite Specialists Meeting

FY-06 Activities

It is planned that the International Nuclear Graphite Specialists Meeting will organized and hosted by ORNL under ORNL Activity 116.

3.3.10 Other Unfunded Graphite Activities (Unfunded in FY-06)

Other unfunded graphite activities are identified in Table 16 by title, description, benefit to the NGNP program, and requested funding.

Table 16. FY-06 Unfunded Graphite Activities

Task Title	Task Description	Benefit to NGNP Program	Total Funding Request, $K	Funding Split, $K	
				ORNL	INL
High dose graphite irradiation creep experiment	Graphite irradiation creep data are needed to high dose (~20 DPA) to support the design of the Pebble Bed variant of the VHTR. This task will be to design a high dose creep capsule and facility for the HFIR target (flux trap), construct and test the capsule, and design and purchase the control system. Irradiation would commence on FY-2007. INL will consult on design of the creep experiment.	At DOE's request, this work is being considered as the US DOE contribution to a CRADA with PBMR Pty Ltd of the Republic of South Africa. The high dose creep experiment is in the current NGNP program plan but is not scheduled to commence until ~FY-2010. This schedule does not support the PBMR needs. A CRADA with PBMR would assure US access to the data produced by ORNL for PBMR at PBMRs expense (i.e. fund-in CRADA). PBMR funds in will be several million $'s so the leverage to the NGNP program is very favorable.	1,100	1,000	100
Graphite billet characterization	Billets of candidate NGNP graphite will be purchased. The billets will be slabbed and test specimens machined to characterize the spatial variations of properties. This will be a coordinated effort between ORNL and INL. ORNL will be responsible for the purchase of the graphite and specimen preparation. Testing will be performed at both INL and ORNL. Data to be obtained includes density, thermal conductivity, CTE, tensile strength (stress-strain behavior), flexural and compressive strength, dynamic and static modulus.	These data are needed to support ASME code development and NGNP design activities. This task accelerates the currently planned schedule.	750	400	350
High temperature graphite irradiation capsules	The construction of the HTV-1 and -2 capsules will be accelerated to meet a HFIR insertion date of June 2006 and an irradiation completion date of Sept 2006. Additional funds are needed to cover full construction and neutron charges in FY-06. Neutron costs are estimated to be <$200k	Accelerating the current schedule (which calls for construction to start in FY-06 and irradiation to begin in mid FY-07) makes high temperature dimensional change data available on a schedule that better supports the ATR high temperature graphite creep experiments	450	450	0
Graphite KIc data analysis	KIc data will be produced by several (>10) labs during the ASTM fracture toughness round robin. These data need to collated and statistically analyzed to establish the precision and bias of the proposed test method. A research report must be prepared and submitted to ASTM prior to balloting the new test methods.	These data are needed to support ASME code development and NGNP design activities. This task accelerates the currently planned schedule.	150	50	100
Effect of air oxidation on physical properties of graphite	Graphite specimens will be oxidized to differing burn-off in air and the effects on properties determined. An experimental rig (TGA apparatus) at ORNL allows up to 200 g samples to be oxidized to controlled burn offs. Real time monitoring of weight loss allows collection of oxidation kinetics data. The samples are sufficiently large to allow mechanical and physical property testing on the oxidized samples. NGNP candidate graphites will be examined. A duplicate system will be established at INL. The properties will be obtained through desctructive testing.	These data are needed to support ASME code development and NGNP design activities. This task accelerates the currently planned schedule.	500	200	300
ASME code development - NASA participation	Support is needed to allow NASA personnel develop the CARES package for graphites and to fully participate in the ASME code development process	Accessing the NASA expertise on the CARES code will accelerate the completion of the ASME graphite design methodology	50	0	50

Task Title	Task Description	Benefit to NGNP Program	Total Funding Request, $K	Funding Split, $K	
				ORNL	INL
Effect of air oxidation on physical properties of carbon-carbon composites	Carbon-carbon composite specimens will be oxidized to differing burn-off in air and the effects on properties determined. An experimental rig (TGA apparatus) at ORNL allows up to 200 g samples to be oxidized to controlled burn offs. Real time monitoring of weight loss allows collection of oxidation kinetics data. The samples are sufficiently large to allow mechanical and physical property testing on the oxidized samples. VHTR candidate composites will be examined. A duplicate system will be established at INL under the graphite program.	These data are needed to support ASTM standard development and VHTR design activities. This task accelerates the currently planned schedule.	500	200	300
Non-Control Rod Carbon-Carbon Composite Characterization	There are many areas of the reactor that will need carbon-carbon, but will not be subjected to the very high fluxes in the core. These composites will not need to be made from the expensive pitch-based fibers and matrices. These composites need to be evaluated and characterized to ensure that the manufacturing techniques can produce reliable and repeatable materials. Testing will be performed at both INL and ORNL. Data to be obtained includes density, thermal conductivity, CTE, tensile strength (stress-strain behavior), flexural and compressive strength, dynamic and static modulus.	These data are needed to support ASTM standard development and VHTR design activities. This task accelerates the currently planned schedule.	750	300	450
Preliminary design of AGC-1 Sizing Apparatus	Perform a preliminary design of a cask/sizing tool that would be used to remove AGC-1 from the ATR core, cut off both ends of the experiment, size the experiment to fit in the GE-2000 cask and seal the ends of the capsule for transportation to the Hot Fuel Examination Facility on the MFC campus.	This capability does not exist at the ATR. This equipment is needed to support the AGC series of experiments	200		200
GIF Graphite Irradiation Review	This task would involve performing a review of both historical and ongoing graphite irradiation data available through the International Atomic Energy Agency (IAEA). This would be a joint effort between INL and ORNL requiring foreign travel to IAEA contributor's sites for discussion with principle investigators and physical collection of data.	Provide prospective on the NGNP Graphite R&D Program and support the GIF PMB collaboration plan for graphite		50	50
		TOTAL FUNDING	**4,450**	**2,650**	**1900**

3.4 High-temperature Design Methodology Project

3.4.1 Task 2A (INL), Procurement of Alloy 617

FY-05 Activities

This task was discontinued because it was decided to use Alloy 617 that was located at the INL and purchased previously for a discontinued program for FY-05 testing activities to be performed at ORNL and the INL. Two pieces of forged plate material meeting ASTM B 168-01 were used: ¾ inch plate thickness obtained from Special Metals, Heat Number XX2834UK and ½ inch plate thickness obtained from Haynes, Heat Number 8617-3-8810.

FY-06 Activities

There will be not FY-06 activities performed in this area.

3.4.2 Task 2B (INL), Procure, Install and Checkout an Environmental Chamber For a Creep-Fatigue Test Machine

FY-05 Activities

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. The system is deemed ready for use. This work will be documented in a report to be issued shortly. A photograph of the system is shown in Figure 20.

Figure 20. Environmental chamber installed on creep-fatigue load frame.

FY-06 Activities

A new mechanical test load will be procured under Activity 201 in FY-06. A new environmental chamber for this load frame will be procured and installed on this frame when additional funding becomes available.

3.4.3 Task 2A (ORNL), Initiate Alloy 617 Database Assembly

FY-05 Activities

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 [33]. 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.

FY-06 Activities

This task was completed in FY-05.

3.4.4 Task 2B (ORNL) Develop Controlled Material Specification for Alloy 617 for Nuclear Applications

FY-05 Activities

An investigation has been 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. 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 17. 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 17. 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. It should be stressed that the manufacturing viability of the refined specification is currently under discussion with the vendors and a meeting date with several vendors has been established in October 2005 at ORNL. It should be pointed out that the composition in Table 17 is recommended as a “best possible shot” effort under the current conditions regarding the historical data status and applicable information availability. Improvements should be expected as more and better information becomes available.

Recent documented personal communications with the Special Metals indicate that their present practice ensures chemistry analysis to be performed after the ESR, an important improvement compared to the old practice. Large grain sizes are usually preferred for good creep resistance. However, it has to be balanced with good crack initiation resistance and other factors. One important consideration is the potential application in the Intermediate Heat Exchanger (IHX). Too large a grain size may cause some problems in the thin sheets required for the IHX. A grain size range of ASTM #3-5 is recommended. Personal communications with the Special Metals indicate that they have gained much better control on the grain size than many years ago when the heats used for the ASME Code stresses were produced. Currently, they can produce the alloy in a typical grain size range of ASTM #3–6. It is likely that the recommended grain size range can be achieved.

Solution annealing at ASME specified minimum temperature of 1149ºC (2100ºF) for a time commensurate with section size, followed by water quench cooling is recommended. A vacuum induction melting (VIM) + electro slag remelting (ESR) melting method for accurate control of the specified chemistry is also recommended. A check analysis after ESR on all the specified elements should be performed and reported. Additionally, heat analysis should be provided as a reference. It is recommended that the product form be hot rolled ¾” thick plate for convenient specimen machining.

The recommended specification is based on the analyses of historical data, literature review, experience, and knowledge about the alloy. As can be seen in the analyses, great difficulty lies in the fact that the historical data were not systematically generated for the purpose of refining the specification to improve properties. The analyses have also suffered from incomplete information, inconsistent, and even erroneous data mostly generated some 20 to 30 years ago during the HTGR time. Many advanced materials manufacturing and processing technologies were not available and many important factors that significantly affected the properties were not well controlled when these data were generated; whereas heats produced in recent years that are much better controlled lack the abundance of existing data. Furthermore, the current investigation was only given approximately half a year to reach a tentative conclusion compared to the five years it took for the Modified 9Cr-1Mo steel to be developed from the standard 9Cr-1Mo steel, the deficiency of time is apparent. Therefore, the recommended specification should be considered tentative and subjected to further refining when more information becomes available.

FY-06 Activities

To verify and further refine the recommended specification, the following actions will be integrated into the work scope in FY-06 and follow-on years based on available funding:

- Much work on Alloy 617 has been undertaken in Germany after the termination of US interest in nuclear application of the alloy in the late 1980's. Efforts have been underway to acquire recently generated data from the international community, especially from the Germans. Such efforts must be continued as more and more countries officially sign the Gen IV collaboration agreements with the United States

- Actions should be taken to gain a complete understanding of the correlation between the high temperature properties and the strengthening mechanisms of the alloy. Currently, such work is being conducted under the Fossil Energy Advanced Research Materials Program for a target temperature of 760ºC (1400ºF) for the Ultrasupercritical Steam Boilers. Collaboration should be initiated with the fossil program to investigate the possibility of retaining γ' and the other strengthening precipitates at the temperature range of interest to the VHTR applications.

- Computational modeling should be conducted on the prediction of the second phases at temperatures and times of interest to the VHTR applications. The modeling results will provide guidance for further refining the specification, or systematically designing metallurgical experiments for refining the specification.

- The high creep strength of CCA 617 should be further verified with more tests.

- If necessary, metallurgical experiments should be conducted to investigate the effects of the variations in Al, Ti, C, Mo, Co, B, and N on properties of the alloy on a well designed systematic basis with the guidance from computational modeling.

It should be pointed out that if the above action is considered necessary, the refining process may need to be iterated until satisfactory results are yielded or conclusions are drawn with sufficient experiment support. The possibility can not be ruled out that in reaching for significantly improved properties the iterations may carry the search beyond the limits of the standard specifications, which implies that the specification refinement is being turned into an alloy modification or development effort. Once the iteration is started, long cycles of manufacturing, testing, and analyses should be expected. Therefore, a managerial decision will have to be made based on the required timeframe and funding availability. Experience shows that such a process usually takes several years.

3.4.5 Task 2C (ORNL) Status and Plans for Initial Scoping Tests for Creep and Stress-Strain Evolution and Code Submittal for Inconel 617

FY-05 Activities

Scoping tests on Alloy 617 were conducted in FY-05 to provide time-dependent input for HTDM constitutive equation development. This work was reported in *Initiation of Scoping Tests to Provide Time-Dependent Input for HTDM Constitutive Equation Development*, ORNL-GEN4/LTR-05-006 [34]. Lever arm creep machines were employed in the testing and focus was placed on providing information for refurbishing additional machines and checking operational procedures and guidelines for use in the Gen IV program. 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. Figure 21 illustrates a typical creep strain vs. time curve measured in the scoping tests. Additional scoping tests included strain controlled tensile tests. Figure 22 illustrates a typically observed stress-strain diagram from an elevated temperature strain-controlled tensile test.

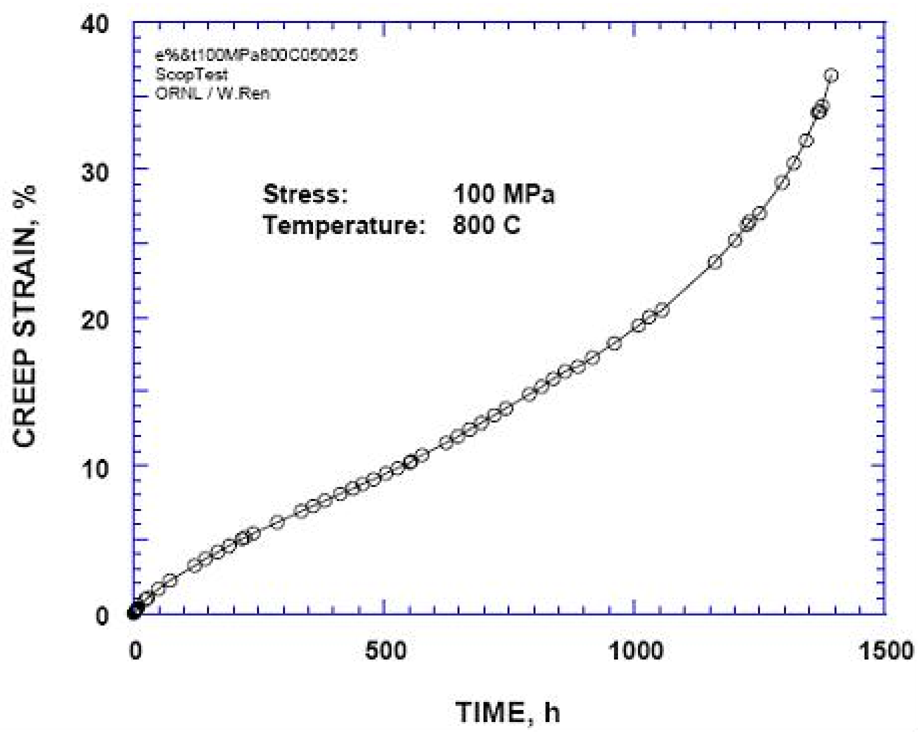


Figure 21. A typical creep strain curve for Alloy 617 generated in the scoping creep testing in air environment [ORNL-GEN4/LTR-05-006]

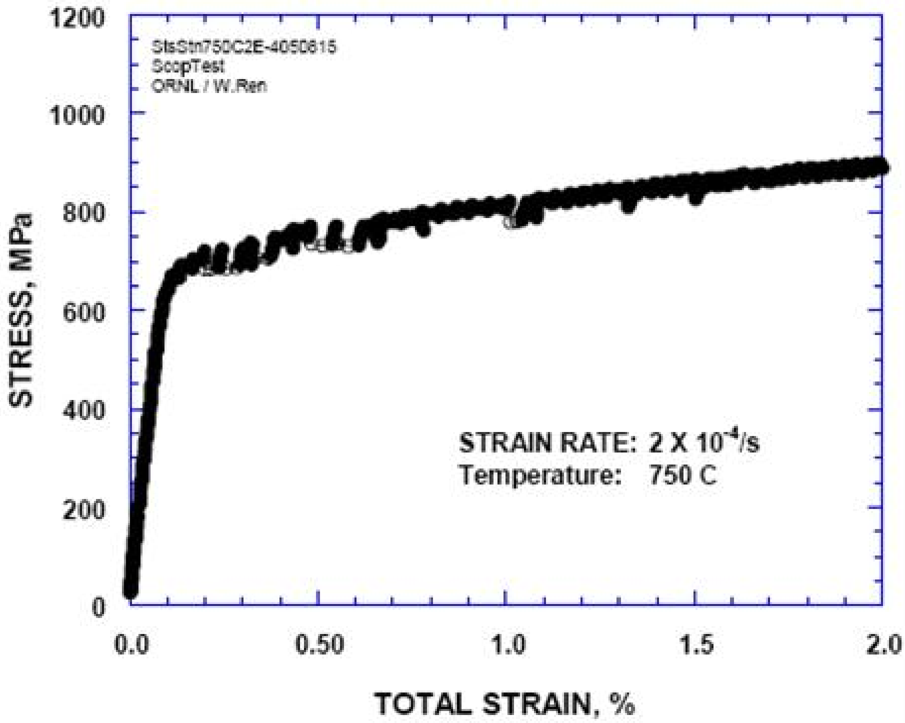


Figure 22. A typical stress-strain curve from scoping strain-controlled tensile testing.

A review of the previous HTGR program data generated during the 1970's and 1980's was also conducted. This effort revealed that the scale of the test facilities and capability at ORNL and GE during the HTGR program was huge – up to 415 simultaneous creep tests in simulated reactor helium environment and 188 simultaneous creep tests in air. Based on the available funding in FY-05, only two environmental creep frames could be refurbished for the scoping test effort. Expansion of the environmental creep testing system with shared gas chromatographs and data acquisition computers is in progress. Two machines only comprises 0.5% of the previous HTGR environmental creep testing capability, but will be used as a prototype to verify the system design with current technology prior to expansion. Figure 23 illustrates the creep frames selected at ORNL for refurbishment. Extensive testing capabilities is required since the test duration required for a single test specimen easily extends to several thousand hours, with a number of tests of tens of thousands of hours or more to support Gen IV reactor design life of 60 years (~526,000 hours). Numerous lessons learned were derived from the review of the HTGR program and will be integrated appropriately into the VHTR program including ceramic retort sensitivity to shutdown and cool-down, metal to metal interaction between metal pull-rods, extensometers, and test specimen, specimen design, and recording of data and data backup methods.

Figure 23. Creep frames selected for refurbishment for environmental creep testing

Progress in initiating scoping tests to provide time-independent input for HTDM constitutive equation development Testing capabilities of the previous HTGR program during the 1980's were evaluated to draw experience for the Gen IV materials testing activities. The evaluation indicated that significant efforts and funds are needed to recover the previous testing scale. Progress and status of the scoping tests on creep properties in both air and impure helium as well as tensile properties at various loading rates were described. Some technical aspects of the testing were illustrated.

FY-06 Activities

This activity will be continued in FY-06 as Activities 207 at the INL and 203-205 at the ORNL to complete the scoping tests and characterization and characterization of controlled material specification (CMS) Alloy 617 and Alloy 230.

3.4.6 Task 2C (INL) Microstructural and Strength Characteristics of Alloy 617 Welds.

FY-05 Activities

The NGNP is being designed as a helium-cooled, graphite-moderated, thermal neutron spectrum nuclear reactor for the demonstration of thermodynamically efficient production of electricity and hydrogen without production of greenhouse gases. For efficient production of hydrogen using a thermochemical cycle driven by nuclear process heat the reactor coolant outlet temperature must be as high as reasonably achievable, ideally in excess of 950°C. Such temperatures will significantly challenge the temperature capability of existing materials of construction, particularly for metallic materials directly exposed to the coolant.

The intermediate heat exchanger (IHX) will perform a critical function in the operation of the NGNP for both direct and indirect cycle applications, transferring heat from the primary reactor helium to a secondary working fluid at a slightly lower temperature. The total operating pressure of the IHX under normal conditions is anticipated to be 7-8 MPa, with a small (0.1 MPa) pressure differential between the primary and secondary legs. Loss of pressure in the secondary leg under off-normal or accident conditions would lead to a full 7-8 MPa pressure drop across the IHX. Current designs call for a compact IHX, with printed circuit, brazed plate-and-fin, and ceramic open-cell foam types suggested. More traditional designs may need to be considered as a backup. High-temperature joints will be a key element of all designs.[35]

High-temperature metallic alloys are the primary candidate materials for the IHX and other components anticipated to operate at temperatures between 800 and 1000°C. Several existing alloys, e.g. nickel-base Alloys 617, 230, and X, are approved for non-nuclear construction under Section VIII of the American Society of Mechanical Engineers (ASME) B&PV design code, but none are approved for construction under the nuclear construction requirements in Section III. A draft code case was developed for Alloy 617 in the 1980's, but the final approval process was not completed. The main data needs identified in review of the draft case include weldment fatigue data, a more complete creep-fatigue database, and a better understanding of the synergistic effects of aging, environment, loading, and temperature.[3,19]

Activities at the INL under the HTDM task of the NGNP materials R&D program are focused on addressing the needs cited above for candidate high-temperature alloys, particularly improved understanding of creep-fatigue-environment interactions in candidate high-temperature alloys and joints made from them. Initial work is being performed on Alloy 617 due to the fact that it appears to be closest to gaining code approval in Section III, but research is also planned on other alloys.

The first task in the study of creep-fatigue-environment interactions in Alloy 617 base metal and joints is 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 [36]. 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. Fusion welding of wrought, solid solution strengthened Ni-base alloys such as Alloy 617 is well established and the weldment tensile properties are known to equal those of the parent metal. The longer-term objective of this project, however, is to study the creep-fatigue properties of these fusion welds, and this has been considerably less well studied. Diffusion bonding is also a well-established technique for Ni alloys, although it can be difficult due to their tendency to form tenacious oxides which prevent diffusion across the bondline. Use of pure Ni or Ni-Cr interlayers is known to assist bondline diffusion [37], and this was shown to be effective in the current study. A potential drawback of an interlayer is its weakness with respect to the base metal, since no strengthening elements are present other than those which diffuse in during the bonding cycle. Measurement of the concentration variation of key elements across the bondline is planned.

Tenacious oxide formation due to the Al and Ti content of Alloy 617 is also believed to be the cause of the poor wetting behavior of the braze joints. Again, application of a layer of pure Ni to the bonding surfaces is expected to prevent oxide formation in the brazing thermal cycle and improve wetting behavior. Due to the high Cr, Si and B content of the braze alloy, the presence of a layer of pure Ni is not expected to present a strength issue, as it does in a diffusion bond. On the contrary, the high concentration of silicides and borides present after brazing may impair the ductility and creep-fatigue strength of the joint by acting as crack initiation sites.

Future work in this project will obviously include tensile testing of both diffusion bonds and braze joints with Ni electroplate; specimen machining is in progress. It is hoped to test diffusion bonds in both butt and scarf joint configurations, similar to the braze joints. Testing in shear with the scarf joint may be more aggressive for the diffusion bond due to the weak Ni interlayer; its weakening effect may be minimized in a butt joint configuration due to the constraint of the surrounding material. Creep-fatigue testing of fusion weld specimens will also soon be initiated.

A similar campaign of joint creation and characterization will be needed for other alloys; the two leading candidates are Haynes 230 and Hastelloy XR. Alloy 230 offers strength similar to Alloy 617 but with less Co (a potential contamination source due to activation). Hastelloy XR is currently used in the Japanese HTTR and contains no Co. Given the multiplicity of alloys, joining techniques, and creep-fatigue test conditions, however, it will certainly be necessary to eventually down-select alloys and joining techniques so that a substantial creep-fatigue database may be generated in a reasonable time on the most promising combinations.

