INL/EXT-06-11701
Rev. 4

Next Generation Nuclear Plant Materials Research and Development Program Plan

T. Burchell
R. L. Bratton
R. N. Wright
J. Wright

September 2007

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

INL/EXT-06-11701
Rev. 4

Next Generation Nuclear Plant Materials Research and Development Program Plan

T. Burchell
R. L. Bratton
R. N. Wright
J. Wright

September 2007

Idaho National Laboratory
Idaho Falls, Idaho 83415

Prepared for the
U.S. Department of Energy
Office of Nuclear Energy
Under DOE Idaho Operations Office
Contract DE-AC07-05ID14517

This page intentionally left blank.

Next Generation Nuclear Plant Materials Research and Development Program Plan

INL/EXT-06-11701
Revision 4

September 2007

Authored by:

D Alleman for R. Wright	9/27/07
R. Wright	Date

Reviewed by:

	9/27/07
W. Windes NGNP Materials R&D Lead	Date

Approved by:

	9/27/2007
M. Patterson NGNP Project Manager	Date

This page intentionally left blank.

Executive Summary

DOE has selected the High Temperature Gas-cooled Reactor (HTGR) design for the Next Generation Nuclear Plant (NGNP) Project. The NGNP will demonstrate the use of nuclear power for electricity and hydrogen production. It will have an outlet gas temperature in the range of 950°C and a plant design service life of 60 years. The reactor design will be a graphite moderated, helium-cooled, prismatic or pebble-bed reactor and use low-enriched uranium, TRISO-coated fuel. The plant size, reactor thermal power, and core configuration will ensure passive decay heat removal without fuel damage or radioactive material releases during accidents. The NGNP Materials Research and Development (R&D) Program is responsible for performing R&D on likely NGNP materials in support of the NGNP design, licensing, and construction activities. Some of the general and administrative aspects of the R&D Plan include:

- Expand American Society of Mechanical Engineers (ASME) Codes and American Society for Testing and Materials (ASTM) Standards in support of the NGNP Materials R&D Program.
- Define and develop inspection needs and the procedures for those inspections.
- Support selected university materials related R&D activities that would be of direct benefit to the NGNP Project.
- Support international materials related collaboration activities through the DOE sponsored Generation IV International Forum (GIF) Materials and Components (M&C) Project Management Board (PMB).
- Support document review activities through the Materials Review Committee (MRC) or other suitable forum.

Graphite

The Next Generation Nuclear Plant (NGNP) will be a helium cooled High Temperature Gas Reactor (HTGR) with a large graphite core. Graphite physically contains the fuel and comprises the majority of the core volume. Graphite has been used effectively as a structural and moderator material in both research and commercial high temperature gas-cooled reactors. This development has resulted in graphite being established as a viable structural material for HTGRs. While the general characteristics necessary for producing nuclear grade graphite are understood, historical “nuclear” grades no longer exist. New grades must be fabricated, characterized and irradiated to demonstrate that current grades of graphite exhibit acceptable non-irradiated and irradiated properties upon which the thermomechanical design of the structural graphite in NGNP is based. This technology development plan establishes the research and development activities and associated rationale necessary to qualify nuclear grade graphite for use within the Next Generation Nuclear Plant (NGNP) reactor.

Background information from past graphite reactor experience, other relevant graphite grades, and the technology developed for past gas reactors is presented to provide a perspective on what has been achieved previously in this area of research. Next the technology required to qualify the graphite for use in NGNP will be developed based on the historical graphite fabrication and performance database, the anticipated NGNP graphite design service conditions, and gaps in the fabrication and performance database.