Three types of high-temperature joints were created from Alloy 617 parent metal: fusion welds, braze joints, and diffusion bonds. The microstructures of all joint and tensile properties of fusion welds and braze joints were characterized. The following conclusions were reached:

1. Sound fusion welds were created by the GTAW process with Alloy 617 filler wire. Cross-weld tensile strengths were equal to the parent metal at temperatures of 25, 800, and 1000°C; ductilities of the joints were only slightly lower than the parent metal. Failure occurred in the weld fusion zone at room temperature and in the parent metal at elevated temperatures.

2. Incomplete wetting occurred in joints produced by vacuum brazing using AWS BNi-1 braze alloy, believed to be due to tenacious Al and Ti oxide formation. Incompletely bonded butt joints showed relatively poor tensile properties. A second set of braze joints has been created with faying surfaces electroplated with pure Ni prior to brazing; characterization of these joints is in progress.

3. Conditions resulting in good diffusion bonds characterized by grain growth across the bondline and no porosity were determined: vacuum bonding at 1150°C for 3 hours with an initial uniaxial stress of 20 MPa (constant ram displacement). A 15 μm thick pure Ni interlayer was needed to achieve grain growth across the bondline. Tensile testing of diffusion bonds is in progress.

<u>FY-06 Activities</u>

This activity will be continued as Activities 208 and 209 during FY-06 - produce and characterize the microstructure and strength of CMS Alloy 617 and Alloy 230 joints.

3.4.7 Task 2D (INL), Initiate Creep-Fatigue Testing of Alloy 617 Joints

<u>FY-05 Activities</u>

Creep-fatigue tests on Alloy 617 fusion welds were initiated after repair of the induction power supply on the refurbished environmental test system. Tests were performed at 1000°C in air to enable direct comparison with results on base metal, also in air. Table 18 lists tests performed, cycles to initiation (10% load drop) and failure (25% load drop), along with data for base metal specimens tested as part of another program at equivalent conditions. Although these results are preliminary, the fusion welds appear to have markedly shorter lives than base metal at the lower strain range. At the higher strain range, tensile hold time has more of an effect in reducing creep-fatigue life for the fusion welds compared to base metal. Fatigue lives for the base metal are constant with increasing hold time for holds greater than one minute, while fusion weld fatigue lives continue to decrease with increasing hold time.

Table 18. Test conditions and results for creep-fatigue of Alloy 617 fusion welds at 1000°C in air.*

Test ID	Total Strain Range (%)	Tensile Hold Time (min)	Cycles to Initiation		Cycles to Failure	
			Fusion Weld	Equivalent Base Metal	Fusion Weld	Equivalent Base Metal
IN617-FUS-1000-06	0.3	0	2687	12,300	2958	13,400
IN617-FUS-1000-07	0.3	1	1167	1200	1380	4100
IN617-FUS-1000-03	1.0	0	417	400	545	570
IN617-FUS-1000-04	1.0	1	200	330	228	510
IN617-FUS-1000-02	1.0	3	179	240	204	400
IN617-FUS-1000-05	1.0	10	97	240	119	400

* All tests performed with triangular waveform, strain rate $1x10^{-3}$ sec-1, fully reversed loading, tensile strain hold.

FY-06 Activities

Creep and creep-fatigue testing of standard Alloy 617 base metal and joints with be continued in FY-06. Similar tests on CMS Alloy 617 will be initiated. This work will be performed as a part of Activity 210.

3.4.8 Task 2E (INL), Initiate Aging of Base Metal and Weldment Specimens

FY-05 Activities

A purchase order was placed early in FY-05 for a box furnace suitable for long-term use at 1000°C in an air environment. The box furnace in which the long-term aging exposures will be carried out was received and set-up. The initial matrix of exposures was established and material submitted for machining into coupons. Table 19 shows the initial aging test matrix.

Table 19. Exposure matrix for Alloy 617 for 1000°C aging

Aging Time (hr)	¾ Inch Plate			½ Inch Plate
	Microstructure	Tensile	Impact	Microstructure
30	√			√
100	√	√	√	√
300	√			√
1,000	√	√	√	√
3,000	√			√
10,000	√	√	√	√

Material exposed for 30 hr was removed and submitted for microstructural evaluation, which will include measurement of oxidation extent to confirm the suitability of air exposure to assess aging effects without an influence of environment; no tests have been performed on this material to date. Material exposed for 100 hr was removed and submitted for microstructural evaluation; the microstructural evaluation has not been performed to date. Material exposed for 300 hr was removed and submitted for microstructural evaluation; the microstructural evaluation has not been performed to date.

A additional 250 hr exposure of machined tensile specimens was started. These specimens will be tested after exposure at a range of temperatures to provide an initial assessment of potential oxygen embrittlement of Alloy 617 during high-temperature air exposure. Pure nickel and some other Ni-base alloys are embrittled by exposure to oxygen-bearing environments at and above 1000 °C due to oxygen diffusion along grain boundaries[38]. The zone of embrittlement can extend well beyond the visible extent of oxidation. Since aging exposures of Alloy 617 was performed in air on the assumption that the alloy is not affected by the environment other than a small zone of surface oxidation, it is important to determine whether this alloy is susceptible. Tests of the pre-exposed specimens were performed at 25, 700, 800, 900, and 1000 °C (strain rate 1x10-3 sec-1); test results are given in Table 20. No sign of significant embrittlement due to the air exposure were observed, either in tensile ductility or fracture surface appearance. The reductions in ductility observed at room temperature and 800 °C are small and could result from the brittle oxide scale formed in air exposure. Air exposure embrittlement, as observed in other nickel-base alloys, would have been manifest as a marked reduction in tensile ductility after exposure, particularly at intermediate temperatures (600-900 °C), coupled with an intergranular fracture mode[38]. Examination of metallographic cross-sections of the tested specimens is planned to confirm this result. Material exposed for 1000 hr was removed; no tests have been performed on this material to date.

Table 20. Tensile properties of Alloy 617 after 250 hr exposure in air at 1000°C. Baseline data shown for comparison.

Test/Specimen ID	Condition	Test Temperature (°C)	Yield Stress (MPa)	UTS (MPa)	Ductility (%)	RA (%)
IN617-1/2P-T-01	Baseline	25	363	807	62	50
IN617-1/2P-T-02	Baseline	25	378	809	65	50
IN617-1/2P-T-05	250hr 1000C	25	395	829	47	32
IN617-1/2P-T-06	250hr 1000C	700	215	608	56	43
IN617-1/2P-T-07	250hr 1000C	700	208	613	58	42
IN617-1/2P-T-03	Baseline	800	257	447	62	82
IN617-1/2P-T-08	250hr 1000C	800	207	439	59	65
IN617-3/4P-T-04	Baseline	900	253	268	57	86
IN617-3/4P-T-05	Baseline	900	254	259	66	87
IN617-1/2P-T-09	250hr 1000C	900	215	261	59	80
IN617-1/2P-T-04	Baseline	1000	146	154	67	84
IN617-1/2P-T-10	250hr 1000C	1000	140	147	70	94

FY-06 Activities

All tests not performed in FY-05 will be completed according to the matrix in FY-06. Creep-fatigue testing of base material aged for 10,000 hours will be performed in FY-06 in Activity 210.

3.4.9 Task 2D (ORNL), Perform Simplified Methods Development

FY-05 Activities

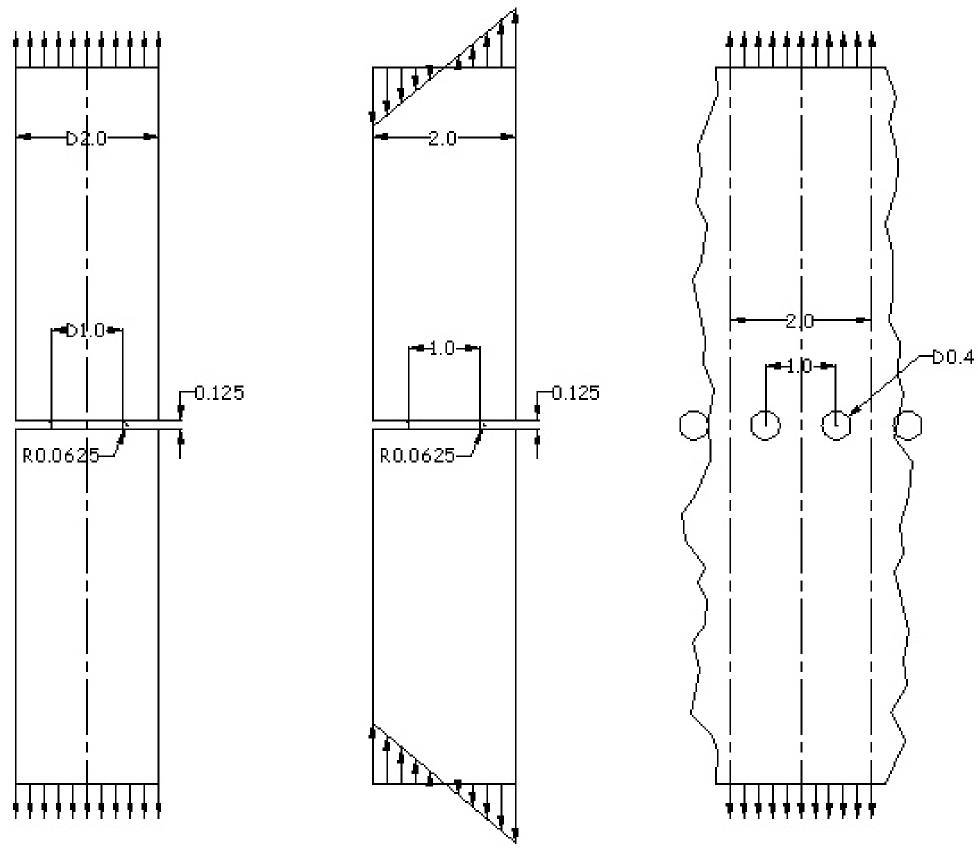
High temperature design methodology (HTDM) includes the integration of simplified design methods and material data generation towards development of ASME B&PV Code for elevated temperature design procedures that address time-dependent failure criteria and assure adequate life. Data generation includes 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.

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.

The very high operating temperatures of the VHTR requires use of new materials and extension of elevated temperature design methodologies far beyond the range that existing ones cover in terms of operating temperature, service duration, environment, etc in ASME Section III, Subsection NH. In order to implement new materials and design methodologies into codes and standards subject to authorization by regulatory bodies such as the NRC, validation tests using structural and/or component models are required. Test program development is required to ensure design procedures address all pertinent failure modes, and that the procedures are adequate. Particular attention must be paid to capturing phenomena that VHTRs may exhibit where prior accumulated experience from design and operation of existing plants do not address, e.g. very rate and time dependent material behavior at very high temperature, irradiation in helium environment and failure modes and degradation mechanisms foreseen associated under such extreme conditions. This will also likely include safety assessment of time-dependent flaw growth and resulting leak rates from postulated pressure-boundary failure.

The process of development of high temperature structural design procedures requires multiple iterations, and will require the constant integration of concepts, ideas, experimental observations, and analysis that results from both numerical and analytical analysis and design approaches, coupon testing of materials, and structural features testing. Hence, efforts in 2005 were focused at the development of simplified methods, re-establishing very high temperature testing capabilities, and the organization and planning of detailed activities and research in support of code needs in support of Gen IV reactors.

An evaluation was conducted of the application of the load based design criteria found in NH to a VHTR. Comparisons with life predictions using isochronous curves, a creep model including "damage" effects (an Omega model), and the limit load reference stress were made on various notched samples, plates, beams, and pressure vessel components of Alloy 617 at 900 °C. Figure 24 illustrates several of the simply notched structures investigated. Figure 25 illustrates the dimensions and cross-section of a sphere/nozzle and cylinder/nozzle intersection that was analyzed; these types of models were used during the LMFBR program for validation of design procedures.



axisymmetric “yoyo” notch plane strain notch in bending plane strain ligament in tension

Figure 24. Dimensions (inches) of several simply notched structures

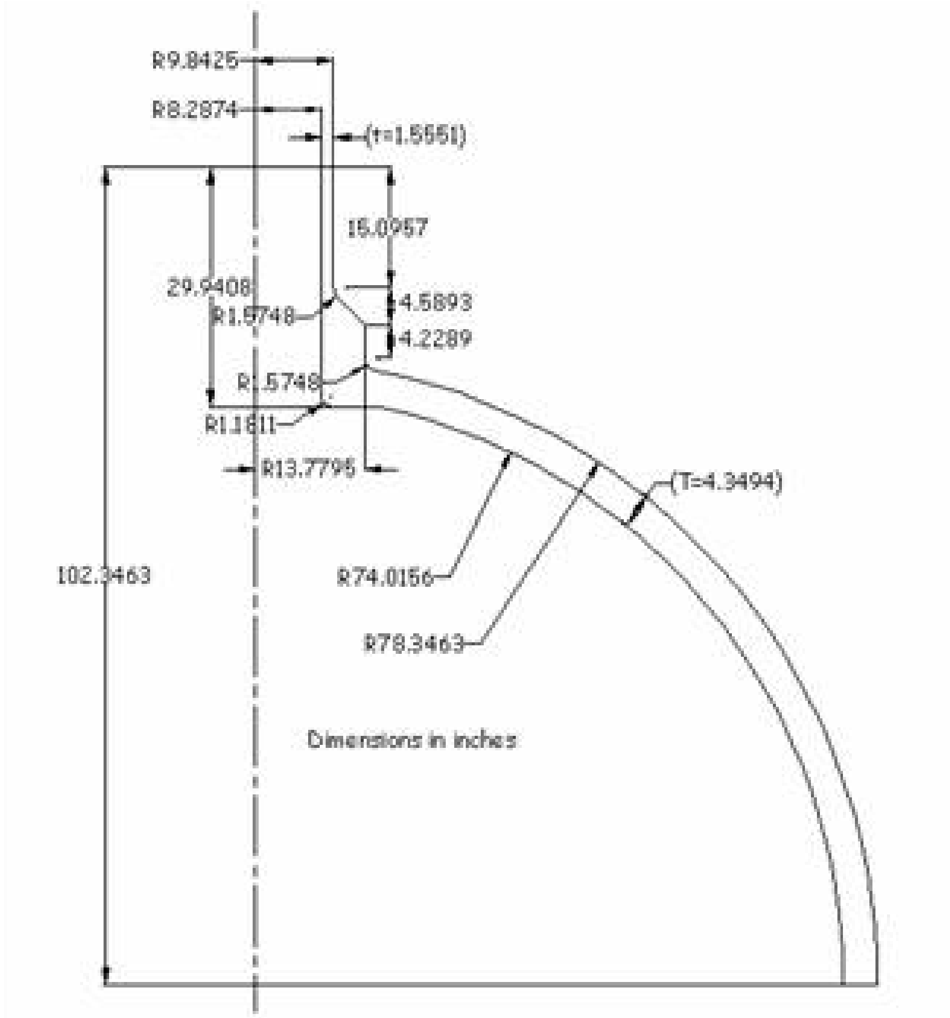
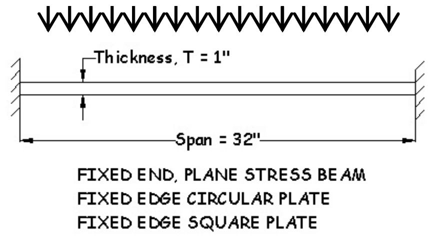


Figure 25. Dimensions (inches) and cross-section of sphere/nozzle and cylinder/nozzle intersection

An illustration of simple beams, plates, and flathead structures with uniformly distributed loads that were investigated is provided in Figure 26. The load based criteria in NH were found to be conservative; however, they were found to be excessively conservative in cases where redistribution of stress occurs during creep. This is illustrated in Figure 27 where the NH predictions deviate from the predictions of several other types of analysis. The NH procedures only deal with relaxation within a section; no allowance is included for the possibility that section bending and membrane forces also undergo long range relaxation. This is clearly evident in beam, plate, and flat head problems analyzed in Figure 27. Furthermore, stress linearization was problematic and resulted in an overly conservative life prediction in the case of a thick tee under internal pressure. Existing NH load based design criteria are deemed acceptable. Other load based design methods that use isochronous curves or the reference stress approach are proposed as alternatives; these methods also eliminate the need for stress classification and may have great value for core internals and various attachments. Additional research in this area should be pursued, including experimental validation of the analysis predictions.



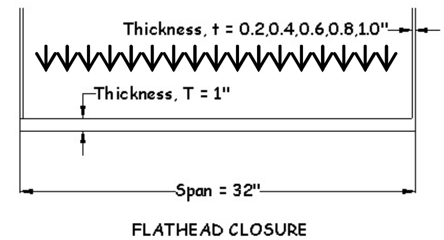


Figure 26. Beams, plates, and flathead structures with uniformly distributed loads were investigated (dimensions in inches).

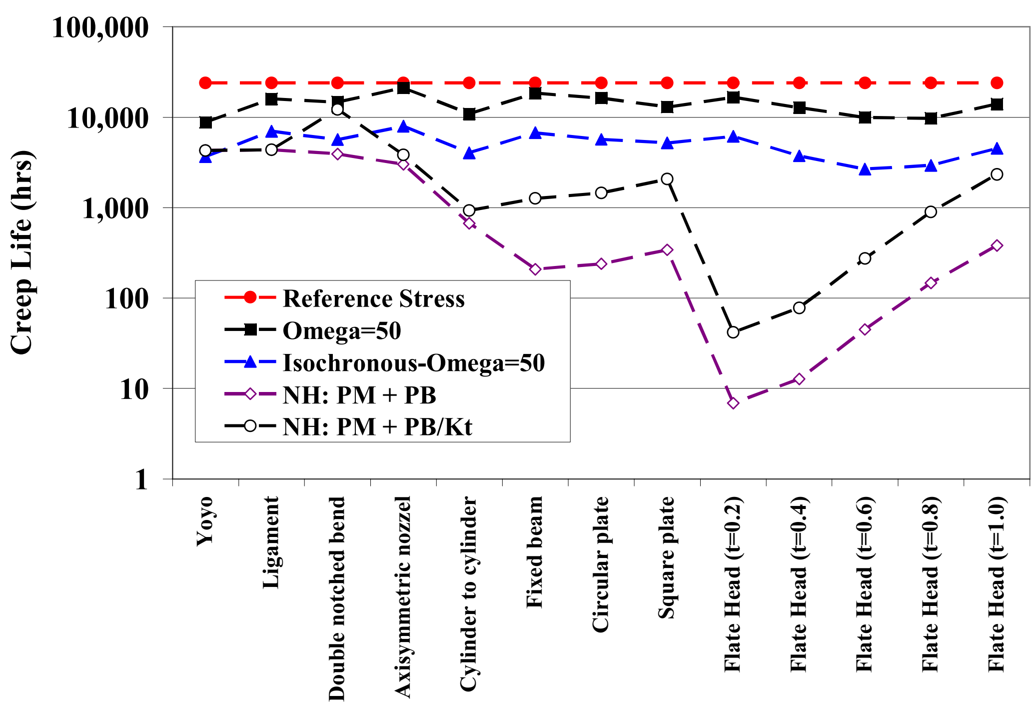


Figure 27. Comparison of predicted creep lives at constant reference stress for notched specimens, pressure vessel components, beams, and plates

NH deformation based design criteria were also evaluated and compared to cyclic reference stress approaches. The B-1 test failed to predict ratcheting for a simple thin tube of Alloy 617 under static pressure and cyclic thermal loading from 400 °C to ~900 °C. Additional analysis of a thick tube and thick tee joint were conducted. In all cases, the normalization procedure required in implementing the B-tests in Appendix T is questionable due to significant variations in the yield strength with temperature. A modified B-test(s) may be required. Normalizing by either (a) the average of SyL and SyH , (b) SyH , (c) an appropriate yield stress from isochronous curves, or (d) some other appropriate yield stress, remains uncertain – additional research is required. The effect of loading rate on yield stress was not examined; additional research is required here as well. Alternative procedures that eliminate the need for stress classification should also be considered, similar to the findings of the load based design criteria in this report. Alternatives include methods that depend upon the geometry of the component, and are typically implemented with finite element methods. The alternatives include: (a) using cyclic reference stresses with a constant yield stress, (b) using cyclic reference stresses where the fictitious yield stress varies with temperature, (c) performing rapid cycle analysis with temperature dependent properties, and (d) the use of isochronous curves (monotonic or cyclic). Such approaches were unrealistic at the time of the development of NH; however, today's tremendous computational power enables these methods to be entertained as a routine analysis tool. A significant amount of additional research in this area remains.

Additional integral parts of simplified methods development and verification are materials & structural feature testing and constitutive modeling. While strictly speaking a part of the Cross Cutting program, a brief summary of efforts in this area is summarized here due to the overlapping technologies and goals. To this end, a creep-fatigue machine was re-established for testing at very high temperatures; Figure 28 is an illustration of a test conducted at 950 °C. The limited test results to date on Alloy 617 at 950 °C are in agreement with literature data. Software to assist in the development of constitutive models that can predict the stress-strain history of material and structures under complex thermo-mechanical loading was identified and purchased; tensile and fatigue scoping tests are underway and/or planned to provide data for use in development of an initial experimentally based constitutive model for Alloy 617. The model should be capable of predicting cyclic hardening/softening, strain rate effects, and aging effects on material behavior. Constitutive model development of additional alloys such as Alloy 230 and Mod9Cr1Mo are planned in future years.

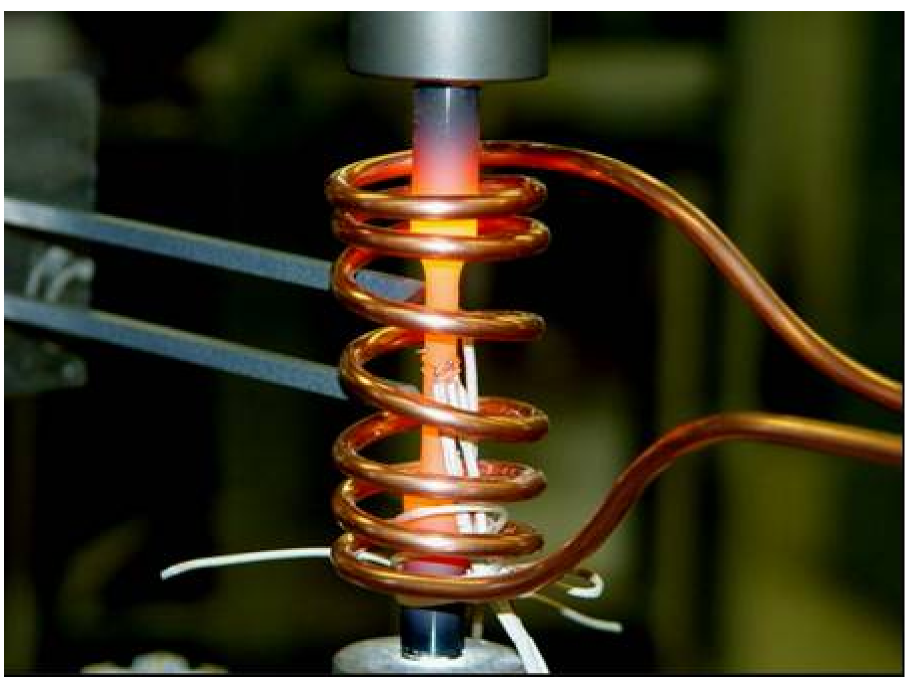


Figure 28. Picture of creep-fatigue test system in operation at 950 °C

The work performed for simplified methods development is documented in *Simplified Design Criteria for Very High Temperature Applications in Generation IV Reactors* ORNL/TM-2004/308, Revision 1[39] and in *High Temperature Design Methods Development Advances for 617: Status and Plans,* ORNL/TM-2005/515[40].

<u>FY-06 Activities</u>

This work will be continued in FY-06 under Activities 207-209, methods development for very high temperature metallic design.

3.4.10 New Activities to be Initiated in FY-06

The following task areas are in the base budget for FY-06:

- Activities 201-203 (INL), Procurement of a new servo-hydraulic load frame
- Activities 204-206 (INL), Procurement of Alloy 230 for testing at the INL and ORNL
- Activities 201 and 202 (ORNL), Procurement of CMS Alloy 617 for testing at the INL and the ORNL
- Activities 204 and 205 (ORNL) and Activities 207-209 (INL), characterization and scoping tests on the alloys procured above
- Activity 206 (ORNL), performance of environmental scoping tests at elevated temperature on the alloys noted above

3.4.11 Other Activities that are Currently Unfunded in FY-06

The following task areas are currently not in the base budget for FY-06 (see Table 21); however, there is merit in funding all or part of these activities when funding becomes available:

Table 21. Task areas are currently not in the base budget for FY-06

Task Title	Task Description	Benefit to NGNP Program	Total Funding Request, $K	Funding Split, $K	
				ORNL	INL
Metallurgical study on Alloy 617	A continuation of FY-05 Task 3A - Development of a Nuclear Specification for Alloy 617. The nuclear specification is based on analysis of available existing data. Such data are not sufficient to provide a thorough understanding of detailed strengthening mechanisms, thermodynamics, kinetics, microstructure evolution, and many other metallurgical aspects of the alloy under the envisioned NGNP service conditions. The investigation is important for life prediction of the material as well as verification and possible modification of the specification developed in FY-05.	Provide improved properties and reduced variability and scatter in elevated materials mechanical properties of one of the prime candidate materials for the IHX	**200**	**200**	
Investigation on weldability of Alloy 230	Alloy 230 is a prime candidate for VHTR service above 760ºC (1400ºF). Welding technology on Alloy 230 is currently underway at ORNL under the DOE Ultra-Super Critical Steam Boiler Materials Project. Currently the investigation focus is on welds produced using Pulsed Gas Metal Arc Welding Process for service temperature up to 800°C. Collaborative studies with diffusion bonding work at INL and ongoing USCSBM project are needed to: 1) Investigate the phase stability of autogenous and heterogeneous weld microstructures and their effects on the performance of welded structures; 2) Use computational and characterization capabilities in an integrated thermo-mechanical-metallurgical approach to develop optimum welding process parameters and weld filler metals compatible with Alloy 230 that match performance characteristics for temperatures above 800°C; 3) Develop welding technology to eliminate micro-fissuring/cracking in multi-pass weldments.	Develop required joining technology needed for both thin and thick sections one of the prime candidate materials for the IHX to enable fabrication of compact heat exchangers	**400**	**400**	
Environmental creep testing	A continuation of FY-05 Task 3F – Initiate creep testing of Alloys 617 and 230 in a controlled helium environment. The refurbishment of two environmental creep machines FY-05 included the procurement and installation of the overall systems for data acquisition and control and for environmental control. At least another 5 environmental creep machines must be refurbished in FY-06 and testing initiated on NGNP prime candidate IHX materials.	Develop data needed for assessment of environmental affects required by the ASME code for NGNP high temperature metallic components	**600**	**600**	
Continued Development of Materials Input for Constitutive Equations	A continuation of FY-05 Task 2C –Re-establish 8 very high temperature uniaxial creep frames (air) for generation of creep curves for use in constitutive modeling of very high temperature alloys, e.g. 617, 230, XR, etc. and conduct uniaxial creep and tensile testing in air for 617 and/or 230 to generate data needed for constitutive modeling be examined. A duplicate system will be established at INL	Develop critical experimental data needed for development of simplified high temperature design methods used by the ASME code for NGNP high temperature metallic components	**350**	**350**	**0**

Enhanced Simplified Methods Development	A continuation of FY-05 Task 2D – Expand on the development of simplified design methods for deformation based design criteria for VHTR materials & applications	Develop improved simplified high temperature design methods needed for design and analysis of NGNP high temperature metallic components	**450**	**450**	
Negligible creep of Mod 9Cr-1Mo steel at low temperature range	It is assumed in ASME Code that creep only becomes significant at temperatures above 371ºC (700ºF) in ferritic/martensitic steels. As a matter of fact, negligible creep may occur below 371ºC (700ºF). For the 60 years of design life of the VHTR RPV, such creep elongation may accumulate and become none negligible. Tests will be initiated to investigate the phenomenon for a better understanding of long-term material behavior and to provide applicable data for the RPV design. Due to the long-term testing time required, the testing should be started as early as possible to obtain results before component design begins. Due to its low temperature and long-term testing cycle, at least 10 machines are needed.	Provide critical experimental data to assess adequacy of ASME Code rules on temperature and loading at which creep must be considered for VHTR pressure boundary materials	**600**	**600**	
Supplemental funding to facilitate a closer collaboration with the U-NERI programs related to the VHTR	Close collaboration with universities involved with R&D associated with the NGNP materials program would be very useful for information sharing, setting program direction and discussion of materials R&D issues of common interest.	Currently, there is one U-NERI program at the University of Michigan associated with alloy development of high temperature alloys development, however, other programs are expected in FY-06. This task would provide funding for close collaboration with these programs as needed.	**50**	**25**	**25**
Supplemental funding to establish direct contracts with universities to assist with HTDM development	Close collaboration with select universities with specific expertise in the areas of high temperature alloys development and HTDM would assist in the development of the NGNP Materials Program.	This task would provide funding to establish collaborations with selected universities by technical meetings or other mechanisms.	**50**	**25**	**25**
Additional laboratory equipment to support high temperature testing	Additional creep-fatigue testing equipment would allow the generation of a greater amount of data that requires very long time periods.	This task would allow the purchase of a second new environmental chamber at the INL.	**400**		**400**

The initiation of a study of welding and through thickness properties for Grade 91 and similar Class 1 pressure vessel steels	Currently, only Framatome/AREVA/CEA are performing welding and through wall thickness studies on thick sections on alloys for RPV applications for the VHTR. It is essential that a program be established in the US to perform R&D in this area.	This task would initiate a program to begin to address these R&D areas.	**800**	**400**	**400**
The development of a prototype component loop testing facility for testing prototype valves, piping, insulation, recuperators, IHX units and other components under VHTR prototypic conditions	Currently, only Framatome/AREVA/CEA are performing loop testing or developing loop testing capability for the purpose of subjecting prototype VHTR components to expected VHTR in-service conditions. Developing the capability to test these components is critical for the US NGNP program to collaborate with the French in this area and to develop an understanding of the complex stress state and thermal hydraulic issues that these components will be subjected to in service.	This task would initiate the development of a test loop capability in the following areas: test loop investigation and the development of a conceptual design and interface with selected universities and component manufacturers and develop test specifications and bid specifications for prototype components to be tested.	**700**		**700**
		TOTALS	**4,600**	**3,050**	**1,550**

3.5 ASTM and ASME Code Support

Currently there are many areas relating to ASTM standard method development and ASME B&PV Code development that need to be pursued to meet NGNP goals. The NGNP Materials R&D Program has initiated a presence at the relevant ASTM and ASME B&PV Code committee and subcommittee level to be able to incorporate new materials or extend the application of materials presently in the Code or existing test standards. Personnel will continue to support appropriate committees and develop required standards and validation testing as required.

3.5.1 Task 3A (INL and ORNL): Support of ASME Section III, Subsection NH, Subgroup on Elevated Temperature Design.

FY-05 Activities

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 [41]. 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 already compiled within crosscutting activities and documented in the *Updated Generation IV Reactors Integrated Materials Technology Program Plan, Revision 1* [42].

At this time, funding up to $1M for these supplementary tasks is anticipated to be provided by DOE, directly to ASME, to accelerate the development of required aspects of codification for critical components and high temperature materials. The ultimate goal is the development and acceptance of ASME Section III Code that will assist stakeholders in obtaining licenses for the design, construction, and operation of Gen IV reactors. The agreement between DOE and ASME is in the final stages of negotiation. This effort is expected to start in FY-06 and extend for a total of three years; however, only the first year activities are funded in the current budget. A brief summary of relevant tasks is provided below. Further details are available in *High-Temperature Design Methods Development Advances for 617: Status & Plans*, ORNL/TM-2005/515 [40]. This plan was presented and discussed at the August 2005 ASME NH meeting.

3.5.1.1 Verification of Allowable Stresses in ASME Section III, Subsection NH with Emphasis on Alloy 800h And Grade 91 Steel (9CR-1MO-V or Modified 9cr-1mo) [Current plans are to fund in FY-06 to completion]. Currently, five materials are approved for the construction of Class I nuclear components other than bolts under the rules of ASME Section III, Subsection NH (III-NH). Two of these materials, namely 800H and 9Cr-1Mo-V steel, are candidates for the construction of components for the VHTR concept included in the Generation IV Nuclear Reactor Program. The major research that produced the database for these materials was undertaken in the 1970s and 1980s. Since then, considerable long-time experience has been gained for both materials and data analysis methods for setting the allowables have been refined. These actions have produced changes in both the time-independent and time-dependent allowable stresses in ASME Section II for Sections I and VIII, D1. There is a need to review these changes and their impact on the allowable stresses in III-NH.

3.5.1.2 Regulatory Safety Issues in Structural Design Criteria of ASME Section III Subsection NH and for Very High Temperatures for VHTR & Gen IV [Current plans are to fund in FY-06 to completion]. The NRC has not accepted (or rejected) Subsection NH of Section III of the ASME Code "Class 1 Components in Elevated Temperature Service." Further, the Advisory Committee on Reactor Safeguards (ACRS) reviewed similar elevated temperature structural design criteria proposed for the Clinch River Breeder Reactor (CRBR) and generated a list of technical issues and safety concerns which they believed still needed to be resolved [NRC 1983]. DOE agreed to fund R&D efforts to answer their concerns to the satisfaction of the U.S. NRC and the ACRS prior to requesting an Operating License for CRBR. The structural design criteria being used at that time was fundamentally similar to the current criteria in Subsection NH of Section III of the ASME Code. A paper on the NRC review summarized the situation as follows: "In a general sense, the NRC review of the CRBRP confirms the adequacy of the high-temperature structural design methodology that has been developed over the last 20 years…" and "The design criteria and basic approach to design evaluation have been accepted, and no major inadequacies were discovered. The review identified and resolved a number of issues relative to Code interpretation, and it identified areas where more detailed evaluation techniques would be useful. The required confirmatory programs would both improve design assurance of the CRBRP, and simplify design and evaluation of future plants." [Griffin 1985]. The four major areas of concern were 1) weldment safety evaluation, 2) notch weakening, 3) design analysis methods, codes, and standards, and 4) adequacy of tube sheet designs for the steam generator. The programs that were developed to address these concerns were not conducted when the program funding was terminated. It is clear that the confirmatory programs need to be completed. Assessment and identification of additional possible safety issues relative to Gen IV, and specifically VHTR, are needed. Ultimately, any safety issues need to be resolved from a regulatory perspective in order to assure that the technology needed to support the licensing of VHTR and Gen IV will be in place to support Design Efforts in a timely manner.

3.5.1.3 Improvement of ASME Subsection NH Rules for Grade 91 Steel—(negligible creep and creep-fatigue) [Current plans are to fund in FY-06 to completion]. Mod9Cr1Mo (Grade 91) is a candidate for the Reactor Pressure Vessel of VHTR and is also thought to be a potential candidate as a material for internals. Two important issues related to the use of Mod9Cr1Mo exist: negligible creep and creep-fatigue.

For the RPV, the issue to be addressed is related to the definition of negligible creep conditions. This need is linked to the choice to operate the RPV in the negligible creep domain so as to avoid the implementation of a surveillance program in the significant creep regime. This point is all the more important in that there is interest to increase the value of the core inlet temperature.

For internals, the major concern is creep-fatigue. Procedures are available in nuclear Codes (ASME, RCC-MR, etc.) to cope with creep-fatigue but most of those procedures have been established for austenitic stainless steels and do not necessarily take account of peculiarities of martensitic steels such as Mod9Cr1Mo (e.g. softening and elastic-relaxation behavior). There is therefore a need to compare existing procedures and to confront numerical application with experimental results. A specific point to investigate is the definition of the creep-fatigue damage envelope for which significant differences are found from one procedure to another.

3.5.1.4 Updating of ASME Nuclear Code Case N-201 to Accommodate the needs of Core Support Structures in High Temperature Gas Cooled Reactors Currently in Development [Current plans are to fund in FY-06 to completion]. ASME Nuclear Code Case N-201 contains rules for construction of core support structures under Subsection NG for service at elevated temperatures. The rules of this Code Case are similar to those contained in Subsection NH, Class 1 Components in Elevated Temperature Service. Both Subsection NH and Code Case N-201-4 were developed before the requirements for Gen IV VHTRs were known and therefore require additions or amendment to be of value in the design and construction of the currently proposed VHTRs.

Part A of the current Code Case N-201-4 provides design rules for the construction of core support structures fabricated from five materials: ferritic steels 1 Cr-0.5 Mo-V and 2.25 Cr-I Mo, Type 304 and 316 stainless steel (SS), and Alloy 800H. Part A applies at times and temperatures where creep effects do not need to be considered. For Part B of the Code Case, "Rules for Construction of Subsection NG, altered for service at elevated temperature to suitably account for creep and stress-rupture effects," the permissible materials are limited to four, 2.25 Cr-I Mo, Type 304 and 316 stainless steel (SS), and Alloy 800H, with varying maximum permitted temperatures for use.

For construction of VHTRs with core outlet temperatures of 900 to 1000°C, the maximum permitted temperature of 815°C (1500°F for SS 304 and 316) and 760°C (1400°F for alloy 800H) these materials cannot be used when exposed to temperatures at or near the core gas outlet temperature. The scope of the code case needs to be expanded to include the materials with higher allowable temperatures or extend the temperature limits of current materials and to confirm that the design methodology used is acceptable for design of core support structure components at the appropriate elevated temperatures.

3.5.1.5 Collect Available Creep-Fatigue Data and Study Existing Creep-Fatigue Evaluation Procedures for Grade 91 Steel and Hastelloy XR [Current plans are to fund in FY-06 to completion]. Creep-Fatigue is a failure mode of great concern for reactors operated at elevated temperatures. ASME Section III Subsection NH incorporates procedures for creep-fatigue damage evaluation, which is one of the major features that distinguish it from other parts of Section III. NH deals with such materials as conventional steels, Mod9Cr-1Mo and Alloy 800H. Temperature range and service duration covered in the code vary in range of temperature and time, up to 750℃ and approximately 34 years, respectively.

There are noticeable deviations between what are required in the design of Gen IV and VHTR reactors and what the current NH covers. Structural materials of primary choice in Gen IV and VHTR reactors are Mod9Cr-1Mo and Hastelloy XR. Alloy 617 and Haynes 230 are also candidate materials similar to Hastelloy XR. Gas temperature ranges expected in the current design study are up to 600°C or higher for Mod9Cr-1Mo and 950°C for Hastelloy XR; various design strategies will lower the actual metal temperature to varying degrees. However, components such as the intermediate heat exchanger will experience the full gas temperature. Design life for the reactor is 60 years. Mod9Cr-1Mo has recently been incorporated in NH, while Hastelloy XR, Haynes 230, and Alloy 617 have not been incorporated yet (a draft code case for Alloy 617 exists). Temperature range and design life are well above the range covered by the current NH. Some experts consider the current creep-fatigue criteria for Mod9Cr-1Mo in NH to be overly conservative because the limits are based on the interim results of Clinch River project. The project was interrupted many years ago when a good understanding of creep-fatigue in Mod9Cr-1Mo had not been achieved; consequently, the interaction diagram was intentionally constructed to err on the conservative side until the need (and associated funding required) to better understand the interaction arose. Nothing has been prepared for creep-fatigue evaluation of Hastelloy XR, and Haynes 230. The degree of conservatism and methods used in the creep-fatigue procedure for Alloy 617 in the draft code also requires a critical review.

Considering the gap between the basis for creep-fatigue procedures in NH and that needed in Gen IV & VHTR, creep-fatigue data acquisition and establishment of better creep-fatigue criteria for primary materials (Mod9Cr1Mo and Hastelloy XR, Haynes 230, and Alloy 617) are desired. However, because performing material tests from scratch needs tremendous money and time, it is appropriate to start with analyzing existing data and creep-fatigue criteria. Therefore, collecting creep-fatigue data on Mod9Cr-1Mo and Hastelloy XR and studying existing creep-fatigue evaluation procedures, which will lead to identification of R&D items in the near future, are required.

3.5.1.6 NH Evaluation and Simplified Methods [current plans are to initiate this task when funds become available]. McGreevy et al addressed the need for simplified inelastic design methods, and future directions [40]. However, closely linked to these methods is the development of creep-fatigue design and assessment procedures. While activity in this area has already been indicated in the previous task, additional activity in this area is required. The activity should include the review of creep-fatigue methodologies, including crack growth, damage-based and strain-based methods. Likely sources will include GE Report DOE-ET-34202-80 and ORNL-5073. Identify applications and areas of difficulty in connection with Grade 91 steel and Alloy 617/230/800H materials. Include aging, crack initiation, surface and environmental effects on these materials. Critically evaluate data and methodology in the light of likely VHTR cycles and assessment requirements. The report will comment on the adequacy of existing methods and will include recommendations to address problems. These could include life prediction models, extrapolation of data, test data and techniques. This activity will not be conducted in vacuum relative to other activities that address creep-fatigue, rather it serves as a parallel but non-duplicate path at addressing creep-fatigue. Addressing such a complicated problem with several different concepts is desired.

3.5.1.7 Identifying Future Test Needs to Validate Elevated Temperature Design of VHTR [Current plans are to initiate this task when funds become available]. VHTR/PBMR has features that no preceding reactors have had. Very high operating temperatures is one of those features and this requires challenging tasks such as development of new materials and extension of elevated temperature design methodologies far beyond the range that existing ones cover in terms of operating temperature, service duration, environment, etc in ASME Section III.

To implement new materials and guide the development and verification of new design methodologies for codes and standards subject to authorization by regulatory bodies, validation tests using structural and/or component models are indispensable. This includes changes in design margins, constitutive equations, and design methods. Test programs should be developed to ensure complete validation of points of concern in the design of VHTR/PBMR, particularly focusing on phenomena of which not enough experience has been accumulated through operation of existing plants, such as very high temperature, irradiation in helium environment, and failure modes and degradation mechanisms foreseen associated with them.

Structural and/or component tests are usually very time consuming and costly. In the development of test programs it is strongly desired that the programs should be developed based on thorough information on what has been accomplished in the past to support the validation, and to identify what has not been addressed or failed to be adequately addressed. Therefore, identifying future test needs by reviewing knowledge and information on what has been accomplished so far is required.

3.5.1.8 Environmental and Neutron Fluence Effects in Structural Design Criteria of ASME Section III Subsection NG & NH and for Very High Temperature VHTR & Gen IV Designs [Current plans are to initiate this task when funds become available]. Subsection NG of the ASME Code for Nuclear Components: “Class 1 Components in Elevated Temperature Service,” does not cover either environmental (corrosion) effects or the effects of irradiation. Moreover, the extension of the design criteria to the higher temperatures (950°C) needed for VHTR and Gen IV reactors introduces much more aggressive environmentally assisted cracking (EAC) issues. It has been the policy of ASME Codes on new construction not to include environmental effects. Recently, however, the ASME Code Subgroup on Fatigue Strength developed proposed new reactor water environmental fatigue design curves. The technology supporting this development is concerned with quantifying the detrimental effects of corrosive attack as a function of the corrosion potential and mechanism, temperature, and strain rate, etc. Crack growth rates are increased by factors of 10 to 50 for carbon, low alloy and stainless steels vs. the crack growth rates in air.

The effects of irradiation have been considered in the design criteria used for reactors, and also in the design of nuclear fuel elements. The strains resulting in creep tensile instability cracking are greatly reduced by irradiation effects. The strain hardening capacity of structural materials is reduced, thereby allowing strain concentrations along very narrow, shear bands or slip lines where the strains are in order of magnitude higher than calculated using continuum mechanics. Tests on fractured irradiated materials show that the strains immediately adjacent to the cracks can be 10 to 100 times higher than the average or continuum strains. As a result, cracking in irradiated materials occurs at calculated creep strains of 1 to 5 percent, where the actual local shear strains are near 100 percent.

The goal of this task is to initiate action to address environmental and neutron effects from a Design Code viewpoint, and to formulate supplemental rules and criteria applicable to VHTR concepts.

3.5.1.9 Development of ASME Code Rules for the Gas Cooled Reactor Intermediate Heat Exchanger (IHX) [Current plans are to initiate this task when funds become available].

"Needs for Intermediate Heat Exchanger (IHX)" has been ranked as a priority item by AREVA to support the VHTR program and appears on the list of items generated by the Board of Nuclear Codes and Standards (BNCS) New Reactors Task Group. From the standpoint of elevated temperature design, the critical section of the IHX is the internal heat transfer matrix. Generally, the outer shell is designed as the primary pressure retaining member and is maintained at a temperature cool enough to minimize creep effects. The inner, heat transfer matrix is, however, exposed to the full reactor outlet temperature. This matrix also serves as the boundary between primary and secondary coolant so it does have a pressure boundary function even though it is not exposed to the full pressure differential between the gas and atmospheric pressure.

Since the heat transfer matrix is not part of the external pressure boundary, and designs will likely include an isolation valve to isolate any failure of the IHX to the nuclear plant, and not a hydrogen plant, one could question the need for ASME Code rules to cover this structure. When this issue was raised with potential reactor system suppliers they reiterated the importance of Code coverage from both the standpoint of achieving a reliable design and also protecting the secondary circuit from contamination from the gasses in the primary circuit. There is also a precedent with ASME Section VIII tube and shell heat exchangers where the tubes are designed as a pressure boundary in accordance with the Code.

The intent of this task is to determine how and where within ASME codes and standards the IHX, safety valve, etc. would be addressed. In order to answer this question, many technical questions need to be addressed to determine how the function of such components affects the plants, safety, etc. While the strict timeline for construction of a reactor with an IHX calls for immediate activity in this area, the level and type of effort, including necessary discussions of many details related to manufacturing, design, and operation of an IHX requires commitment on behalf of the DOE, reactor firms, and IHX manufacturers, and ASME. As such, activities in this area will likely be on hold until that time.

3.5.1.10 Flaw Assessment and Leak Before Break (LBB) Approaches in ASME [Current plans are to initiate this task when funds become available]. In the current version of ASME-NH, little information is given on how to address flaw assessment in the elevated temperature domain. Actions have been carried out in Europe to cover this topic and these actions led to the writing of rules in UK (R5, R6) and to French rules in the RCC-MR, Appendix A16 [R5 2003, R6 2003, RCC-MR 2002]

In addition, Leak Before Break approaches can provide useful arguments in the frame of defense-in-depth analyses which are aimed at demonstrating the robustness of a given design. LBB methodologies have been developed for Pressurized Water Reactors (PWRs) and Fast Reactors (FRs) but their application to High Temperature Reactors (HTRs) and VHTRs would require further investigations.