The resultant data needs are outlined and justified from the perspective of reactor design, reactor performance, or the reactor safety case. The approach will allow direct comparison between data needs and the resulting technology development activities. Irradiated and non-irradiated characterization of material properties and the inter-relationships between the experimental and modeling activities will be presented to establish a complete picture of the graphite technology requirements. Finally, the variables affecting this R&D program are discussed from a general perspective. External factors that can significantly affect the R&D program such as funding, schedules, available resources, multiple reactor designs, and graphite acquisition are discussed. Materials Plan elements specific to graphite technical development include:

- Validate cutting plan for characterization, review testing methodologies and optimal specimen size for characterization as applicable to grain sizes.

- Execute thermo-mechanical testing of graphite and evaluate strength, thermal limits, and oxidation resistance. Establish non-irradiated baseline properties for comparison to irradiated material property changes.

- Design and build radiation experiments and determine high temperature limits and dose (dpa) on small graphite specimens for irradiation material property characterization. Use PIE to evaluate and validate material models for the irradiation induced dimensional and material property changes in graphite components and throughout the core.

- Qualify graphite components for ASME code case and define NQA-1 qualification requirements for graphite vendors.

High Temperature Materials

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 ASME Codes for reactor applications reach about 800°C but some primary system components for the NGNP will require use of materials at higher temperatures. Although outside the scope of potential postulated accident conditions resulting in temperatures above nominal operational temperatures would suggest the use of composite or ceramic materials. 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.

The primary 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 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 areas are currently addressed in the R&D program:

- Develop 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 HTGR.

- Develop a materials handbook/database in support of the Generation IV Materials Program, collecting and documenting in a single source the information generated in this and previous HTGR materials R&D programs.

- Develop an improved understanding of, and models for, the environmental effects and thermal aging of the metallic alloys for potential application in the NGNP.

- Develop, evaluate and certify welding and joining procedures for the various materials and components, including very thick plate and thin sheets.

Other program elements may be added or deleted in the future as required to support development of the NGNP.

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.

This page intentionally left blank.

Contents

Figures

Tables

Acronyms

AGCNR	Advanced Gas-Cooled Nuclear Reactor
AGC	Advanced Graphite Capsule
AGR	Advanced Gas-Cooled Reactor
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATR	Advanced Test Reactor
AVR	Albeitsgemeinschaft Versuchsreaktor
B&PV	Boiler and Pressure Vessel
BWR	Boiling Water Reactor
C_f/C	Carbon/Carbon Composite
CRBRP	Clinch River Breeder Reactor Plant
CT	X-ray Tomography
CTE	Coefficient of thermal expansion
DCC	Depressurized Conduction Cool-down
DHI	Doosan Heavy Industries, South Korea
DOE	Department of Energy
DSC	Differential Scanning Calorimetry
ETD	elevated temperature design
FEM	Finite Element Model
FSAR	Final Safety Analysis Report
GA	General Atomics
GFR	Gas-cooled Fast Reactor
GIF	Generation IV International Forum
GT-MHR	Gas Turbine-Modular Helium Reactor
HFIR	High-Flux Isotope Reactor

HTDM	high-temperature design methodology
HTGR	High-Temperature Gas Reactor
HTR	High-Temperature Reactor
HTTR	High-Temperature Engineering Test Reactor
HTV	High Temperature Vessel
IHX	Intermediate heat exchanger
INERI	International Nuclear Energy Research Initiative
INL	Idaho National Laboratory (formerly the Idaho National Engineering and Environmental Laboratory)
ISI	In-Service Inspection
ITRG	Independent Technical Review Group
KAERI	Korean Atomic Energy Research Institute
LBB	Leak Before Break
LMFBR	Liquid Metal Fast Breeder Reactor
LFR	Lead-cooled Fast Reactor
LRFD	Load and Resistance Factor Design
LWR	Light-Water Reactor
M&C	materials and components
MRC	INL Materials Review Committee
MSR	Molten Salt Reactor
MTR	Material Test Reactor
NDE	nondestructive examination
NE	DOE Office of Nuclear Energy
NERAC	Nuclear Energy Research Advisory Committee
NERI	Nuclear Energy Research Initiative
NGNP	Next Generation Nuclear Plant