The objective of this activity should be to perform a status report of rules presently available and to propose recommendations for further work within ASME. The work should consist of a synthesis of approaches available for LBB assessment and more generally for fracture mechanics methods (crack growth and stability calculations). The work should clarify to what extent existing methods would be applicable for VHTR and Gen IV applications.

The output of the task would be recommendations for the definition of rules to be introduced in the ASME Code. A program would be defined indicating necessary tests to be carried out to establish a set of material properties for flaw assessment methods and/or specific tests to validate LBB approaches for HTRs and VHTRs. The results will be useful in discussions with USNRC before launching significant activities on this subject.

FY-06 Activities

FY-06 participation in ASME subsection NH will continue under ORNL Activity 304 and INL Activity 303.

3.5.2 Task 3B1 (INL and ORNL): Support of ASME, Section III, Working Task Group on Graphite Core Support Structures

FY-05 Activities

Both INL and ORNL staff participated on the ASME Graphite Project Team on Core Supports. Previously, ASME Section III Division 2 Subsection CE, Design Requirements for Graphite Core Supports, was intended to develop code requirements for nuclear graphite core supports. However, this subsection never received consideration by the Section III subcommittee, therefore, Subsection CE was never approved. Currently, the project team has official jurisdiction in this area. Currently the project team's official title is the Section III Project Team on Graphite Core Components. The permanent home for the committee is a matter being discussed in the ASME Executive Board. The project team sent a letter to the ASME Executive Board requesting assignment to the Section III of the ASME B&PV Code. The other option was for the ASME Executive Board to reserve a new section in the ASME B&PV Code for the graphite design codes. The project team members feel the code will address irradiated graphite behavior, and thus a permanent home in Section III will add more creditability to the code. This year's committee's activities are detailed in the report *Status of ASME Section III Task Group on Graphite Core Support Structures, INL/EXT-05-00552*[43],

The current charter for the project team is as follows:

> The committee shall establish codes, standards and guides for materials selection and qualification, design, fabrication, testing, installation, examination, inspection, certification, and

the preparation of reports for manufacture and installation of nonmetallic internal components for graphite-moderated fission reactors, where nonmetallic internal components are defined as components, including control rods and assemblies, contained within a graphite-moderated fission reactor pressure vessel and manufactured from graphite, carbon, carbon/carbon composites, ceramics, or ceramic matrix composites. The codes, standards, and guides shall apply to nonmetallic components as defined above. The codes, standards, and guides shall not apply to graphite fuel matrix materials, fuel compacts, fuel pebbles, bushings, bearings, seals, blanket materials, instrumentation, or components internal to the reactor other than those defined above.

The project team feels graphite design codes are of higher importance than design codes for carbon/carbon composite and ceramics materials. The carbon/carbon composites and ceramic materials are not yet mature to the point where ASTM standards can be developed. Without ASTM standards, the actual material being considered does not have the pedigree to be recognized by the ASME Board. ASTM standards for graphite are undergoing final balloting or round robin testing at this time. A departure from the prior drafted code is that the charter includes the graphite used in prismatic fuel blocks, but not the graphite matrix containing the fuel. The project team feels the graphite used in the prismatic fuel blocks should be included in the design codes responsibility contrary to prior drafted Subsection CE, which excluded the graphite in the fuel blocks.

Subsection CE was structured in a similar manner as a metallic design code. Subsection CE did not have the required databases for material properties and performance qualification for graphite as metallic design codes have. The ASTM nuclear graphite specification undergoing final balloting will establish a minimum material specification, but it is not an absolute specification. Graphite is manufactured by several different methods and uses different mined and man-made precursors. Therefore, nuclear graphite will have minimum specifications, but maximums will vary from grade to grade and billet to billet. Metallic databases deal with unirradiated properties of the material, while graphite properties will change with absorbed dose and temperature of irradiation. There will be no ASTM specification on the irradiated properties because the effects will vary among the different grades of graphite. Graphite is a brittle material with statistical variability in material properties (i.e., graphite strength). Therefore, statistical consideration of graphite strength is required to develop probabilistically based stress criteria.

Metallic design codes are termed deterministic because stresses are assigned to certain categories before being arranged as stress intensities and then compared to different allowables. The allowable stress state is determined from destructive testing with consistent results. Current computation techniques used to model the stress states in metallics have shown excellent predictive capabilities. The current state-of-the-art modeling capabilities for graphite materials do not permit stress predictions to the same accuracy as metallic components, and therefore, using deterministic methods to predict safety margins in graphite component stress states is unreliable.

Failure criteria used in Section III of the ASME B&PV code represent primarily ductile metallic materials, which follow maximum shear stress theory. Nuclear graphite is generally a heterogeneous, isotropic to slightly anisotropic, brittle material whose compressive strength is higher than its tensile strength, and its stress-strain behavior is nonlinear and dependent on hydrostatic stress. Most of the failure theories used for graphite have been generalizations of the von Mises theory. Subsection CE established strength limits based on maximum stress theory, because it will result in a more conservative design. Maximum stress theory states failure occurs when one of the three principal stress components at a point in a body reaches either the uniaxial tensile or uniaxial compressive strength with the weakest axis strength used. The over conservatism could result for example in core support pedestals requiring excessive diameters to meet code. Subsection CE allowed more rigorous multiaxial failure theory to be used, but this section of the code had not been developed.

An underlying axiom in all metallic deterministic ASME design codes is the use of different stress limits for primary and secondary stresses. A primary stress is produced by a mechanical load. As the mechanical load increases, the primary stress reaches a stress level where gross material yielding would occur. To mitigate this situation, a limit is placed on stress state in the material. This limit is determined from extensive destructive testing and has a factor of safety applied to ensure there is an acceptable margin. A secondary stress is produced by mechanical load or differential thermal expansion. A secondary stress indicates a strain controlled situation exists, that local yielding can be accommodated, and that the deformations are self limiting. Thus as load increases, the ratio of stress to load will decrease. Therefore, exceeding primary stress limits is judged to potentially much more damaging than exceeding secondary stress limits.

Nuclear graphite under load and undergoing irradiation produces secondary stresses which potentially could exceed primary stresses. Thus the superposition of primary and secondary stresses into one stress state will provide a more inclusive stress state to base a stress limit on. Therefore, the underlying axiom in metallic deterministic ASME design codes of primary and secondary stresses is not directly applicable to nuclear graphite under load and experiencing irradiation.

Subsection CE was developed for graphite core support structures that do not experience the neutron fluence and thermal loads as seen by graphite core components. As stated in the committee's charter, the new ASME graphite design code will include the graphite in the core. In extending the code to core components, consideration will be given to development of separate design stress limits based on safety importance and absorbed neutron fluence.

As identified in the previous paragraph, core supports may not see the high neutron fluence that other sections of the core would experience. This situation requires two design codes: one for unirradiated material and the other for irradiated material. The project team is working on individual codes for the irradiated and unirradiated.

The project team will recommend that a designer take into account the material properties variability within lots to lots and billet to billet when selecting representative values needed for the design. The collection of data must be made in a meaningful statistic process. The team will emphasize that it will be the owner/operator/designer's responsibility to obtain the irradiated graphite performance parameters necessary for designing the core and core supports as well as address the uncertainty in those measurements and their effects on the safety margins.

Graphite is a brittle material whose failure can be described by a Weibull distribution. Kerntechnische Ausschuss (KTA), the German ASME equivalent, has developed a graphite design code taking into account the Weibull strength distribution in the analysis of the stress state in the graphite. The team has reviewed KTA's documents and found them useful and instructive. The team will employ sections of the KTA rules in sections of the ASME code where appropriate. The team recognizes that ASTM will be forthcoming with a standard to perform measurements obtaining a Weibull strength distribution in a graphite sample population.

The NRC has published a guideline in SECY-03-0047 addressing risk informed licensing:

> Use a risk informed and performance-based approach, wherever practical, consistent with the Commission's 1995 policy statement on the "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities," (60 CFR 42622); SECY-98-144, "White Paper on Risk-Informed and Performance-Based Regulation," dated June 22, 1998; and Yellow Announcement #019, "Commission Issuance of White Paper on Risk-Informed and Performance Based Regulation," dated March 11, 1999.

In order to meet the future NRC requirements, a coupled probabilistic risk analysis with probabilistic stress analysis approach must be taken to develop structural design criteria that incorporates uncertainty and can be used to demonstrate how well component designs achieve their goal and meet plant risk limits. The project team will develop the procedures to ascertain irradiated graphite strength limits based on probabilistic stress analysis to statistically quantify the uncertainty in predicted mean stress. This strength limit is key to establishing an allowable stress ratio that is used plant probabilistic risk assessment analysis. The stress ratio will also incorporate the consequence of failure into the basis by assigning different ratios to core support, reflector, fuel, and control elements. The project team will be assessing the NASA CARES package to assist in determining the strength limits and ratios. The outline of the strength analysis is found in Figure 29.

FY-06 Activities

Funding for INL and ORNL staff to participate and attend subcommittee meetings will continue in FY-06 under the ORNL Activity 301 and INL Activity 301.

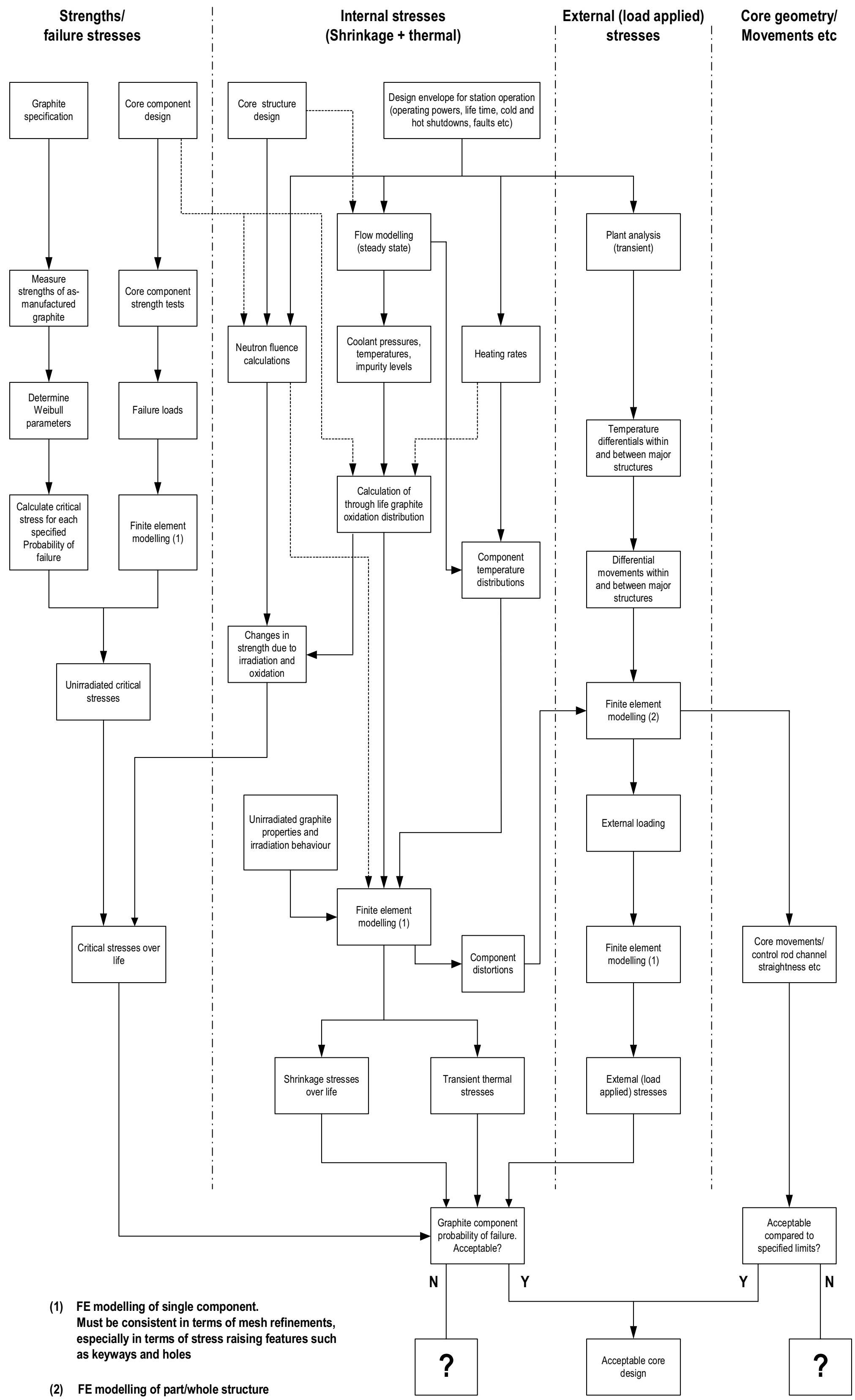


Figure 29. Design flowsheet for unirradiated and irradiated graphite.

3.5.3 Task 3B2 (ORNL): Status of ASTM Subcommittee D02.F Graphite Activities

FY-05 activities

A major activity of D02.F over the past three 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 (affirmatives with comments and a single negative vote) where discussed at the June 2005 meeting. The negative vote was withdrawn after clarification of the point of issue, and the comments were resolved by amendment of the draft standard specification. The revised standard specification (revision 13) was approved for main committee ballot in June 2005 and will be balloted in August/September 2005. 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 has been 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 will commence in August 2005.

Committee D02.F, which includes ORNL and INL staff, has also been working on 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. The results of the second ruggedness test where 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. Twelve labs in six different countries wish to participate in the fracture toughness round robin.

Committee D02.F is currently conducting a round-robin on a draft test method to determine the graphite crystal spacing and crystal size parameters (c, a, lc, la) via x-ray diffraction (XRD). A set of standard samples was distributed to seven labs. Data has already been received from three of the labs (ORNL, GrafTech, and WVU).

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. As minuted at the June meeting new test methods/text will be proposed for inclusion in an expanded version of C 781 and discussed at the December 2005 meeting. Text will be added to C781 for the following tests methods:

Ash determination

- Specific Electrical Resistivity (SER)
- Weibull Parameters
- Coke CTE
- Boron test method

At 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 [44].

<u>FY-06 Activities</u>

The air oxidiaton D02-F standard will continue under ORNL Activity 308 and INL Activity 306. The results of the fracture toughness round robin will be documented under INL Activity 308.

3.5.4 Task 3C (ORNL and INL): Status of Support of the Formation of an ASTM working group on SiC/SiC Composite Testing Development

<u>FY-05 Activities</u>

Unlike other structural materials, initial standardization efforts for SiCf/SiC composites were concurrent with their development because it was recognized that their commercial diffusion and industrial acceptance could be hampered by lack of standard test methods, databases or design codes[45, 46].

Numerous standardized mechanical testing methodologies have been developed for characterizing the mechanical properties of engineering materials. Noteworthy are the standards developed for the ASTM. Typically these standards are based on testing experience including both independent research and round-robin evaluations. Such standards, so developed, are the result of consensus on the part of ASTM participants and, therefore, address the needs of the participants at the time the standards are developed. In the United States, sub-committee C28.07 on Ceramic Matrix Composites of the ASTM has spearheaded the widespread introduction of standard test methods for SiCf/SiC and other ceramic matrix composites[56].

These standards have primarily concentrated on the evaluation of test coupons to determine the intrinsic mechanical properties of these materials and little work has been focused on the development of standards for the evaluation of ceramic matrix composite components. The potential use of SiCf/SiC composites in the VHTR will require the existence of:

- Design codes, which list "rules" and guidelines for designing and testing SiCf/SiC composite components and incorporating them into advanced designs;

- Design codes which regulate the certification procedures for processing materials, fabricating components, and assembling final designs; and

- Databases that provide statistically significant and complete material properties and performance.

Since 1995, one noteworthy national effort has been initiated in design codes for advanced ceramics: ASME B&PV Code. Of particular importance for the Next Generation Nuclear Power (Gen IV) applications (such as control rod cladding and guide tubes) are acceptance of aspects of codes (including standards) by the NRC.

The primary technical objectives of this project are:

1. To coordinate efforts that lead to the introduction of national (ASTM) and international (ISO) test standards for the thermo-mechanical evaluation of SiC_f/SiC composites and components fabricated with these materials;

2. To coordinate round-robin testing programs for establishing precision and bias statements for the new standards;

3. To carry on efforts for developing national design codes that address the use of SiC_f/SiC composites as part of such national efforts as the ASME B&PV Code; and

4. To facilitate efforts for development and expansion of databases for SiC_f/SiC composites.

This project addresses specific needs in the characterization of SiCf/SiC composites for ultimate use in the engineering design and fabrication of control rod cladding and guide tubes in nuclear power plants. This work has been prioritized based on the expected modes of failure of these components.

A portion of the work performed in this area in FY-05 is documented in *Development of Standardized Test Methods, Design Codes and Databases for SiC/SiC Components in Next Generation Nuclear Power Plant Systems*, ORNL-GEN4/LTR-05-004[45]

FY-06 Activities

The FY-06 activities for will continue under ORNL Activity 307 and INL Activity 305.

3.5.5 Task 3D (INL), Test New Fracture Toughness Standard for Graphite

The work performed in this activity is discussed in Section 3.3.6.

3.5.6 Task 308 (ORNL) and Task 306 and 307 (ORNL), Develop Draft ASTM DO2.F Air Oxidation Test Standard For Graphite

A draft air oxidation test standard for graphite will be written in FY-06.

3.5.7 New Activities That Should be Considered for Funding in FY-06 not Currently in the Baseline Budget

The activities that should be considered for funding in FY-06 are given in Table 22.

Table 22. Activities that should be considered for funding in FY-06

Task Title	Task Description	Benefit to NGNP Program	Total Funding Request, $K	Funding Split, $K	
				ORNL	INL
Spiral Notch Torsion Test of Graphites and Structural Composites	Test technique will be evaluated for applicability for specific non-metallic materials needed for NGNP including, size effects, round robin testing in support of a nuclear graphite and composites fracture toughness as an accepted standard. The minimum diameter and length of the cylindrical specimens required to provide accurate data in subsequent irradiation testing will be determined. Results for the nuclear graphite in particular will be compared with the current Gen IV single edge notched beam data. Simplification of technique for use with irradiated samples will also be addressed in this task. Following definition of technique and relevant sample geometry, a minimum of three laboratories will be involved in a round robin test. Data will be analyzed to define precision and bias of the test. SNNT will be assessed for determination of irradiation and environmental affects on NGNP materials.	Provide improved testing efficiency and cost reductions for evaluating performance of non-metallic NGNP materials for high-temperature, irradiated behavior.	545	340	205
ASTM Standard Development	Support is needed to interface with the standards committees to develop proper testing procedures for these composites. Currently, there is no home for carbon-carbon composites within the ASTM structure, so funding is required to get this developed and progressing towards accepted standards that will be required for qualification of data for NRC.	Without accepted standards, it will be increasingly difficult to present data to the VHTR program and get acceptance of these materials by the NRC.	70	35	35
		TOTALS	**615**	**375**	**240**

3.6 Environmental Testing and Thermal Aging Project

The three primary factors that will most affect the properties of the structural materials from which the NGNP components will be fabricated are effects of irradiation, high-temperature exposure, and interactions with the gaseous environment to which they are exposed. An extensive testing and evaluation program will be required to assess the effects of these factors on the properties of the potential materials to qualify them for the service conditions required.

Procedures for the evaluation of aged and "service-exposed" specimens will be developed. Properties evaluation will be performed on a limited number of materials including Inconel 617, Alloy 800H and Alloy X that have been aged at temperatures as high at 950°C for long times in helium. It is expected that aging exposures will be performed to at least 25,000 hours.

Mechanical and microstructural properties of bulk and weld structures will be evaluated and the determined experimental properties will also service as input and checks of computational continuum damage modeling activity for high-temperature life prediction. Results of mechanical testing and microstructural evaluations of candidate alloys aged 1000, 3000, and 10,000 hours will serve as input to computational continuum damage models. The predictions of these models will be compared to results of testing of materials aged to at least 25,000 hours so as to provide for validation of these models. The mechanical and microstructural data will also provide input into code rules for accounting for aging effects.

The overall stability of the proposed helium environment must be evaluated in order to ensure that testing proposed in various sections of the program is performed in environments that have consistent chemical potentials. In addition, the corrosion of metals and nonmetals will 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 NGNP is being developed to produce hydrogen as well as electricity. Conceptual designs call for a gas cooled reactor with an outlet temperature greater than 850°C required to efficiently operate the hydrogen generation plant. While the design concepts are not yet final, it is highly probable that helium will be the working fluid in the reactor. The primary material in the core will be graphite and the prime candidates for metallic internal components are the nickel based alloys Inconel 617 or Alloy 230. An artist's representation of the reactor and power conversion vessel and the associated hydrogen generation plants is shown in Figure 30 below. A heat exchanger in the power conversion vessel (shown in orange in the figure) will take approximately 10% of the thermal energy of the reactor and divert it as process heat to the hydrogen production plant.

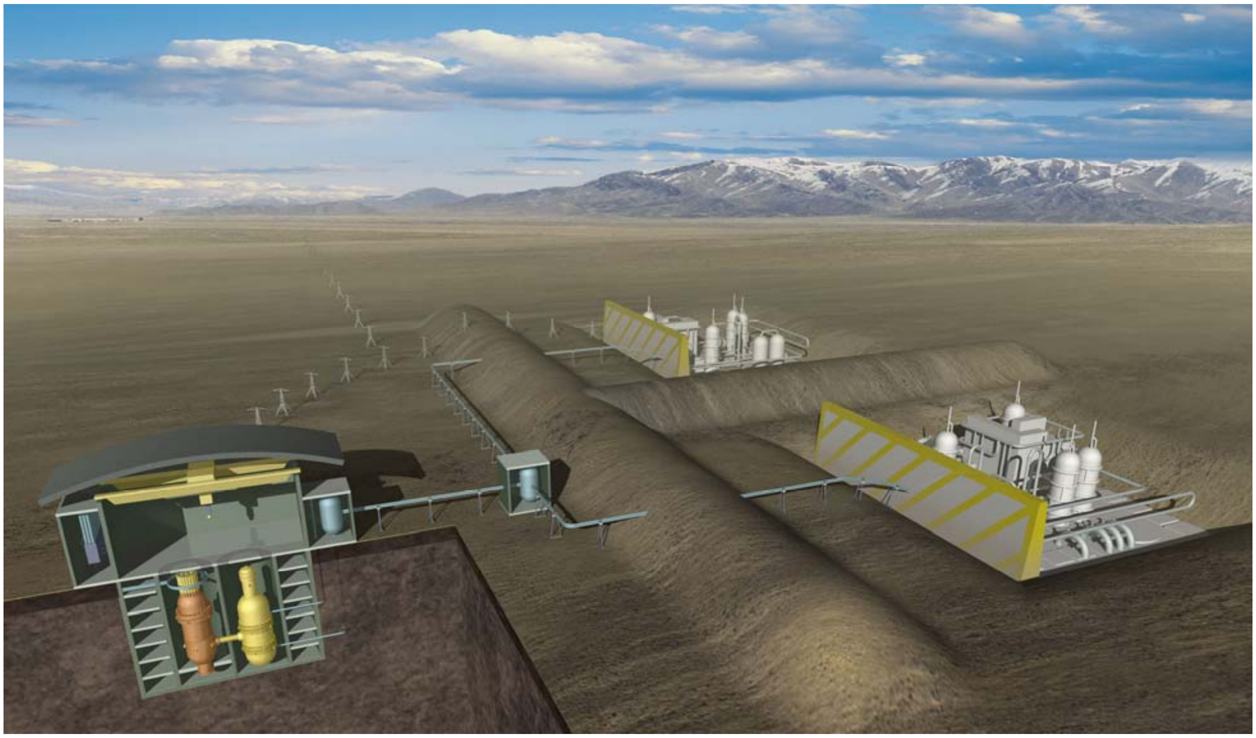


Figure 30. Schematic representation of the NGNP reactor and power conversion vessel and associated thermo-chemical hydrogen generation plants.