NRC	Nuclear Regulatory Commission
ODS	Oxide Dispersion-Strengthened
ORNL	Oak Ridge National Laboratory
PBMR	Pebble Bed Modular Helium Reactor
PCHE	Printed Circuit Heat Exchangers
PBR	Pebble Bed Reactor
PIE	Post Irradiation Examination
P-LOFC	Pressurized Loss-of-Forced-Convection
PMB	GIF HTGR Materials and Components Project Management Board
PMR	Prismatic Modular Reactor
PSAR	Preliminary Safety Assessment Report
PWR	Pressurized Water Reactor
PWHT	Post Weld Heat Treatment
QA	Quality Assurance
RCS	Reactivity Control System
R&D	Research and Development
RPV	Reactor Pressure Vessel
SCWR	Supercritical Water Reactor
SGL	Name of a graphite company
SFR	Sodium-cooled Fast Reactor
SiC_f/SiC	Silicon-Carbide Fiber/Silicon-Carbide composite
SS	Stainless Steel
THTR	Thorium Hochtemperatur Reaktor
TRISO	Tri-isotopic (fuel)
UIUC	University of Illinois in Urbana-Champaign
UT	Ultrasonic Testing

This page intentionally left blank

Next Generation Nuclear Plant

Materials Research and Development Program Plan

1. Introduction and Purpose

The Department of Energy (DOE) has selected the High Temperature Gas-cooled Reactor (HTRG) design for the Next Generation Nuclear Plant (NGNP) Project. The NGNP will demonstrate the use of nuclear power for electricity and hydrogen production. The reactor design will be a graphite moderated, helium-cooled, prismatic or pebble-bed, thermal neutron spectrum reactor. The NGNP will use very high burn-up, low-enriched uranium, Tri-Isotopic (TRISO)-coated fuel and have a projected plant design service life of 60 years. The HTGR 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 basic technology for the NGNP was established in former high-temperature gas-cooled reactor plants (e.g., DRAGON, Peach Bottom, Albeitsgemeinschaft Versuchsreaktor [AVR], Thorium Hochtemperatur Reaktor [THTR], and Fort St. Vrain). These reactor designs represent two design categories: the Pebble Bed Reactor and the Prismatic Modular Reactor (PMR). Commercial examples of potential NGNP candidates are the Gas Turbine-Modular Helium Reactor (GT-MHR) from General Atomics (GA), the High Temperature Reactor concept (ANTARES) from AREVA, and the Pebble Bed Modular Reactor (PBMR) from the PBMR consortium. Furthermore, the Japanese High-Temperature Engineering Test Reactor (HTTR) and Chinese High-Temperature Reactor (HTR) are demonstrating the feasibility of the reactor components and materials needed for NGNP (The HTTR reached a maximum coolant outlet temperature of 950 °C in April 2004). Therefore, the NGNP is focused on building a demonstration plant, rather than simply confirming the basic feasibility of the concept.

For graphite, lifetime predictions of graphite components with the service demands and reactor schedule anticipated for NGNP is a practical but much more complex problem than simply determining whether a graphite type is more stable or less stable in an irradiated environment. Graphite properties such as strain to failure, dimensional change rate, and irradiation dependence of thermal expansion coefficient can constrain the reactor design through limiting lifetimes for critical components. For example, irradiation induced dimensional changes to graphite can be severe enough to require limiting the temperature and flux gradients within graphite components or possibly requiring the need for added design features to physically hold components in position over time. This plan will address the main issues of importance necessary for establishing reliable graphite component lifetime predictions.

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 ASME Codes for reactor applications reach about 800°C but some primary system components for the NGNP will require use of materials at higher temperatures. 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. Potential postulated accident conditions with associated temperatures above nominal operational temperatures would dictate the use of composite or ceramic. 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 Research and Development (R&D) Program.

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 NGNP Material R&D Program Elements

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

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

The objective of the NGNP Materials R&D Program is to provide the essential materials R&D needed to support the design and licensing of the reactor and balance of plant, excluding the hydrogen plant. The materials R&D program was initiated prior to the design effort to ensure that materials R&D activities are initiated early enough to support the design process. The thermal, environmental, and service life conditions of the NGNP will make selection and qualification of the graphite and 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:

- Expand American Society of Mechanical Engineers (ASME) Codes and American Society for Testing and Materials (ASTM) Test Standards in support of the NGNP Materials R&D Program.
- Define and develop inspection needs and the procedures for those inspections.
- Support selected university materials related R&D activities that would be of direct benefit to the NGNP Project.
- Support international materials related collaboration activities through the DOE sponsored Generation IV International Forum (GIF) Materials and Components (M&C) Project Management Board (PMB).

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

- Validate cutting plan for characterization, review testing methodologies and optimal specimen size for characterization as applicable to grain sizes in nuclear graphites.

- Execute thermo-mechanical testing of graphite and evaluate strength, thermal limits, and oxidation resistance. Establish baseline properties for comparison to PIE.

- Design and build radiation experiments and determine high temperature limits and dose (dpa) on small graphite specimens for irradiation material properties. Use PIE to evaluate and validate models for the irradiation induced dimensional and material property changes.

- Qualify graphite components for ASME code case and define NQA-1 qualification requirements for graphite vendors.

- Develop 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 HTGR.

- Develop an improved understanding of, and models for, the environmental effects and thermal aging of the metallic alloys for potential application in the NGNP.

- Develop, evaluate and certify welding and joining procedures for the various materials and components, including very thick plate and thin sheets.

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

1.2 Mission Statement and Assumptions

The mission of the NGNP Materials Program is noted below:

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

- Provide materials related support required to meet the goals given in the PPMP

- Provide leadership and support related to international collaboration associated with HTGR materials R&D issues

- Provide leadership and support associated with university materials R&D programs that are of direct benefit to the objectives of the NGNP Program

- Initiate materials related studies as required to support the NGNP Program

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

1. The NGNP Program including the materials program will continue to be directed by the Idaho National Laboratory (INL) based on the guidelines given in the Energy Policy Act of 2005. The scope of work will be adjusted to reflect the level of congressional appropriations.

2. The reactor design has not been formally selected but is assumed to be either a helium-cooled, prismatic graphite block or a pebble-bed core design fueled with TRISO-design fuel particles in carbon-based compacts or pebbles.

3. The NGNP must demonstrate the capability to obtain an Nuclear Regulatory Commission (NRC) operating license; however, the licensing strategy for the NGNP has not been developed to date. It is assumed that the design, materials, and construction will need to meet appropriate Quality Assurance (QA) methods and criteria and other nationally recognized codes and standards.

4. The NGNP is expected to be a full-sized reactor plant based on the reactor concept selected (400-600 MWt) capable of electricity generation with a hydrogen demonstration unit of appropriate size.

5. The demonstration plant will be designed to operate for a nominal 60 years.

6. Application for an NRC operating license and fabrication of the NGNP will occur with direct interaction and involvement of one or more commercial organizations.

1.3 NGNP Materials Program Scope

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

1. Qualification and testing of nuclear graphite 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 Reactor Pressure Vessel (RPV) are covered in this task.

3. Expansion of ASME Codes and ASTM Test Standards in support of the NGNP Materials R&D Program.

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

5. Support of a program to study, design, test, and qualify potential candidates for use as NGNP components.

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

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

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

1.3.1 Program Organization

The NGNP Program is the primary interface with DOE. Program execution is performed by principal investigators who are scientific specialists related to the specific program area being addressed under the direction of the NGNP Research and Development Director.

1.3.2 Materials Review Committee

An MRC was established in FY-04 as a senior independent review body for the materials R&D program. This Committee was suspended in FY-07.