Experience with similar 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 1, along with values used previously for a number of reactor programs[47]. 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 1 since there were substantial leaks of steam into primary circuit from the power generation circuit in this plant.

It has been determined that the values for the impurities in determined for the Advanced Gas-Cooled Nuclear Reactor (AGCNR) program are the appropriate nominal levels for the NGNP program. The gas compositions in Table 23 are not in thermodynamic equilibrium, rather they represent a kinetic steady state. The actual gas chemistry at a point within the reactor will be determined by reaction rates between the impurity species and are a strong function of temperature[47, 48, 49, 50]. Impurity content is influenced by reaction of materials with the coolant, for example oxidation of metallic components will deplete the gas stream of oxygen. Methane (CH_4) is produced in the core by a radiolytic reaction between graphite and water vapor and will decompose to carbon and hydrogen at high temperature in the absence of radiolysis.

Reactions which can occur and are thought to be quite rapid include:

$$2C + O_2 \leftrightarrow 2CO$$

$$C + H_2O \leftrightarrow CO + H_2$$

$$C + CO_2 \leftrightarrow 2CO.$$

Other possible reactions, that are thought to proceed more slowly include:

$$CO + H_2O \leftrightarrow H_2 + CO$$

$$C + 2H_2 \leftrightarrow CH_4.$$

In addition to the balance between these competing reactions, gas cooled reactors have also employed gas purification systems of several different types and with varying capacity. The gas chemistry is thus a complex balance that will vary depending on how the system is operated and as a function of temperature within the plant. With respect to interaction with materials, either graphite or metallics, the potential for oxidation or carburization is the critical concern. The oxidation potential of the gas is determined by the partial pressure ratio H_2/H_2O and the carburization potential is related to the ratios CH_4/H_2 and CO/CO_2.

3.6.1 Task 4A (INL): Design and Construct a Recirculating Low Velocity He Loop

<u>FY-05 Activities</u>

A closed circuit low flow velocity test loop has been designed and assembled at the INL, see Figure 31. This loop has the ability to expose coupons and mechanical test specimens in controlled impurity atmosphere at high temperature for long periods. Other test systems have been designed where the gas chemistry is controlled by bleeding off a portion of the test atmosphere and refreshing to the desired chemistry with controlled additions of gas. The INL system is designed so that it can operate in this mode, this new system is also designed to with the potential to continuously getter excess impurities and adds necessary trace impurities based on mass spectroscopy measurements in a closed loop system. Details of the system are discussed in *Controlled Chemistry Helium High Temperature Materials Test Loop*, INL/EXT-05-00653[51].

Figure 31. Photograph of the assembled low velocity controlled chemistry test loop. The computer shown controls the measurement of gas chemistry using the mass spectrometer; a second computer that is not shown controls automation of the gas chemistry through operation of the valves shown in Figure 32.

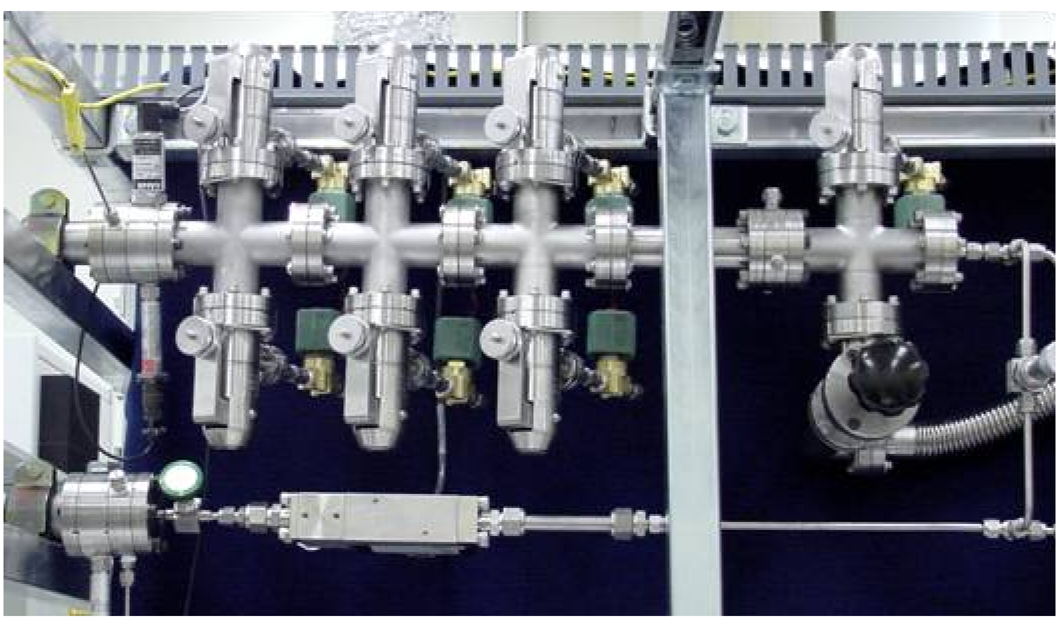


Figure 32. Photograph illustrating the seven sapphire seated needle valves used to introduce very precise amounts of impurity and the attached rotary valves (shown as green in the photo)

A schematic of the components of the test loop is shown in Figure 33. The schematic shows that the system has a vent system that will only be used in the so called “bleed and feed” mode of operation. The system can be evacuated to a pressure of 10^{-6} torr in the specimen chamber using a turbo-molecular pump. All of the tubing for the system is stainless steel and can be heated during evacuation to help remove adsorbed impurities. There is a metal bellows pump in the loop capable of 40 l/m flow; the total system volume is approximately 20 liters.

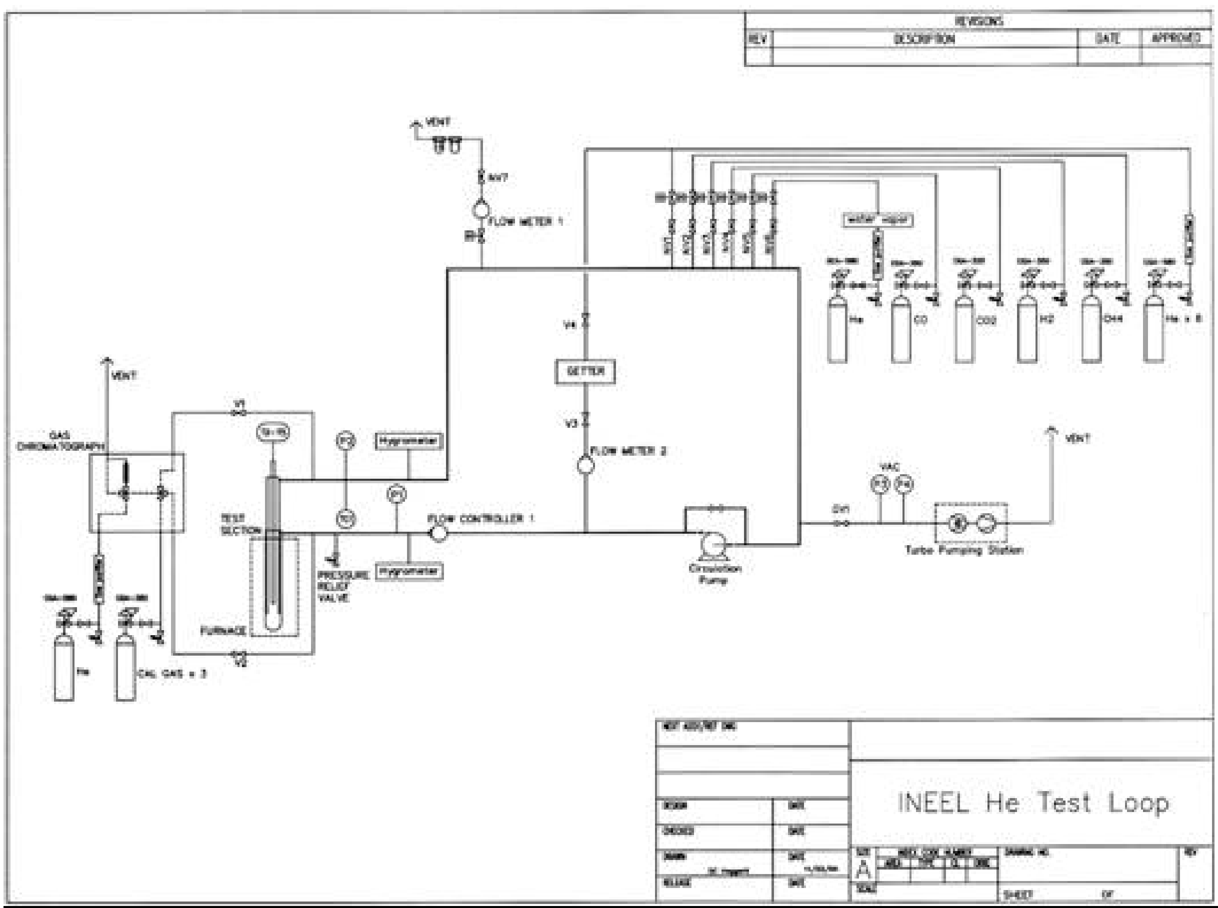


Figure 33. Schematic of the INL low velocity controlled chemistry helium materials test system.

The test section consists of concentric quartz tubes inside the furnace section where is cool gas moves down through the annulus and returns up past specimens that will be suspended on a central rod. The hot zone of the retort is approximately 0.75 m in length and is designed to operate up to 1000°C. The quartz tubes are sealed to the system using an o-ring assembly that allows tubes to be disassembled to insert and retrieve specimens and to replace quartz tubes as necessary. Details of the retort and fittings are shown in Figure 34. Although provision has been made to water cool the aluminum fittings that hold the o-ring seal assembly for the quartz tubes, experiments with flowing helium up to 1000ºC have indicated that it will probably not be necessary.

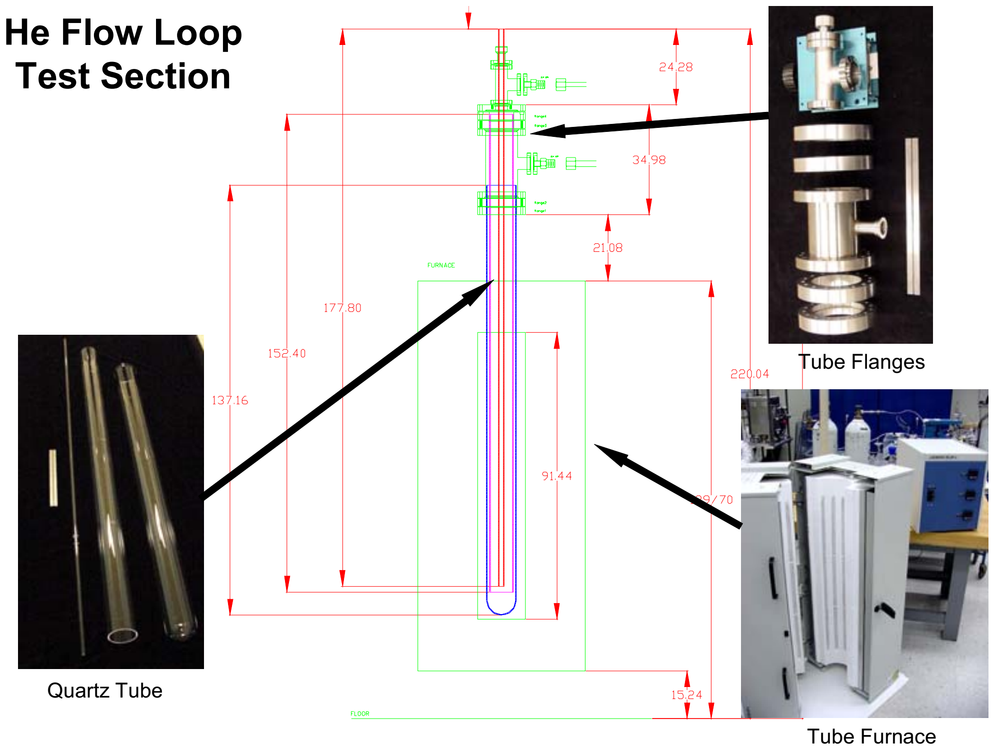


Figure 34. Details of the retort for exposure of test coupons.

Gas composition is measured going into the retort and upon exit from the hot zone. Water vapor content is measured using solid state hygrometers. The content of the other impurities is measured using a gas chromatograph with a pulsed ionization source. All of the gas compositions will be measured to one part per million or less. As shown in Figure 33, all of the impurities are added to the system as individual gases, rather than premixing gases. A sapphire seated needle valve is used to very precisely control the introduction of each impurity gas into the system. To automate impurity additions, solenoid operated rotary valves are used to control the gas that is introduced to the needle valve. A photograph of the needle valves with the attached solenoid operated rotary valves is shown in Figure 32.

FY-06 Activities

One important function of this closed loop system is to determine the steady state gas composition as a function of starting gas chemistry. Experiments will be conducted as a function of time and temperature without gettering excess impurities to monitor the evolution of gas chemistry toward the steady state. It is expected for example that initial concentration of CH_4 will decline over time as the methane is decomposed at high temperature to C and H_2. After determining the dynamics of gas composition with time at temperature it will be possible to develop appropriate getter systems to remove excess impurities. These experiments will form the basis for design of a much larger high velocity flow loop that will be built in the future to test materials in conditions that more closely simulate the NGNP reactor system where helium flow velocities on the order of 50 to 75 m/s are anticipated.

A series of tests will be conducted with standard Alloy 617, CMS Alloy 617, and Alloy 230 with exposures up to 5,000 hours in a range of temperatures from 800 to 1000ºC under controlled chemistry conditions. These tests will include both coupons for microstructure analysis and mechanical test coupons. The corrosion behavior will be determined on exposed coupons using optical and scanning electron microscopy to characterize the nature and extent of environmental interaction. Transmission electron microscopy may be employed to a very limited extent if determination of the extent or chemistry of phases requires use of this higher resolution tool. A small number of tests will be carried out in the INL test loop for comparison to experiments in ORNL loops with identical conditions to verify that identical results are obtained. This work will be performed as a part of Activity 401 (INL and ORNL).

3.6.2 Task 4B (INL) and 4D (ORNL): Acquisition of Long Term Thermal Aging Test Specimens.

<u>FY-05 Activities</u>

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 35. 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.

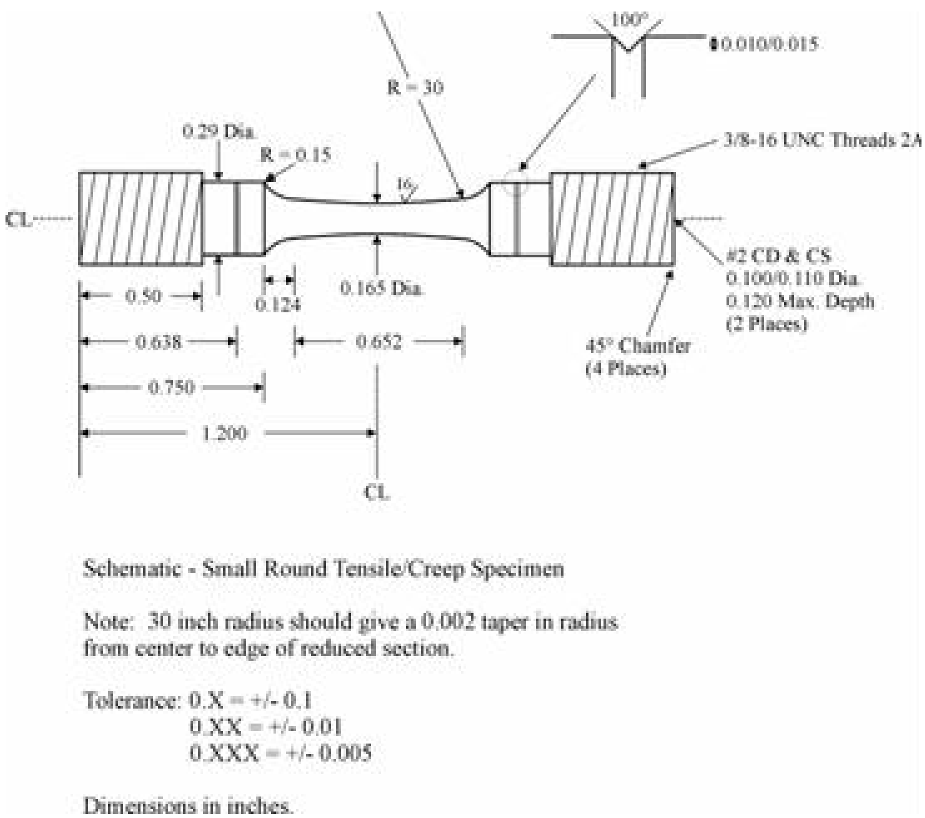


Figure 35. Schematic of the mechanical testing coupons used for long term aging and environmental exposure effects testing.

<u>FY-06 Activity</u>

The aging and environmental specimens will be treated and mechanically tested under INL Activity 401. The plate specimens will be tested under INL Activity 404.

3.6.3 Task 4E (ORNL) and 4C (INL): Test Plan for Long Term Thermal Aging and Environmental Effects Program for Alloy 617 and Other NGNP High Temperature Candidate Materials.

FY-05 Activities

This work was documented in *Aging and Environmental Test Plan*, ORNL/TM-2005/523[52]. 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. 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 Code Case.

FY-06 Activities

The test plan will be updated as new information becomes available. Activities 404-406 (INL) will be performed to continue this work in FY-06.

3.6.4 Task 4A (ORNL): Review of Aging Effects in Alloy 617.

FY-05 Activities

This work was documented in *A Review of Aging Effects in Alloy 617*, ORNL/TM-2005/511[53]. 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.

FY-06 Activities

The review will be updated as new information becomes available.

3.6.5 Task 4B (ORNL), Refurbishment and Restart of Two Recirculating Low Velocity Loops at ORNL.

FY-05 Activity

Because of significant in-leakage problems, which necessitated special rebuilding of valve components by a commercial vendor, as of September 2005, one helium recirculating loop has been rejuvenated. It is anticipated that the rejuvenation of a second loop will be completed and gas/gas studies will be initiated by the end of 2005. These loops are designed to allow for recirculation of helium gas with controlled levels of impurities (H_2, H_2O, CO, CO_2, CH_4, and N_2) and with inlet and outlet chemistries within 10% of each other.

FY-06 Activities

Gas/gas reactions will be evaluated and specimen exposures will begin in FY-06 under Activity 401.

3.6.6 Task 4B (ORNL), Assess Past Helium Test Environments to Determine the Range of Impurities

FY-05 Activities

This work was documented in *Potential Helium Test Environment for next Generation Nuclear Plant Materials*, ORNL/TM-2005/92[47]. An analysis of potential helium environments for the NGNP was performed. 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 previous 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.

FY-06 Activities

Composition of the test environment will be updated as new information becomes available.

3.6.7 Task 4C (ORNL), Review the Existing Data/Information on the Environmental Effects of Impure Helium on Alloy 617

FY-05 Activities

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[54]. This review was performed 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.

FY-06 Activities

Testing will be performed to evaluate the effects of the environment on microstructure and germane mechanical properties in Activity 401 (ORNL) and Activity 405 (INL).

3.6.8 New Unfunded Activities Proposed in FY-06

New unfunded activities proposed in FY-06 are given in Table 23.

Table 23. Unfunded Activities Proposed in FY-06

Task Title	Task Description	Benefit to NGNP Program	Total Funding Request, $K	Funding Split, $K	
				ORNL	INL
Environmental creep-fatigue testing	Initiate scoping creep-fatigue testing including design and construction of an environmental test chamber and gas handling system to evaluate the performance of Inconel 617 under creep-fatigue testing conditions in the presence of helium with controlled additions of impurities and in a "reference" environment (air, vacuum, or high purity helium; to be decided) at temperatures in the range of 850 to 950°C. Small cross-sectional specimens will be employed to magnify the environmental interaction effects. As prepared specimens and specimens that have been pre-exposed with be evaluated. Two to three "helium with controlled additions of impurities" environments will be employed.	Develop data needed for assessment of environmental affects required by the ASME code for NGNP high temperature metallic components	550	550	
High Velocity Helium Environmental Effects Test Loop	Conceptual design of a high velocity recirculating helium test loop will be completed. The loop will be designed to test coupons and mechanical test specimens at temperatures up to 1000C in controlled impurity helium at velocity from 50 to 75m/s. There will be capability for particle injection at the specimen location to simulate expected dust loading in the reactor coolant. Critical design issues to be resolved are design and sizing of pumps, gas impurity control systems and particle injection methods.	It is anticipated that there will be very high velocity helium flows at certain points in the NGNP reactor. Material degradation mechanisms are expected to be different in high flow conditions compared to quasi-static conditions that are currently being examined. This test loop will more closely simulate the in-service environment.			250
		TOTALS	**800**	**550**	**250**

3.7 Develop and Qualify Materials for Irradiation

In order to evaluate the irradiation effects of candidate alloys under relatively low flux test reactor conditions, evaluations will be initiated regarding design of an irradiation facility. This facility will replace the irradiation facility that was shutdown last year at the Ford Test Reactor at the University of Michigan. An irradiation facility to accommodate a relatively large complement of mechanical test specimens will be designed and fabricated for placement in a material test reactor. The facility will, of course, 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, anticipated to be a joint DOE facility with the NRC, will be established in one of the material test reactors surveyed. Preliminary design concepts envision two separate and independent operating capsules in the facility, one for the NRC-funded Heavy-Section Steel Irradiation Program and the other for the Generation IV Reactor Materials Cross-cutting. The capsules can be readily designed and fabricated to operate from 250 to 650 °C, with a preliminary fast neutron flux of about 1 to 2 x 10^{12} n/cm2·s (>1 MeV).

<u>FY-05 Activities</u>

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 and (2) maintaining contact with and updating technical and financial input from potential candidate host reactors. Since it was decided that DOE and NRC would require that a Memorandum of Understanding be developed between the two sponsoring organizations, 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 is still pending.

Since final selection of the host reactor must await final agreement upon 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.