1.3.3 Relationship with the Generation IV International Forum

The primary mechanism for international collaboration for materials R&D activities in support of the HTGR is through the GIF. The GIF is an international effort to advance nuclear energy to meet future energy needs of ten countries–Argentina, Brazil, Canada, France, Japan, the Republic of Korea, the Republic of South Africa, Switzerland, the United Kingdom, the United States–and the European Union.

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 the current generation of commercial nuclear plants that are producing electricity today. These plants include PWR, BWR, Canadian Deuterium-Uranium, and Advanced Gas Reactor (AGR) plants. Generation III plants are considered to be advanced Light Water Reactors (LWR) 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.

The primary mechanism for collaboration of materials R&D for the HTGR is through the Materials and Components PMB. This board is currently composed of members from France, Switzerland, Japan, Korea, South Africa, the United Kingdom, the United States, and the EU and usually meets on a semi-annual basis. The board has the charter to address each materials R&D program area and is developing detailed collaboration plans for each. As each plan is developed, collaboration will begin immediately.

It is currently envisioned that collaboration will involve (as a minimum) the coordination of test and irradiation programs, purchase of testing materials, use of specialized testing facilities, support for the establishment of an integrated Generation IV materials data base and support of codes and standards committees. These collaboration activities are expected to result in a spirit of cooperation between the participating countries, the acceleration of design and licensing activities of HTGR systems and the reduction of the cost for materials R&D.

1.3.4 INERI Collaboration Programs

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

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

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

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

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

1.3.5 University NERI Related Collaboration Programs

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

2. Summary of Pre-Conceptual HTGR Designs and Material Issues

The High Temperature Gas Reactor (HTGR) is an inherently safe nuclear reactor concept with an easily understood safety basis that permits substantially reduced emergency planning requirements and improved siting flexibility compared to current and advanced light water reactors. The viability of a graphite core planned for the NGNP has previously been demonstrated in former high-temperature gas-cooled reactor plants (e.g., DRAGON, Peach Bottom, AVR, THTR, and Fort St. Vrain). 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 (HTTR reached a maximum coolant outlet temperature of 950°C in April 2004). These reactor designs represent two categories: the Pebble Bed Reactor (PBR) and the Prismatic Modular Reactor (PMR).

This section describes the current HTGR pre-conceptual designs in summary form which are described in detail in the NGNP Pre-Conceptual Design Report [1]. In FY07 pre-conceptual design work was initiated by the NGNP Project at the INL. This work was completed by three contractor teams with extensive experience in HTGR technology, nuclear power applications and hydrogen production. Each contractor developed a recommended design for NGNP and a commercial version of the HTGR. R&D, data needs, and future studies required to achieve operation of the NGNP were identified as part of the work. The three designs developed are as follows:

1. General Atomics; The GT-MHR concept.
2. AREVA NP, Inc.; The ANTARES concept.
3. Westinghouse Electric Company, LLC, The PBMR concept.

All three designs utilize TRISO fuel, graphite moderation and high temperature helium coolant in the primary system in the 800^0C -950^0C temperature range. All of the concepts feature various passive neutronic design features which result in a core with relatively low power density and a negative temperature coefficient of neutron reactivity. The shut-down cooling system, the secondary reactivity shut-down system, and the control rod design are all similar among the three designs. All of the reactor concepts could be used as a basis for the NGNP HTGR concept. The designs will not be presented in detail here. Primarily the features that relate to material selection and challenges will be discussed. The key operating parameters and design features for all three designs are listed in Table 1 along with information for the Fort St. Vrain high-temperature gas reactor, the largest and most recent gas-cooled reactor to operate in the U.S.

Table 1. Key operating parameters for the NGNP designs and the Fort St. Vrain HTGR.