<u>FY-06 Activities</u>

Once approval of the MOU is obtained from the NRC and DOE, the design effort will proceed. An RFP will be issued and the responses evaluated. Site selection will be performed and a contract put in place for facility construction. Following site selection, design and fabrication will be performed for the irradiation hardware to be used in the facility, incorporating any useful hardware remaining from the Ford Test Facility.

3.8 Composites Development Project

Fiber reinforced ceramic composites have been identified as possible material candidates for high temperature nuclear reactor components. 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-m 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."[55] 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, see Figure 36. As seen, 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.

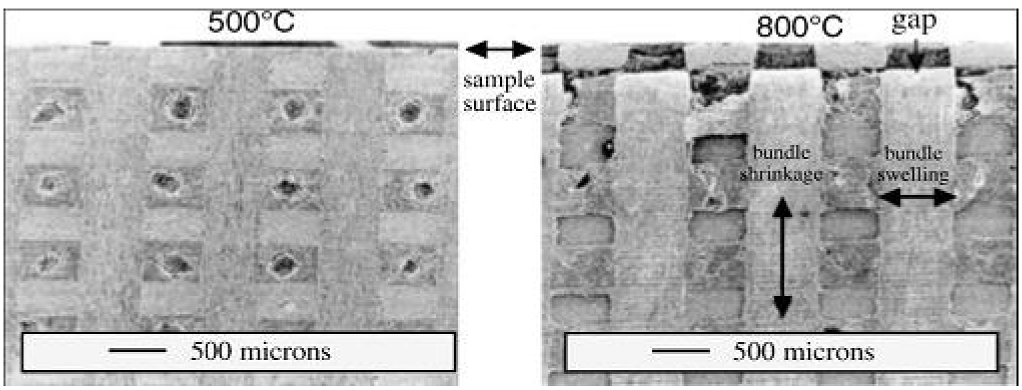


Figure 36. Irradiation damage in Cf/C composites due to dimensional changes in the carbon-based microstructure. (From L. Snead et al, J. Nuc. Mater., 321 (2003) 165–169)

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.[56] 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 VHTR, see Figure 37. Composites fabricated using the latest SiC fibers (Hi-Nicalon Type-S) show considerable stability up to 8 dpa as shown by the Hi-Nicalon Type-S curve (◆). While slightly less stable than monolithic SiC, the composites show a threshold behavior where the mechanical properties do not change significantly after about 1 dpa. While the current data only extends to 7-8 dpa rather than the required 30 dpa, the irradiation stability trends shown for SiC composites are encouraging.

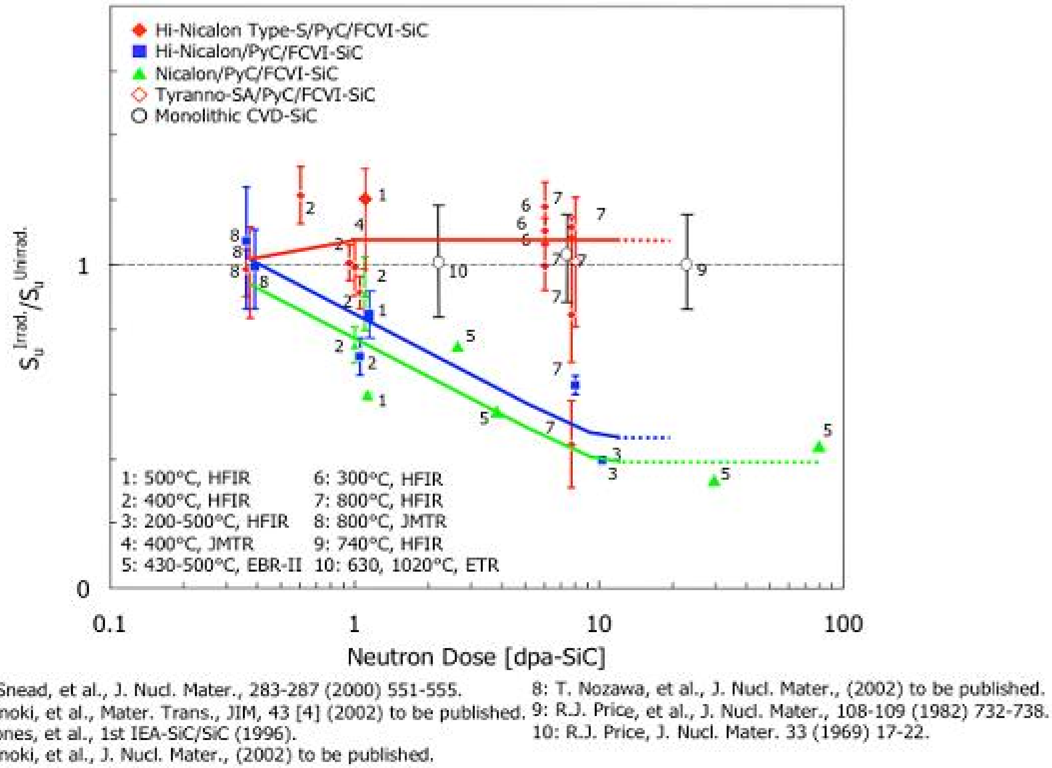


Figure 37. Irradiation stability of different SiCf/SiC composite types. The irradiated-to-non-irradiated ultimate strength ratio ($S_U^{Irrad}/S_U^{Unirrad}$.) plateaus after 1 dpa illustrating no change in mechanical properties for composites using Hi-Nicalon Type-S fibers. This stability is seen up to 8 dpa.

Nearly as thermally stable as C_f/C composites and potentially stronger, these composites are considered a viable alternative material system for control rod applications. The primary issue for SiC_f/SiC composites is the small amount of manufacturing experience and relatively few suppliers available to meet the demands of building this complex component. Therefore, the challenges for SiC_f/SiC composites lie in their fabricability, material supply, and the cost of manufacture.

3.8.1 Task 6A (ORNL): Summary of SiC Tube Architecture and Fabrication

FY-05 Activities

This work is documented in *Summary of SiC Tube Architecture and Fabrication*, ORNL-GEN4/LTR-05-007[61]. As a part of 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. Evaluation of the fabricated materials / components is planned for FY-06. Partially densified reference tubes, to which the interphase deposition and the first stage matrix infiltration was applied, are shown in Figure 38.

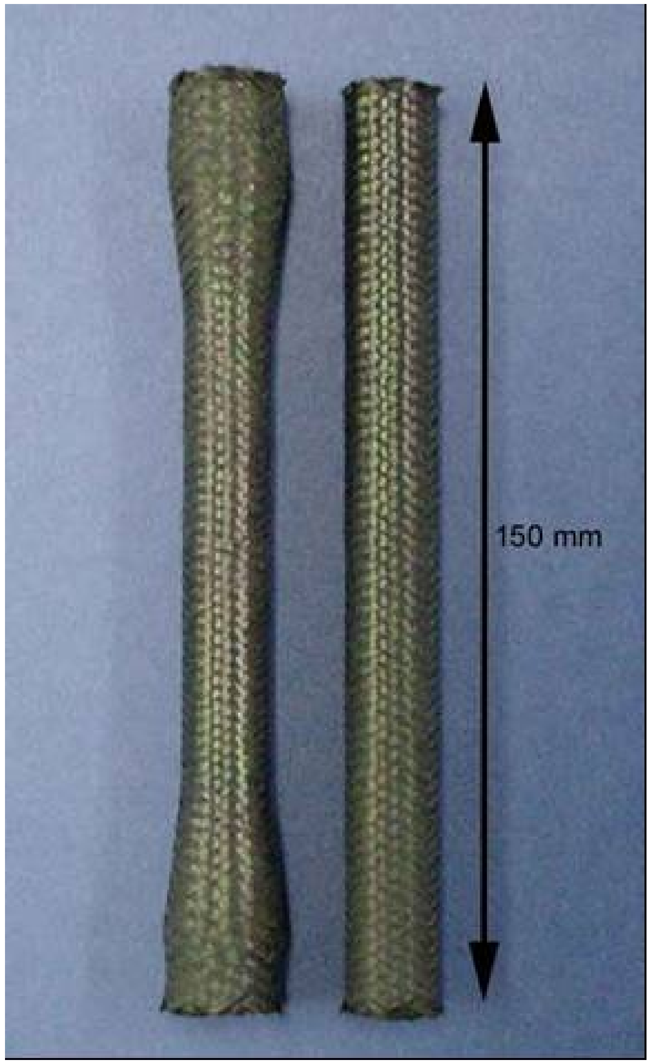


Figure 38. Partially densified reference NGNP grade tubes and non-nuclear grade tubes

FY-06 Activities

ORNL Activity 601 continues this work in FY-06.

3.8.2 Task 6B (ORNL): Status of Irradiation of Multilayer SiC/SiC and Graphite Composites

FY-05 Activities

This work is documented in *Status of Irradiation of Multilayer SiC/SiC and FMI-222 Graphite Composites*, ORNL/TM-2005/508,[57]. The objective of 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 await funding for post-irradiation examination (PIE). The remaining capsules will continue to be irradiated and tested over the next two fiscal years.

FY-06 Activities

The continuation of this work will be supported by Activities 603 and 604 in FY-06.

3.8.3 Task 5A (INL): Environmental Effects on SiCf/SiC Composites

FY-05 Activities

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. This model is currently being expanded to include flat, thin specimens (i.e. to simulate flat dog-bone shaped tensile specimens). It is anticipated that the model may be further expanded to include the 3-dimensional tubular geometry if applicable/desirable at a future time.

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 creep growth results (experimental and modeling): 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 we 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.

3.8.4 Tasks 5B and, 5C (INL): Progress on NGNP Composites Development Activities

FY-05 Activities

Task 5B (INL), initiate the determination of the geometry effect in composite tubes will be documented in *Status of Geometry Effects on Structural Nuclear Composite Properties*, INL/EXT-05-00756[58]. 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 and the discussion given in this section

Task 5C (INL), establish a program for creep testing of composite tubes, is documented in *Creep of Structural Nuclear Composites*, INL/EXT-05-00747[59]. 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. Actual testing of both tubular and flat, "dog-bone"-shaped tensile composite specimens will begin next year. Since there is no precedence for using ceramic composites within a nuclear reactor, ASTM standard test procedures are currently being established from these high temperature mechanical tests. This is discussed further in the report noted and the discussion given in this section

This a more general discussion of these activities is documented in *Structural Ceramic Composites for Nuclear Applications*, INL/EXT-05-00652[60].

Fiber reinforced ceramic composites are being considered for possible use as control rod cladding and guide tubes within a VHTR design. A research program has been established to investigate these materials within the parameters of a VHTR core during service. Two candidate systems have been identified, carbon fiber reinforced carbon (C_f/C) and silicon carbide fiber reinforced silicon carbide (SiC_f/SiC) composites. Each material system exhibits high thermal stability, good mechanical strength, and relatively high fracture toughness.

Extensive thermo-mechanical testing will be required to determine whether SiC_f/SiC or C_f/C ceramic composites are truly viable in a control rod cladding application. Since the temperatures within a VHTR core are anticipated to be extreme (~ 1600º C) thermal stability, or creep resistance, is recognized as a primary degradation mechanism over the lifetime of the reactor (up to 60 years). In addition, core components are expected to receive the largest flux levels with a corresponding total dose of 30 dpa for these lifetime components. This requires that an extensive thermal and irradiation creep program be conducted upon these composite systems to determine the structural stability over the reactor lifetime.

High temperature creep studies will be required to determine a baseline creep response of the tubular geometry components. Mechanical tensile tests conducted at expected normal reactor temperatures (~1100º C) and anticipated off-normal temperatures (1600º C) at nominal stress levels (~10 MPa) will provide the baseline creep response for these material systems. Once the non-irradiated creep response has been established the enhanced creep effects resulting from neutron irradiation will 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.

Traditionally, it is standard practice to use small, representative test samples in place of full-size components. However, a real problem exists for scale-up of composite materials. Unlike monolithic materials these are composites engineered from two distinct materials using complicated infiltration techniques to provide full density and maximum mechanical properties. The material properties may be significantly affected when the component geometry or size is changed. This is a major consideration since small sample sizes and more suitable geometries are required for test samples especially for irradiated sample studies where the material must be placed within the limited space of a reactor. It must be demonstrated that the smaller test samples adequately represent the true response of larger composite tubes used for control rod applications.

To accomplish this, standardized mechanical tests will be developed from these studies to provide the necessary data required for codification of these materials for use in a nuclear environment. This data, even though it is recognized as preliminary only, will most likely be used in support of a code case for use of composite materials as control rod tubes.

3.8.4.1 Test Sample Design. Both non-irradiated and irradiated high temperature mechanical testing must be performed to ascertain the response for these two ceramic composite systems. While standardized sample test sizes and tubular geometry can be used for non-irradiated baseline testing, irradiated studies will require miniaturized samples that can be easily accommodated within the restricted space of a reactor. To fit into any nuclear reactor, test samples much smaller than the actual control rod diameters (~ 100-mm dia.) will be required. In addition, to further reduce the test sample volume and provide a larger number of irradiated samples, flat, "dog-bone"-shaped tensile specimens are considered to be an optimal geometry for test specimens. However, before these smaller, flat tensile specimens can be utilized it needs to be established that they are truly representative of large tubes, which would be used for the control rods.

Two different test programs are required 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, see Figure 1. These studies will be performed on non-irradiated test specimens with similar fiber preform architectures and infiltration techniques.

3.8.4.1.1 Geometry Effects—Small tubular specimens approximately 125-mm long x 10-mm diameter have been fabricated along with large 254-mm x 76-mm flat plates for future testing[61]. Flat tensile specimens with similar outer dimensions as the tubular specimens will be machined from the flat plate stock. Both flat plate and tubular specimens will be compared and analyzed in "head-to-head" mechanical tests.

3.8.4.1.2 Size Effects—Once the geometry effects have been accounted for, the effects resulting from sample size on the mechanical response will be investigated. The size effects study will be conducted with tubular samples only. Flat tensile specimens will not be used. A series of variable sized tubular samples will be fabricated for tensile testing. Sizes will range from 10-mm dia. – 50-mm dia. and may include a full-size (i.e. 100-mm) series of test specimens if deemed appropriate based upon the results from the smaller diameter testing.

3.8.4.2 Mechanical Testing. A round robin testing program has been initiated for all labs (ORNL, INL, PNNL, and University of Bordeaux-France) with the appropriate number of tubular and flat plate specimens. Once the sample matrix has been established the participating laboratories will mechanically test the samples using similar testing methods. The results will be fed back to the appropriate ASTM subcommittee (or working group) and analyzed. Experts from all labs must work within ASTM guidelines and methods to produce a defensible test matrix and testing procedures for ceramic composite tubes.

Initial tensile strength tests will be required to determine failure response, high temperature "yield strength" or tensile matrix-cracking stress, and geometry effects for both tubular and flat plate specimens. The tests will be conducted over a range of temperatures (RT-1600º C) to envelope the anticipated operating conditions within a VHTR. The high temperature failure response and matrix-cracking stresses will be used to determine the optimal stress loads for long-term creep studies.

3.8.4.2.1 Room-Temperature Studies—Room temperature tensile tests of both tubular and flat plate specimens are primarily designed to investigate the geometry effects study. The quantitative geometry effects between the tubular and flat plate specimens will be determined using a series of head-to-head comparison tests between the flat "dog-bone" and tubular tensile specimens. These tests will be conducted both at the USA national laboratories and with our French collaborators as part of the international test standards development for structural ceramics in nuclear applications.

In addition, the tensile strength results will also provide a comparative study to previous work in these composite systems for the fusion materials program. The fusion materials program used specimens fabricated from different fiber preform architectures and different geometries (i.e. flat, loom-woven plate material). A comparison of the new tensile strength results to the previous results used in the fusion work will illustrate the fabricability of the new tubular geometry components. Any dramatic changes from the expected strength levels would affect the viability of these composites.

To date, new ceramic load grips designed for both tubular and flat specimens have been designed (see later section) as well as an ASTM test matrix for both geometries. Tensile tests and comparison studies on both composite types will begin next year.

3.8.4.2.2 High Temperature Studies—Similar to the room temperature studies, both the Hi Nicalon Type-S tubular and flat tensile specimens will be tested. The tests will be conducted from 900 ºC to 1600º C over a range of times to provide a non-irradiated baseline of tensile strength, matrix-cracking stress, and creep data for these ceramic composite systems.

Both sample geometries will be tested within high temperature load frames using vertical clam-shell heaters and a static load. However, due to the anticipated service in the VHTR, the test frames will necessarily be outfitted with an environmental chamber allowing the samples to be tested at temperature in a He atmosphere, see Figures 39 and 40.

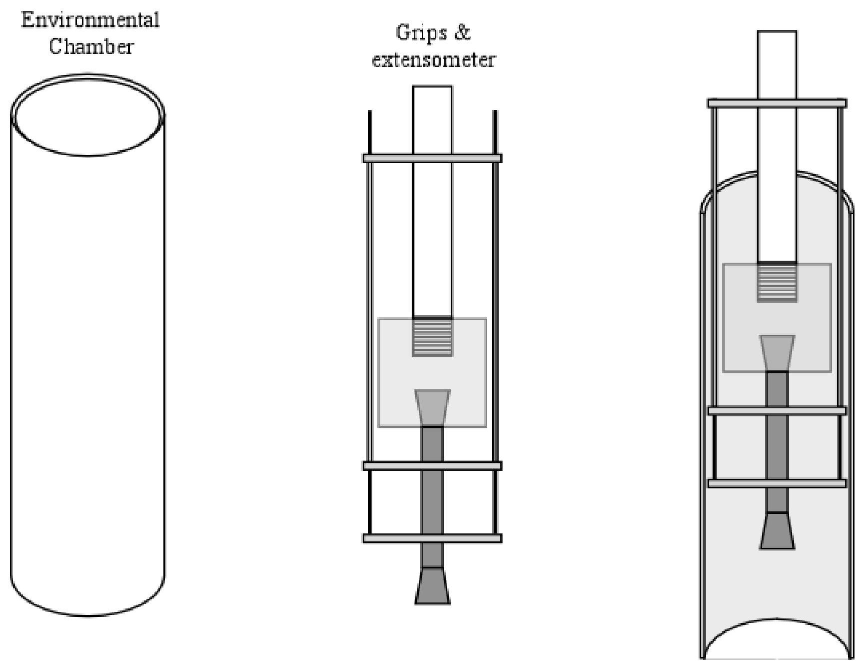


Figure 39. Schematic illustrations of (a) environmental chamber surrounding sample and grip assembly (b) high temperature grips and extensometers with sample, and (c) grip assembly inside cut-away environmental chamber (retort).

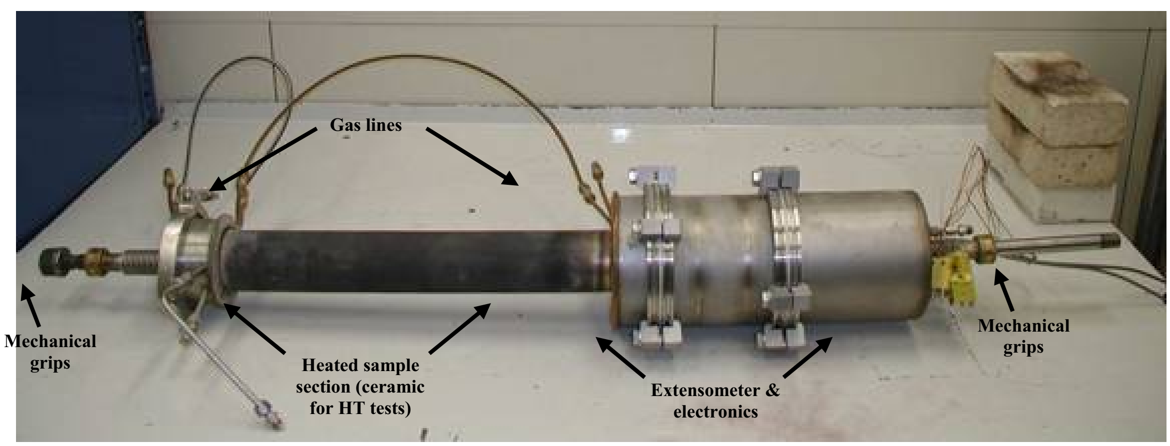


Figure 40. Typical environmental chamber housing required electronics, mechanical grips, and extensometers inside an Inconel chamber capable of withstanding test temperatures of 1000° C.

Currently, load frames capable of very high temperature operation (<1700º C), using compatible environmental chambers, and appropriate controllers have been modified in support of the high temperature tests. The new furnaces will be capable of sustained temperatures in excess of 1600º C to envelope the anticipated operating temperatures of the control rods while the ceramic retort environmental chambers will provide the necessary environmental conditions. The modifications include the following:

- Two resistive heating element (Super Kanthal, MoSi2) furnaces with nominal 100 mm diameter by 200 mm long heated zones, capable of sustained temperatures above 1600°C have been obtained from Advanced Testing Systems (ATS), along with independent power supplies.

- Two environmental retorts have been purchased and received from ATS. These retorts will allow atmospheric control surrounding specimens within the furnaces at temperatures up to 1600°C. The inside diameter of the hot section (approximately 150-mm long) of the retorts is nominally 92 mm.

- The central tubes of the retorts (ceramic environmental chambers) are 100-mm diameter x 775-mm long high-purity alumina tubes with precision ground end sections. These tubes have been ordered from McDanel Advanced Ceramics in Pennsylvania. They are due to arrive sometime about November 1, 2005.

- The two ATS creep frames have had their test sections extended by 300 mm to accommodate the new retorts and furnaces. The vertical members of the frame were sectioned and 300 mm lengths of steel channel were inserted and welded into place. These modifications are completed.

- Control of both the creep loading frames and furnaces, as well as continuous collection of test data is accomplished through a new control system (WinCCS II) purchased from ATS. The control system was delivered along with the furnaces and retorts.

- Frame interface boards and related wiring modifications to the two frames are currently underway. A request has been submitted to facility electricians and pipe fitters to relocate or provide new (as appropriate) electrical power for the test frames and control system, and the furnaces, and to provide plant water cooling supply and drain lines for the retorts.

High purity ceramic alumina environmental chambers will be required for the very high test temperatures (up to 1600º C) and long duration of the creep tests. The relatively small sample sizes for both the tubular and the flat plate specimens will be accommodated within the limited volume of the environmental chambers, see Figure 39. A high purity helium gas environment will be used within the chambers to simulate the VHTR reactor environment. High temperature tests and comparison studies on both composite types will begin next year.

3.8.4.2.3 High Temperature Grip and Extensometer Design—Appropriate high temperature grips, extensometers, and insulation requirements inside the environmental chamber were also addressed. At the expected high test temperatures actively loaded grips will not be possible for long-term creep studies (i.e. mechanically tightened grips will creep and relax inside the environmental chamber). Therefore, a shoulder-mounted gripping system was designed to allow passive gripping for high temperature testing. This required the tensile specimens to have a tapered end, or shoulder, fabricated on each end of the specimen.

High temperature grips capable of being used for both tubular and flat tensile specimens are being designed and fabricated. These are passive grips that use the flared ends of the test samples to load the specimens in tension, as opposed to active grips, which are spring loaded and may lose their gripping force if exposed to high temperatures over long periods of time, see Figure 41.

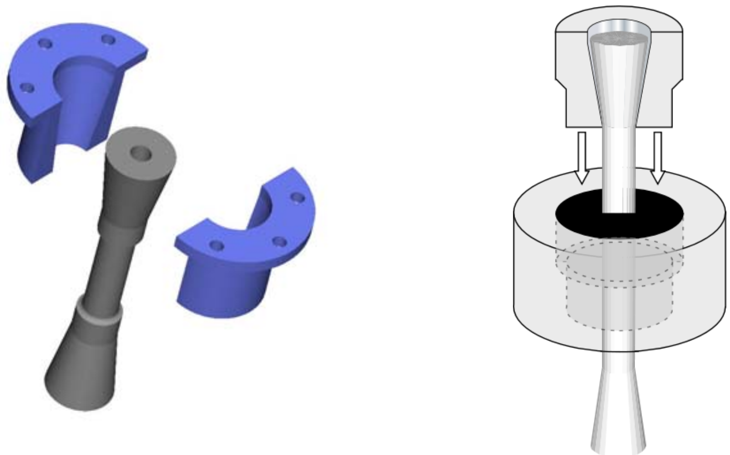
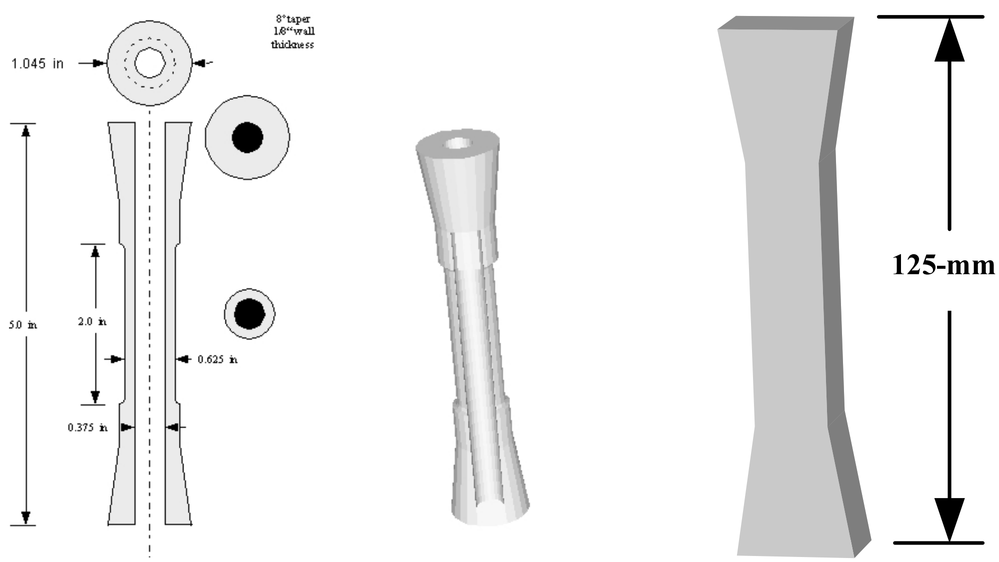


Figure 41. High temperature grip design for passive loading of tubular test specimen.

The grips will be made from either graphite or silicon carbide for maximum thermal stability and strength. Over-sized components will reduce creep induced failures for long-duration tests.

3.8.4.3 Irradiation Creep Studies. As stated previously, the primary degradation mechanism identified for composite control rod components is irradiation creep. Preliminary discussions focusing on irradiation sample dimensions, loading methods, and the design of an irradiation canister for insertion into a reactor have been conducted.

Irradiation samples will necessarily need to be small and compact to minimize volume within the reactor core. Using previous experience and sample designs for creep studies of metals and monolithic materials a general sample size approximately 50-mm long and 12-mm wide (at the ends) has been determined (see Figure 42). No detailed dimensions for the samples have been finalized to date. A final design for load grips or the irradiation canister has also not been determined. Detailed discussions of the sample dimensions and canister design will be initiated next year.



a. b.

Figure 42. Schematic of (a) tubular test samples and (b) flat, “dog-bone” tensile test specimen.

FY-06 Activities

ORNL Activity 601 will fabricate the specimens from the SiC/SiC and C/C stock, which will be received in FY-06. These specimens will be mechanically tested under ORNL Activity 601 and INL Activity 504 in FY-06.

3.8.5 Task 6C (ORNL): Testing Plans for Failure Mode Assessment of Composite Tubes Under Stress

FY-05 Activity

This work is documented in *Summary of Testing Plans for Failure Mode Assessment of Composite Tubes Under Stress*, ORNL-GEN4/LTR-05-002[62].

U.S. DOE's NGNP program considers potential application of silicon carbide (SiC) fiber-reinforced SiC-matrix composite materials (SiC/SiC composites) to the control rod sleeves and guide tubes in the helium-cooled thermal-spectrum nuclear reactors which operate at very high temperatures. 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 [63]. The result shown in Figure 43 indicates that the maximum von Mises stress will be ~1.4 MPa higher than the external tensile stress.

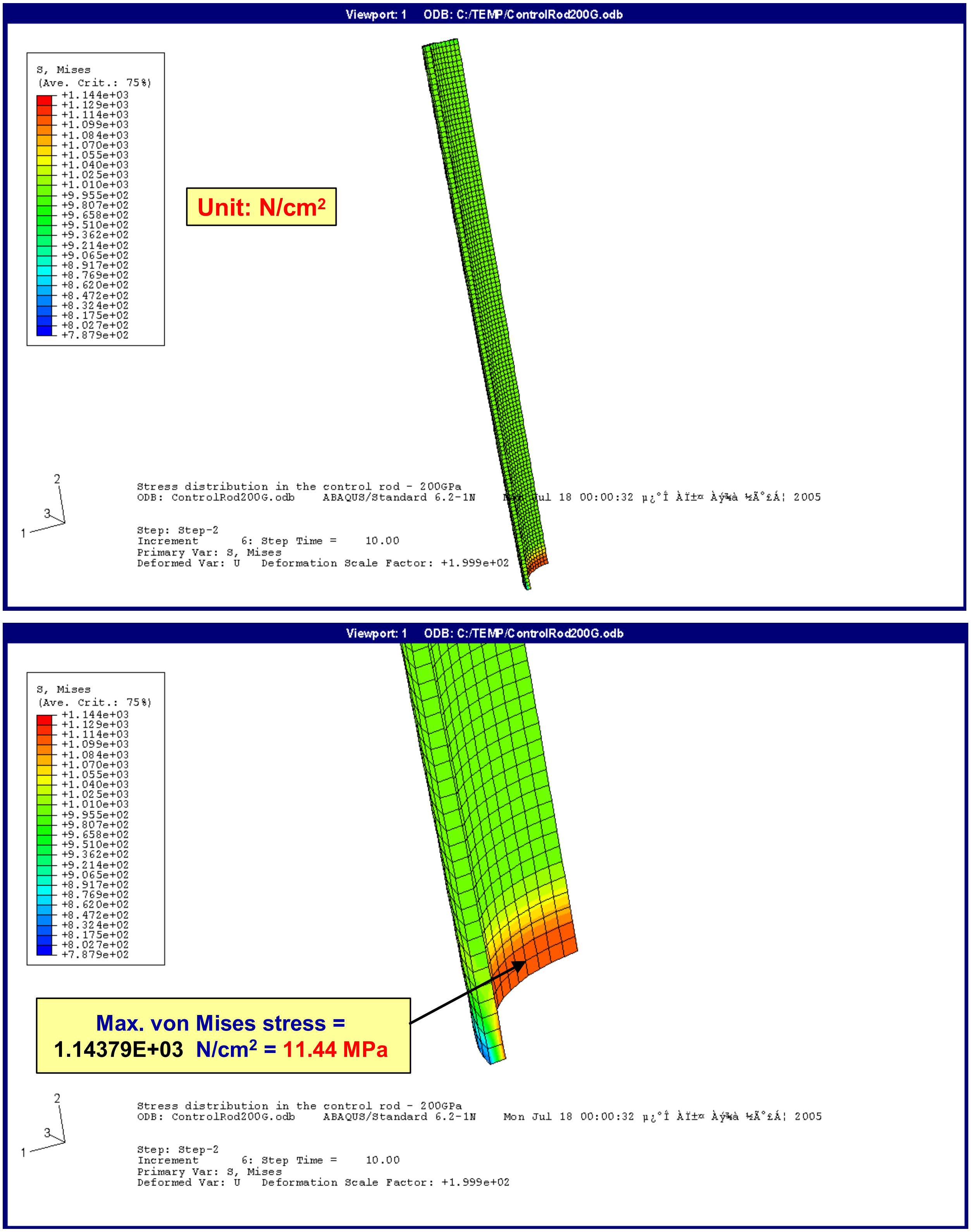


Figure 43. Parameters used in the finite element analysis.

The maximum radial (trans-thickness) temperature gradient in control rod sleeve wall is expected to be small, according to the design data of HTTR [63] ($\Delta T << 10K$ 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 [64] compared to ~25 W/m-K for alloy 800H at ~800°C [65]), 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. These stresses will have to be estimated in a separate work.

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 [66], 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 [67, 68, 69], which 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 is expected to provide the first dependable irradiation creep data. It is worth noting that internal stresses such as thermal stress and differential swelling-induced stress may be relaxed to some extent due to irradiation creep. Therefore, the irradiation creep study on SiC composites is of particular importance also in this aspect.

Other potential mechanical failure modes should be evaluated in a phase of engineering validation when the engineering design activities get appropriately advanced. Other failure modes include (1) axial compression, (2) hoop tension / compression, (3) tube flexure, (4) axial / radial shear and tube torsion, and (5) trans-thickness tension. Corrosion of SiC should also be evaluated when the range of operation temperature and partial pressures of corrosive impurities are assessed.

FY-06 Activities

INL Activities 501 and 504 and ORNL Activity 601 will support the continuation of this work in FY-06.

3.8.6 Task 6D (ORNL) and 5D (INL): Survey of Potential Vendors for C/C Composites

FY-05 Activity

This work was documented in *NGNP Carbon Composites Vendor Survey*[55]. The report contains Export Controlled Information and therefore will not be discussed here.

FY-06 Activity

This work was completed in FY-05 and will not be continued in FY-06.

3.8.7 Task 5E (INL): Purchase Candidate C/C Composite Materials for NGNP Control Rod Applications

FY-05 Activities

The purchase of candidate C_f/C composite materials for NGNP control rod applications was initiated.

FY-06 Activities.

Procurement of this composite material will be completed in FY-06. Adequate funds were carried over from FY-05 to FY-06 to complete the procurement of this material.

3.8.8 New FY-06 Activities

Activities 507 and 508 (INL), develop preliminary creep irradiation design and plan, and Activity 607 (ORNL), perform characterization of C/C control rod cladding materials and prepare for C/C screening irradiations, will be performed in FY-06.

3.8.9 Activities that should be considered for Funding in FY-06 not in the Base Program

The activities that should be considered for funding in FY-06 are listed in Table 24.

Table 24. Activities that should be considered for funding in FY-06

Task Title	Task Description	Benefit to NGNP Program	Total Funding Request, $K	Funding Split, $K	
				ORNL	INL
Irradiation of SiC-SiC Composites	The existing SiC-SiC irradiation program aimed at comparing the performance of carbon-carbon and SiC/SiC composites was limited to 800°C for budgetary reasons. The irradiation matrix currently calls for 10, 20 and 30 dpa irradiations at 800°C. It would be very valuable to examine these materials under irradiation at other temperatures. Fortunately, identical materials to those already being irradiated at 800°C, have begun irradiation under a separate program which has since begin cancelled. This proposal is a simple extension of the current irradiation program to include continued irradiation and testing of materials at 300 and 500°C. The addition of these lower temperatures is also considered beneficial to the ongoing program aimed at defining the applicability of SiC composites to control rod application.	Provide improved understanding of the irradiation performance of SiC-SiC for control rod applications and simultaneously provide a viable contribution to a proposed funds-in CRADA with South Africa that will significantly expand the NGNP graphite irradiation program at no further cost to the United States	290	290	
Post-irradiation testing facility	The composite samples will need to be examined, characterized, and mechanically tested after they have been irradiated in reactor (possibly ATR). While these materials could potentially be characterized in a hot cell at the INL's Materials & Fuels Complex, post-irradiation mechanical testing will be very difficult and expensive to achieve. Since these ceramic specimens have relatively low radiation levels it would be much easier to conduct the examinations and mechanical testing within appropriately equipped glove boxes.	This facility would significantly increase the INL's capabilities within nuclear material testing and characterization for a wide range of NGNP applicable material types (i.e. ceramics, composites, graphite, etc.). This would constitute direct cost savings in handling, transportation, and additional facilities costs to have the work performed elsewhere. Possibly an even greater incentive is that this facility, in combination with irradiation services at the ATR, would provide unique capabilities that could leverage new materials research opportunities in composites and ceramics. These new research opportunities would provide supplemental or even completely offset the costs of performing material tests for some NGNP related activities.	500		500
		TOTALS	**790**	**290**	**500**

3.9 Data Management and Handbook (Jointly funded in FY-05 with the Gen IV Materials Cross-cutting)

The organizational structure to be used in the preparation, control, etc. of NGNP data needs will be finalized. Existing materials handbooks will be examined to determine what information might be extracted and incorporated into the Gen IV Materials Handbook. The primary documents to be reviewed will be the DOE-funded Nuclear Systems Materials Handbook and the AFCI Materials Handbook, followed by relevant portions of other ASME, Pressure Vessel Research Committee, American Society for Metals, etc. documents. A Gen IV Materials Handbook plan will be prepared to identify needed management structures, advisory groups, working bodies, etc.

This will establish the details of the Handbook scope and format including what materials to include (at least initially), what properties to incorporate, and how these are to be presented. It may be that hands-on physical preparation and maintenance of the Handbook would best be done by an outside organization familiar with preparation of similar documents. This task will assess this possibility and, if appropriate, identify and down-select among the qualified outside sources.

A Gen IV Materials Handbook "Implementation Plan" will be prepared. It will provide details of purpose, preparation, publication, distribution, and control of the Handbook. It will also prescribe records required, QA, and review and approval responsibility and authority.

Once fully implemented, the Gen IV Handbook will become the repository for the NGNP materials data and serve as a single source for researchers, designers, vendors, codes and standards bodies, and regulatory agencies. It is also planned to evaluate the potential for including similar data from GIF international partners. Near term activities in this area will include assembling and inputting existing data on materials of interest to NGNP.

The current status of the Gen IV Materials Handbook is given in *Initial Development of the Gen IV Materials Handbook*, ORNL-GEN4/LTR-05-012[70].

3.10 Power Conversion Turbine and Generator Project (Not funded in FY-05)

3.10.1 Turbine and Generator Baseline Materials Test

Preparation of a materials test program in support of power conversion system (PCS) component materials requires knowledge and understanding of the materials requirements for those applications. For the turbine inlet shroud collar and the turbine shroud insulation package container/boundary, the property of greatest importance is very high-temperature creep strength. Further, it is extremely important that the creep behavior (strength and ductility) not be degraded by gas-metal interactions (reaction of the material with impurity gases in the primary coolant helium to cause carburization, decarburization, and/or internal oxidation) or by microstructural changes resulting from holding at elevated temperatures for long periods of time (thermal aging).

Early work should be initiated on the turbine shroud material to assure that adequate long-term creep data is available in the temperature range 950 °C to 1050 °C. Long-term in reference to the collar may be relatively short as the collar could be replaced at each 7-year maintenance period; however, it is likely that a much longer life is desired for the insulation package container. Confirmatory demonstrations of the manufacturing processes are needed for the forming and welding procedures required for the turbine inlet shroud collar and the shroud thermal insulation boundary/container and the recuperator.

The situation relative to the turbine blade material is essentially identical to that described above. Temperatures, environments, service periods, and many of the candidate materials are identical. In addition to the creep and environmental work it will be necessary to address questions relative to both low-cycle and high-cycle fatigue at very high temperatures and the effects of gas-metal interactions on fatigue behavior. Creep-fatigue interactions will also require study.

A large number of wrought Ni-base alloys are potentially appropriate to the turbine disk application. Of these, Hastelloy X and Hastelloy XR and Alloy 617 (also a candidate for the turbine inlet shroud collar) have been studied extensively in simulated gas-cooled reactor environments; all have received some attention. Creep and tensile strength data should be available for all candidate materials; further studies will likely be needed on fracture toughness and crack growth properties. Some confirmatory environmental exposures are desirable on the down-selected materials but effects at the temperature of application (~750 °C) should be relatively minimal. Testing efforts aimed at the materials for the recuperator should be minimal. All needed mechanical property data are available; confirmatory environmental exposures are desirable but no adverse effects are expected.

The manufacturing technology is an important issue for the bellows. The hot ducting and bellows operate at 600 °C but could reach 700 °C in event of an accident. Alloy 800H is the leading candidate. Nevertheless, there have been several instances of early failures in bellows fabricated from alloy 800H and operating at temperatures in the range of 600 to 650 °C. These failures may be related to fabrication technology. Some testing will be undertaken to demonstrate that failures of 800H components in the refining and petrochemical industries are understood and can be avoided in the NGNP components. The testing will be largely confirmatory and will include aging effects and environmental effects studies under simple and complex loading conditions.

The helium circulator operates at 600 °C. There are no pressure stresses, but some concern exists in regard to high-cycle fatigue and creep-fatigue. Stainless steels may be considered for this application. However, ferritic steels, such as 2 1/4Cr-1Mo steel, and ferritic/martensitic steels, such as 9Cr-1Mo-V steel, are potential candidates. The material selection will be based to some extent on the fatigue or creep-fatigue resistance of the candidate alloys. It is expected, for example, that the high yield strength of the ferritic/martensitic steels will produce favorable fatigue resistance in the absence of severely oxidizing environments. It is important that an assessment of the loading conditions be undertaken before the leading potential candidates are identified.

FY-06 Activities

There are no plans to fund this work in FY-06.

3.10.2 Turbine and Generator Surface Engineering/Coatings Test Program (Not Funded in FY-05)

Thermal barrier coatings (TBC) have been developed for turbine blades in recent years to provide some thermal insulation between the operating fluid and the metal substrate. In both aircraft and stationary power generation turbines the TBC is a multi-layer system consisting of an insulating ceramic outer layer (typically Y_2O_3-stabilized ZrO_2) on top of a metallic bond coat that is applied to the substrate material. The ceramic layer is deposited using either vapor deposition or thermal spray methods, and contains a carefully designed grain structure and level of porosity that act to decrease the thermal conductivity. Porosity in the ceramic layer allows transport of combustion gases to the bond coat. Bond coats have been developed to resist oxidation in the operating environment and have compositions designated as MCrAlY where M is one or more of the metals Ni, Co or Fe. These compositions form tightly adherent Al or Cr scales that are protective at service temperatures.

Thermal barrier coatings are applied in existing turbine systems to allow operation at higher temperatures in order to increase efficiency. At the expected outlet temperature of the NGNP it is not clear that a TBC will be necessary or desirable. A vast amount of data exists with vendors for turbine applications that will indicate if a TBC needs to be considered.

Should it be determined that a TBC is required, extensive testing and performance validation will be required. TBC systems have been developed for relatively short time service (thousands or tens of thousands of hours) in an oxidizing environment. Testing will be required to determine if the bond coat material will serve to protect the substrate under NGNP conditions where there may be insufficient oxygen partial pressure to maintain a protective scale. Continued performance of TBCs in service is assured by inspection during shut down of the turbine and coatings are refurbished as needed. Periods between service and inspection for the NGNP are likely to be longer; methods to apply more durable coatings or to inspect coatings in service may need to be developed.

FY-06 Activities

There are no plans to fund this work in FY-06.

3.11 RPV Transportation and Fabrication Project (Funding Cancelled in FY-05)

This task was initiated in FY-05 but then cancelled due to the required redistribution of funding within the program.

3.11.1 Task 8A

The issue of RPV heavy section fabrication is a major issue that needs to be evaluated. Several potential candidate pressure vessel steels have been previously identified for the RPV and CV (Table 25). These steels were initially chosen in the order shown for potential operation at temperatures as high as 650 °C, due to considerations of high-temperature strength, maturity of the database, and near-term needs for material selection. For operation at 500 °C, the order would remain the same; however, for operation at 400 °C, the 2.25Cr-1Mo alloy would likely assume a higher position in the hierarchy due to its extensive database, industrial experience, and demonstrated fabricability.

Table 25. Potential candidate steels for the RPV and CV of the NGNP

Material Class	Primary Advantages	Primary Disadvantages
9Cr-1MoVNb	Mature high-strength database; 9Cr-1MoVNb, grade 91 is ASME Code approved to 593 °C for Section III, Class 1; other variants may offer even higher high-temp strength	As with all alloys except for 2.25Cr-1Mo, thick section fabrication must be demonstrated
7-9Cr2WV	Possibility of higher strength than 9Cr-1Mo class; low activation	Smaller database than 9Cr-1Mo class
3Cr-3WV	Apparently very good high-temp strength; maybe less cost with low Cr	Not much data and no data in thick sections
12Cr-1MoWV	HT-9 extensive data. Newer alloys have improved high-temp strength	HT-9 has lower high-temp strength than 9Cr-1Mo class; sparse data for newer alloys
2.25Cr-1Mo	Extensive data and industrial experience	Lower high-temp strength than other classes

From the above list of potential candidate materials, selection of preliminary candidate steels for the RPV and CV will be based on results of an extensive literature review, initial results of experiments with the unirradiated steels, and initial scoping irradiation experiments.

Based on the investigation noted above, a Phase 1 evaluation will be performed of heavy section fabrication technology of the RPV and a letter report issued that summarizes this evaluation.

3.11.2 Task 8B

It is very unlikely that the manufacturing of the RPV would take place in the United States without a significant investment. Preliminary considerations and discussions indicate that Japan Steel Works is the most likely source of forgings of the required size. The physical size of even the largest required forging appears to be within their range of capability; however, the specific material selection is critical in that very large forgings of most of the potential candidate alloys listed have not been manufactured, including the 9Cr-1Mo-V alloy. The main issue is attaining the required through-thickness properties of the higher-alloy steels in such thick sections. Additionally, weldability of the steels in thick sections is also an issue.

However, because of the relatively short lead-time available for ordering of components for the Primary coolant pressure boundary system, fabricability and availability will also be major considerations in the selection of materials. Besides the technical issues, transportation of the completed RPV or even the large ring forgings to the reactor site may be problematic. The diameter of the RPV is relatively well known from the design, but the thickness and, therefore, the weight is not as well known. It is possible that the RPV will require field fabrication, meaning welding of the ring forgings, heads, etc. onsite. In this case, the conduct of Post Weld Heat Treatment (PWHT) takes on more significance in that a PWHT is more difficult to conduct and control than that performed in the shop environment. Additionally, since the flange forging is likely to be the limiting forging component of the RPV, the option of eliminating the flanged closure and instead designing for cutting and rewelding the RPV if access is required may be a valid consideration. This issue has not been addressed at this point, but may be addressed in future revisions.