Condition or Feature	Fort St. Vrain HTGR	General Atomics GT-MHR	AREVA ANTERES	Westinghouse PBMR
Power Output [MW(t)]	842	550-600	565	500
Average power density (w/cm^3)	6.3	6.5		4.8
Power Conversion Configuration	Direct	Direct	Indirect	Indirect
PCS Cycle Type	Reheat Steam	Brayton	Steam Rankine	Rankine
Secondary Fluid	Steam	He	He	He-N
Moderator	Graphite	Graphite	Graphite	Graphite
Core Geometry	Cylindrical	Annular	Annular	Annular
Reactor type	Prismatic	Prismatic	Prismatic	Pebble Bed
Safety Design Philosophy	Active	Passive	Passive	Passive
Plant Design Life (Years)	30	60	60	60
Core outlet temperature (°C)	785	Up to 950	950	950
Core inlet temperature (°C)	406	490 - 590	500	400
Fuel – Coated Particle	HEU-Th/ ^{235}U (93% enriched)	TRISO UCO (startup UO_2)	TRISO UCO (backup UO_2)	TRISO UO_2
Coolant Pressure (MPa)	4.8	7	5	9
Coolant Flow Rate (kg/s)	428	320	240	193
Fuel Max Temp – Normal Operation (°C)	1260	1250	1300	1057
Fuel Max Temp – Emergency Conditions (°C)	NA - Active Safety System cools fuel.	1600	1600	1600
RPV Design	Not Cooled	Not Cooled	Not Cooled; insulated?	Cooled by primary coolant
RPV size (m) dia x length	18 x 32	8.2 x 31	7.5 x 25	6.8 x 30
RPV Material	Pre-stressed concrete	2.25Cr-1Mo or Mod. 9Cr-1Mo	Modified 9Cr-1Mo	508/533
IHX Design Power	NA		Shell & Tube	
Process		PCHE	PCHE or Fin-Plate	PCHE
IHX Material	NA	In-617	In-617	In-617
Core Barrel Design		Not specified	Double wall	Single wall
Core Barrel Material	steel	Not specified	800H	316H SS
Control Rod Cladding	800H	800H (backup C_f/C composite)	C_f/C composite	800H

2.1 General Atomics – GT-MHR Concept

General Atomics recommended a prismatic reactor design. Figure 1 is a cutaway view of the reactor vessel showing details of the inside of the core. They argue that a prismatic reactor inherently allows higher reactor power levels, resulting in better plant economics, involves fewer uncertainties (and therefore less risk) and allows more flexibility with respect to the use of alternate fuel cycles, such as those fabricated from surplus weapons grade plutonium or transuranics separated from spent LWR fuel.