Fabrication and transportation for the RPV ring forgings are critical issues to be included within the literature review. This review will enlist the assistance of consultants with expertise in large vessel fabrication, particularly with low alloy and medium level chromium ferritic steels. As mentioned earlier, there currently is no domestic manufacturer that can supply the very large ring forgings that are needed for the RPV. Japan Steel Works) appears to have the capacity to produce forgings of the needed size, but there may be other fabricators as well. Fabrication of the RPV with rolled and formed plates joined with longitudinal welds will remain an option, but is not desirable because it results in a significant increase in weldments in the beltline region, the most highly irradiated region, of the RPV. If high chrome low alloy steels are retained as the prime candidate materials, fabrication, heavy-section welding and PWHT will require development. The production of such forgings with the potential candidate alloys will be evaluated during the literature review.

The assessment will also include transportation of individual ring forgings or a partially completed RPV to the United States, and a fully completed RPV to the construction site in Idaho. The assessment will include evaluation of domestic welding and heat-treating capabilities for the potential case of final fabrication of the RPV in the United States and transport of the completed RPV to the construction site. Transportation of other than a fully completed RPV to the construction site will also entail assessment of field fabrication issues and capabilities.

3.11.3 Additional Developmental Tasks Required

The initial purchase of welding consumables will be based on the results of the literature review and preliminary materials selection documents. This test program will include enough materials to ensure the capability to fabricate weldments of sufficient size to enable machining of the numbers of mechanical test specimens needed for inclusion in the baseline, aging, and irradiation tasks. Additionally, development of welding techniques and processes, to include PWHT schedules, will be included in the program for those preliminary candidate materials for which a mature welding technology does not exist.

Field fabrication of any part of the RPV involving welding will likely require development of a PWHT procedure and evaluation of the procedure and the weldments must be included in the testing program. Following identification of fabricators deemed to have the capability for manufacture of the required ring forgings, a fabricator will be chosen to fabricate a forging of sufficient size to represent the largest and thickest one required for the RPV. This forging would be evaluated with mechanical testing and microstructural characterization. As a part of this task, a review of non-destructive examination (NDE) procedures for the preliminary candidate materials will be conducted. If the review indicates the need for development of procedures specific to those materials, NDE procedures will be developed with a view towards satisfying the requirements of the ASME Code and the NRC, to include incorporation of the procedures in the required in-service inspection program.

FY-06 Activities

There are no plans to fund this task in FY-06.

3.12 Reactor Pressure Vessel Emissivity (Not funded in FY-05)

Emissivity data on the various potential candidate materials for the RPV are needed. This is necessary because cooling of the RPV occurs partially by radiation from the outer surface to the air in the cavity between the RPV and surrounding concrete. Further, the pressure vessel must be able to radiate sufficient heat during any anticipated accident conditions throughout the life of the reactor. It is therefore necessary to have a stable, high emissivity surface on the external surface of the pressure vessel at elevated temperatures. Depending on the emissivity of the selected material, it may be necessary to incorporate a high emissivity coating on the outer surface of the RPV. Early testing to establish limitations of potential candidate materials emissivity and the performance and durability of proposed surface modifications to improve emissivity must be performed early to provide design feedback and limitations. Preliminary emissivity screening testing of the potential candidate materials will be performed to determine the detailed experimental program needed for developing a stable surface with minimum emissivity required for adequate cooling of the RPV. Concurrent with that testing, a surface treatment/coatings program will be conducted to investigate the efficacy of various potential concepts for either increasing the emissivity of the RPV materials or providing a coating that would have the required emissivity. If the tests of the potential candidate materials indicate a high probability that the materials will have sufficiently high emissivity under operating conditions, a special coatings development program will not be required.

FY-06 Activities

There are no plans to fund this activity in FY-06.

3.13 Internals Project (Not funded by the NGNP Materials Project in FY-05)

The first step in the research program on materials for the metallic reactor internals will be a comprehensive and detailed review of the potential candidate alloys identified in Table 6. The existing database for those alloys will be assembled, analyzed, and evaluated with respect to the design and operating requirements described above. Principal topics for review will include: high-temperature strength, stability, and long-time performance under irradiation of the materials, effects of helium typical of gas reactor coolant on the mechanical and physical properties of the materials, codification status, prospects, and needs, including maturity and limitations of the CV for each material selected. The status of the joining technology will be reviewed. The weld metal and weldment database will be collected for the candidate alloys. The technology behind the weld strength factors under development by the ASME and other international codes will be reviewed in collaboration with activities on design methodology. The neutron fluences accumulated in the metallic core internal materials are expected to be low relative to the tolerances of the structural alloys. Nevertheless, these will be reviewed and details developed for confirmatory testing and evaluation. Based upon the results of the review, details of the program to evaluate the mechanical and fracture properties of the leading candidates, along with their environmental and irradiation response will be developed. Major anticipated research activities are provided below.

Joining technology will be developed and experimental work started. Weldments will be produced for mechanical testing, aging studies, and microstructural characterization. Creep-rupture and creep crack growth testing will be started. Environmental testing and creep-fatigue will be performed and computational models will be used to predict weld microstructures. Microstructural evaluations will be completed on aged materials. Microstructural parameters will be quantified for use in damage prediction models. Preliminary estimates of weld strength reduction factors will be made. Candidate weld metals will be ranked on the performance. Data will be provided to the design methodology activity to explore the constitutive behavior of weld metal relative to base metal. Weldment test data required for the efforts on design methodology will be produced and testing of welds will establish confidence in the modeling efforts and the code rules developed from testing and modeling.

A *Survey of Metallic Materials for Irradiated Service in Generation IV Reactors and Pressure Vessels*, ORNL/TM-2005/519[71], was performed and funded under materials crosscutting.

3.14 Intermediate Heat Exchanger and Piping Fabrication Test (Not Funded in FY-05)

A detailed assessment of the materials requirements for heat exchanger designs will be undertaken prior to any experimental work. The leading potential candidate alloys will be identified in the course of this assessment. Most likely, these materials will be Alloy 617, Alloy XR, and Alloy X. New alloys such as CCA617, Alloy 740, and Alloy 230 will be considered as alternates. An assessment will be undertaken of the potential of C_f/C composites for the compact IHX.

The baseline materials data generation program for the IHX will focus on the characterization of the material of construction as it is influenced by the specific fabrication procedures needed to produce the compact IHX configuration. The material performance requirements will be developed and a list of leading candidates will be identified. It will be necessary to decide if the fabrication processes should be selected to produce a material of optimum metallurgical condition or if an off-optimum material condition is satisfactory. At 1000 °C, most of the wrought nickel base alloys require relatively coarse grain size for good creep strength but fatigue resistance is best for fine grain size.

Exploratory testing will be undertaken to establish the effect of fabrication variables on the subsequent creep and fatigue properties. Materials of comparable chemistry, grain size, and processing history will be used to produce data, which can then be used to model the performance of the IHX.

It will be determined if the metallurgical state of materials included in the testing program for the core supports and internals are suitable for the IHX. If so, a mechanical testing and aging work on materials for the IHX will not be needed. Bench testing small models of the IHX will be performed to add confidence to life prediction methodologies. Metallurgical evaluations will be undertaken.

Manufacturing issues related to the compact counter-flow IHX will be addressed as part of the research and testing activities. It has yet to be demonstrated that such a unit can be manufactured from the high-temperatures alloys that are the leading candidates, so it is clear that the manufacturing of such a unit will produce several issues to be resolved. Issues include the production of a high-integrity diffusion bond between the sheets of metal used to build the module, the control of conditions that result in an optimum grain size in the metal ligament, the development of methods for NDE of the unit, and the design and fabrication of joints between the unit and the inlet and outlet piping systems.

A part of the development of the fabrication technology for the IHX, the interfaces of the bonded plates will be metallurgically and mechanically evaluated. The specific mechanical tests will be determined after completion of exploratory testing.

A research effort that helps to develop the fabrication technology will be undertaken, and a testing plan will be developed to examine the performance of the configuration under various loading conditions. Included in the testing will be thermal transients.

A review will be undertaken of German and Japanese experiences with materials in “more conventional” IHX units for gas-cooled reactors. Any materials technology needed to advance the conventional units will be identified after this review.

3.15 Hot Duct Liner and Insulations Test (Not funded in FY-05)

Data on the performance of fibrous insulation are needed to ensure that the selected materials are capable of lasting for the life of the plant. 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 or small assemblies of thermal insulation panels and application of appropriate ASTM standards. This standards development work will be supported within this program. Moreover, application of current non-destructive evaluation techniques, especially in support of the monolithic insulators, is 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, this decision will be made following the neutron and gamma irradiation testing.

FY-06 Activities

There are no plans to perform this work in FY-06.

3.16 Valves, Bearings, and Seals Qualification Test (Not funded in FY-05)

The qualification test program for these materials will be added with a later revision of this Program Plan. There is insufficient design information to support such a program at this time.

FY-06 Activities

There are no plans to perform this work in FY-06.

3.17 Management and Administration Tasks (Funded in FY-05)

The NGNP Materials R&D Program Manager is responsible for performing technical and programmatic activities including program management, project administration, progress tracking support and working with the SIM and the NTD for Materials to ensure integration of tasks with the overall NGNP Project and related materials activities. These tasks include program plan development, work package and schedule coordination, interface with program control activities, coordination of technical reviews and development of communication products. This support covers secretarial, travel, project management reserve, miscellaneous expenses, procurement support, and project controls. The Program Manager will also provide support to public and private web sites of the Gen IV International Forum.

The NGNP Materials Program Manager develops and reports the R&D plans, material requirements and qualification plans. Concurrence will occur on scope, schedule and estimated cost with the SIM and the NTD for Materials.

The NGNP Materials Program Manager supports the MRC's work in providing independent assessment of the NGNP Materials R&D Program. The Program Manager identifies ASME and ASTM committees that need to be supported for efforts in developing codes and standards. All procurements and test methods will be reviewed and approved by the Program Manager.

The program performance is measured by incorporating the input from the approved work packages to develop a performance baseline for each fiscal year. Each month the work package managers will determine the percent complete of planned work and the status of reportable milestones. This will be used to calculate the earned value of the work completed for the month. Performance metrics, cost performance index and schedule performance index, shall be calculated as well as an estimate at completion. The Program Manager will identify reasons for out of norm performance and if necessary undertake corrective actions to return the work to within acceptable bounds.

FY-06 Activities

This work will be continued as Activity 701 (ORNL) and Activities 601 and 602 (INL).

4 DELIVERABLES AND MILESTONES

The deliverables listed are for FY-06 based on current projected program funding. The deliverables for follow-on years will be documented in the next revision of the NGNP Materials R&D Program Plan. The deliverables and milestones with an I and O designation are for the INL and the ORNL, respectively.

4.1 Graphite

Deliverable: Act. O103	Provide input documenting fabrication of the specimens for the AGC-1 test to joint INL/ORNL report: AGC-1 final experiment design and test plan	8/31/2006
Deliverable: Act. O106	Issue a draft report: PIE of the high-dose scoping graphite irradiations	4/30/2006
Deliverable: Act. O109	Issue draft report: Status of nuclear graphite model development (joint ORNL/INL report)	6/30/2006
Deliverable: Act. O112	Issue draft report: Test plan and capsule design for high temperature graphite irradiations in HFIR	8/31/2006
Deliverable: Act. O114	Complete the PIE of the high-temperature scoping graphite irradiations in HFIR METS capsules	7/31/2006
Deliverable: Act. O115	Issue draft report: PIE of the high temperature scoping graphite irradiations	7/31/2006
Milestone: Act. O102	Complete fabrication and inspection of specimens for AGC-1 experiment	8/31/2006
Milestone: Act. O105	Complete the PIE of the high-dose scoping graphite irradiations in HFIR rabbit capsules	4/30/2006
Milestone: Act. O108	Complete interim nuclear graphite model development including incorporation of new data and FEMLAB and CARES codes work at INL (D1O9)	6/30/2006
Milestone: Act. O111	Complete test plan development for high temperature capsule irradiation in HFIR perform capsule design and initiate fabrication of the capsule (D1O12)	8/31/2006
Deliverable: Act. I104	Issue draft report: Design and fabrication of AGC-1 gas control system	8/31/2006
Deliverable: Act. I108	Issue draft report: AGC-1 final design and test plan (joint INL/ORNL report)	8/31/2006
Deliverable: Act. I110	Provide input to draft Level 2 ORNL/INL joint report: status of nuclear graphite model development and development of CARES-FEMLAB module and example problems.	6/30/2006
Milestone: Act. I103	Complete the design and fabrication of AGC-1 gas control system (D1I4)	8/31/2006
Milestone: Act. I107	Complete AGC-1 final experiment design and test plan (D1I10)	8/31/2006

4.2 HTDM

Deliverable: Act. O202	Provide input to joint INL/ORNL report: Procurement and initial characterization of controlled material specification of Alloy 617 and Alloy 230.	4/30/2006

Deliverable: Act. O205	Issue Draft report: Status of testing and characterization of CMS Alloy 617 and Alloy 230 (joint ORNL and INL report)	8/31/2006
Deliverable: Act. O209	Issue draft report: Interim development of the methods for very high temperature metallic design	7/31/2006
Milestone Act. O204	Complete initial scoping testing and characterization of CMS alloy 617 and Alloy 230	8/31/2006
Milestone: Act. O208	Complete interim development of the methods for very high temperature metallic design	7/31/2006
Deliverable: Act. I203	Issue draft letter report: Procurement and checkout of mechanical test load-frame (INL report)	2/28/2006
Deliverable: Act. I206	Issue draft letter report: Procurement and initial characterization of controlled material specification of Alloy 617 and Alloy 230 (joint INL/ORNL report)	4/30/2006
Deliverable: Act. I211	Provide input to the draft level 2 ORNL/INL joint report: Status of testing and characterization of Alloy 617 and CMS Alloy 617 and Alloy 230	8/31/2006
Milestone: Act. I202	Complete procurement of load-frame	2/28/2006
Milestone: Act. I205	Complete procurement of Alloy 203 material	4/30/2006

4.3 Code Support

Deliverable: Act. O303	Issue draft report: Status of graphite core support working group (joint ORNL/INL report)	3/30/2006
Deliverable: Act. O306	Issue draft letter report: Status of ASME Subsection NH (joint ORNL/INL report)	3/30/2006
Deliverable: Act. O310	Issue draft letter report: Status of development of ASTM DO2-F standard test method for air oxidation of graphite (joint ORNL/INL report)	7/30/2006
Milestone: Act. O302	Provide update on status of graphite core support working group	3/30/2006
Milestone: Act. O305	Provide update on status of ASME Subsection NH	3/30/2006
Milestone: Act. O309	Provide update on status of development of ASTM DO2-F standard test method for air oxidation of graphite	7/30/2006
Deliverable: Act. I302	Provide input to level 3 draft ORNL/INL letter report: status of graphite core support working group	3/30/2006
Deliverable: Act. I304	Provide input to Level 3 draft ORNL/INL letter report: status of ASME Subsection NH Activities	3/302006
Deliverable: Act. I307	Provide input to Level 3 draft ORNL/INL letter report: Status Report on the Development of the Air Oxidation Test Method for Graphite	7/30/2006
Deliverable: Act. I310	Issue draft INL/ORNL report: Development of a Fracture Toughness Testing Standard for Nuclear-Grade Graphite Materials, Final Report	9/15/2006
Milestone: Act. I309	Complete development of a fracture toughness testing standard for nuclear-grade graphite materials (INL/ORNL joint activity	9/15/2006

4.4 ETTA

Deliverable: Act. O402	Provide input to joint INL/ORNL report: Current knowledge of high temperature gas phase kinetics and gas/metal reactions for various prototype VHTR atmospheres	4/30/2006
Deliverable: Act. O403	Provide input to joint INL/ORNL report: "Summary and analysis of environmental and thermal aging testing performed on Alloy 617 and CMS alloy 617 and Alloy 230".	9/15/2006
Deliverable: Act. I403	Issue draft INL/ORNL report: Current knowledge of high temperature gas phase kinetics and gas/metal	6/30/2006
Deliverable: Act. I406	Issue Draft INL/ORNL report: Summary and Analysis of Environmental and Theram Aging Testing Performed on Alloy 617 and CMS Alloy 617 and Alloy 230	9/15/2006
Milestone: Act. I402	Complete initial studies on kinetics of the gas phase and gas-metal reactions for various prototype VHTR atmospheres	6/30/2006
Milestone: Act. I405	Perform post exposure examination on microstructure and mechanical properties on specimens tested	9/15/2006

4.5 Irradiation Facility

Deliverable: Act. O504	Issue draft report: Site selection and design concept for low-flux irradiation facility	9/15/2006
Milestone: Act. O503	Complete documentation of site selection and design concept of low-flux irradiation facility	9/15/2006

4.6 Composite

Deliverable: Act. O602	Provide input to joint INL/ORNL report: Procurement, Fabrication of Test Specimens and Characterization Performed on SiC/SiC and C/C Structural Composites in Preparation for Irradiation Effects Evaluations	7/30/2006
Deliverable: Act. O606	Issue draft report: "Post Irradiation Evaluation of SiC/SiC and C/C Composites after 10 dpa Exposure"	7/30/2006
Deliverable: Act. O609	Issue draft report: "Baseline characterization of C/C Control Rod Cladding Material"	8/31/2006
Milestone: Act. O605	Complete the PIE of the specimens removed from the 10 dpa capsules tested in HFIR and document the results	7/30/2006
Milestone: Act. O608	Complete characterization of C/C control rod cladding materials and preparations for C/C screening irradiations	8/31/2006
Deliverable: Act. I503	Issue draft INL/PNNL letter report: Summary and Analysis of Environment And Thermal Aging Testing Performed on Alloy 617 and CMS Alloy 617 and Alloy 230	9/15/2006
Deliverable: Act. I506	Issue draft INL/ORNL report: Procurement, Fabrication of Test Specimens and Characterization Testing Performed on SiC/SiC and C/C Structural Composites in Preparation for Irradiation Effects Evaluations	7/30/2006
Deliverable: Act. I510	Issue draft INL report: VHTR Structural Composites	9/15/2006

	Preliminary Irradiation Creep Experiment Plan and Specimen Design	
Milestone: Act. I502	Complete PNNL time-dependent modeling	9/15/2006
Milestone: Act. I505	Complete high temperature creep and strength testing on selected structural composites in support of planned irradiation experiments	7/30/2006
Milestone: Act. I509	Complete preliminary specimen design and irradiation creep experiment plan	9/15/2006

5 COLLABORATIONS

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, and the United States–and the European Union. 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. The minutes of the first meeting of the first meeting of this PMB are given in Appendix B. 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 quarterly basis in various locations in the world. This board will be addressing each materials R&D program area noted previously and will develop detailed collaboration plans for each of these areas. These plans are being developed in the same approximate order of priority noted in Section 1.3. It is currently envisioned that this process will not be fully developed and implemented until the end of 2006, however, as each plan is developed, implementation of collaboration activities will begin immediately. Currently, the collaboration plan for nuclear graphite R&D is being developed and should be available by April 2005. This will allow further discussion and development of this plan at the next Materials and Components PMB meeting at ORNL scheduled at that time.

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 supported by the board 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.

To make efficient use of program resources, the development of the required databases and methods for their application, it will also be useful to incorporate both the extensive results from historic and ongoing programs in the United States and abroad outside the GIF partnership that address related materials needs. These would include, but not be limited to, DOE, NRC, and industry programs on liquid-metal-, gas-, and light-water-cooled reactor, fossil-energy, space-reactor, and fusion materials research programs.

Two GIF M&C PMB meeting were held during FY-05. The first meeting was held at ORNL in April and the second meeting was held in Paris in September. . The focus of PMB activities in FY-05 was to finalize the collaboration plan for graphite R&D and to develop a similar plan for high temperature metallic materials and design methods. Potential areas for additional collaboration plans in structural composites and oxide dispersion strengthen materials for longer term, higher temperature systems were also considered. Minutes of meetings and other information are available by contacting the co-chair at george.hayner@inl.gov.

FY-06 Activities

Two meeting are also planned in FY-06. The first meeting will be held in Korea in April 2006 and the second meeting is not scheduled at the present time.

6 I-NERI COLLABORATIONS

International Nuclear Engineering Research Initiatives (I-NERI) 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. Two I-NERI collaborations have been proposed between the United States and France, the United States and Japan, and the United States and Korea.

6.1 France

A three-year I-NERI grant between U.S. - French research institutions (INL, ORNL, PNNL, CEA, and University of Bordeaux) has been approved for R&D 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. Mechanical, thermal, and radiation-damage response of the French fabricated tubular 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 will benefit from the United States' full-scale composite testing and irradiation program. The U.S. research program is much more focused upon application oriented testing and verification. 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 for these exchanges.

6.2 Japan

A U.S.-Japan I-NERI is currently being discussed and negotiated. The proposed research will investigate development issues surrounding tubular C/C composite material for control rod and guide tube applications. Similar to the SiC/SiC composite research, the mechanical, thermal, and radiation-damage response of both the U.S. and the Japanese fabricated tubular composites will be studied[72].

6.3 Korea

A U.S.-Korean I-NERI is currently being discussed and negotiated. The proposed research will investigate development issues surrounding the use of high temperature materials for reactor service. This proposed I-NERI would combine components of work on irradiation effects, environmental effects, and development of a materials handbook containing properties of high-temperature metallic materials. The U.S. Gen IV contributions will include selected, interrelated activities from four separate work packages: two WPs from Materials Crosscutting on Materials for Radiation Service and Materials for High-Temperature Service, respectively and the two NGNP Materials at INL and ORNL.

The primary contributions from the U.S. Gen IV Program will comprise development of effects of VHTR-type helium environments on the mechanical properties of high-temperature metallic alloys and the establishment of the *Gen IV Materials Handbook* containing a database on high-temperature materials.

The primary contributions of KAERI will be the development of effects of irradiation on high-temperature metallic alloys in the HANARO reactor and their irradiated materials evaluation facility (IMEF).

7 PROGRAM COST AND SCHEDULE

7.1 Program Schedule

The NGNP summary program schedule can not be projected at this point and therefore has been eliminated from the current Program Plan revision.

The materials R&D program is designed to deliver materials data and recommendations that will support the NGNP design process. Final materials design data and the final materials selection reports need to be available to complete the preliminary design and are essential at the start of the design to support the initiation of long lead procurements for reactor components.

7.2 Cost and Schedule Estimates

The overall cost for the NGNP Materials R&D Program was estimated to be $212.5 M, however, a revised estimate was not made in FY-05 and has been deleted from the current Program Plan revision. The costs for the NGNP material program are broken down for fiscal years FY-04, FY-05 and projected for FY-06 in Table 26. All FY-06 cost projections are subject to change. Cost projections through FY-15 were listed in Revision 1 of the NGNP Materials R&D Program Plan, however, based on several factors, these projections can no longer be made and are not listed in the current Program Plan revision.

Table 26. Summary Cost ($K)

Major Program Elements	Totals	FY-04	FY-05	FY-06
Graphite	4385		1540	2845
Hi Temp Design Methods	2744	300	1409	1035
Code & Standards	1018		500	518
Environ Testing & Aging	1747	300	750	697
RPV Irradiation Facility	600		350	250
Structural Composites	2699		1074	1625`
Database & Handbook				
Turbine & Generator				
RPV Transport & Fab				
RPV Emissivity				
Metallic Core Internals				
Hot Duct and Insulation				
IHX and Piping				
IHX Pressure Vessel				
Valves, Bearings, Seals				
Administration	1970	400	1040	530
Totals	**15163**	**1000**	**6663**	**7500**

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