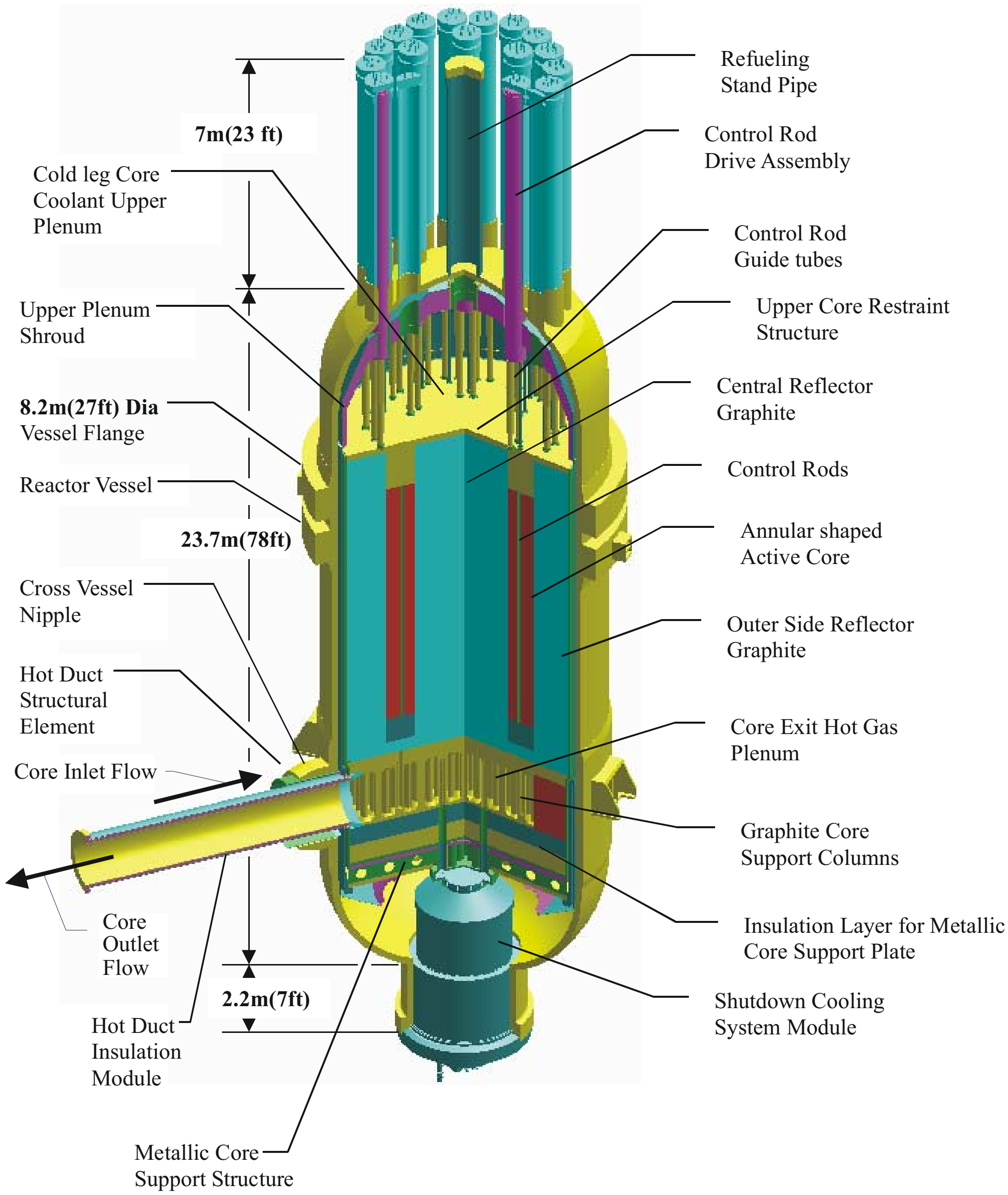


Figure 1. GT-MHR reactor system cutaway showing the metallic internals structures, core, control rod guide tubes, and shutdown cooling system.

The core consists of graphite blocks with an annular-fueled region of 1020 prismatic fuel blocks arranged in three columns, surrounded by reflector elements; the center of the core is a non-fueled graphite reflector. The reflectors act as a heat sink, and boron pins placed in the outer reaches of the reflectors reduce thermal neutron fluxes on the metallic internal structures and reactor vessel.

The reactor helium inlet coolant flows upward in the annulus between the reactor pressure vessel and the metallic core barrel surrounding the side reflector. The reactor pressure vessel upper head is protected by fibrous "Kaowool" insulation blankets supported by high-temperature metallic plates. The insulation protects the head from hot plumes that could occur during a pressurized loss-of-forced-convection (P-LOFC) accident. Between the core exit plenum and the bottom metallic core support plate is an insulation layer ~1.2 meters thick. It is composed of a meter of nuclear graphite and 200 mm of carbon-carbon composite blocks.

The inlet flow then passes down through the core's upper support elements, which are made of carbon-carbon composite material that must also withstand the hot gases in a long-term P-LOFC. The coolant then flows primarily into the coolant channel holes in the fuel elements. Some of the flow bypasses these channels, passing through the gaps between the fuel elements and reflector blocks. Thus the temperature rise of the coolant in the various flow paths through the core varies over a wide range. Good mixing of the outlet coolant is needed to avoid excessive thermal stresses in the downstream components resulting from large temperature gradients and fluctuations, and to assure that the gas entering the turbine has a uniform mixed mean temperature.

2.1.1 Reactor Pressure Vessel

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

The RPV of the GT-MHR is approximately 31 m high, 8.2 m in diameter, and 281 mm thick. The reference RPV material is modified 2-1/4Cr-1Mo, but this particular material has low strength at the temperatures of interest, which will require very thick sections. The 2-1/4Cr-1Mo-V material has better strength at the temperatures of interest (similar to mod 9Cr-1Mo), but is not in Section III (nuclear section) of the ASME code. General Atomics is also considering a design alternative to incorporate cooling of the reactor vessel, which could potentially lower reactor vessel temperatures to a level that would allow use of proven light water reactor vessel materials (e.g., SA508/SA533 steel).

2.1.2 Core Barrel

The core barrel material was not specified in the pre-conceptual design report.

2.1.3 Reactivity Control Rods

The Reactivity Control System (RCS) consists of 36 control rods and 12 start-up rods. Neutron absorber compacts are enclosed in Incoloy 800H canisters for structural support. Alternately, carbon-fiber reinforced carbon (C_f/C) composite canisters may be used for structural support; however use of C_f/C composite materials for the control rod cladding requires significant R&D actions to qualify this component and facilitate its approval by the Regulator.

2.1.4 Cross Vessel

General Atomics defined two other vessels – the cross-vessel and intermediate heat exchanger vessel, which will both be made out of the same material as the RPV. The cross-vessels connect the lower portion of the RPV to the lower portion of the power conversion system and IHX vessels. The cross-vessels include a concentric duct (primary hot gas duct) that separates the hot (core exit) and the cold (core inlet) gas flow streams. The hot gas duct is insulated to reduce heat losses to the core inlet cold gas stream, and the insulation assemblies are designed to be remotely removed and replaced if needed during the 60-year plant life. However, it is not clear whether the cross-vessels will be allowed to be defined as vessels rather than pipes.

2.1.5 Power Conversion and Intermediate Heat Exchanger

General Atomics recommends the use of a direct Brayton Cycle vertical PCS for electricity generation and an indirect heat transport loop to transport thermal energy to the hydrogen production plant. The primary loop and the hydrogen heat transport loop would both use helium at 7 MPa as a heat transport medium. The hydrogen heat transport loop would be sized to transmit up to 65 MWt. The assumed operating temperatures of the primary loop and hydrogen heat transport loop are 950°C and 925°C, respectively. The use of molten salt as a heat transport medium for the hydrogen heat transport loop was examined, and it was determined that molten salt is not yet ready to be deployed. There are issues of corrosion and materials compatibility, and concerns about the cost of the pipe materials, since it is believed that internal insulation could not be used to protect the heat transport pipes against the temperature.

A compact metallic heat exchanger module is assumed to serve as the intermediate heat exchanger for the hydrogen heat transport loop. Helical coil intermediate heat exchangers were also discussed, but General Atomics believes that compact heat exchangers will offer better performance and will be more economical.

The hydrogen heat transport loop are parallel hot and cold transport pipes that use internal and external pipe insulation in order to lower the temperature of the metallic pipe, so that metals less expensive than Inconel 617 can be used for the long-distance (i.e., 90 m) loop.

2.2 AREVA - ANTARES Concept

AREVA recommended that the NGNP be a 565MWt prismatic reactor, citing greater economic potential, higher power level and passive safety, more useable power, greater design flexibility, higher degree of license-ability (concept previously licensed for Fort St. Vrain), higher degree of predictability in core performance, forced outages and scheduled outages than a pebble bed reactor design.

The ANTARES design[2, 3] is also based in part on the GT-MHR concept, with 1020 fuel blocks arranged in three columns to form the annular core between inner and outer graphite reflectors, 36 operating control rods and 12 start-up control rods. The primary loop pressure is limited to 5.5 MPa which is substantially less than the 7 to 9 MPa specified by the other contractors. The design is shown in Figure 2.

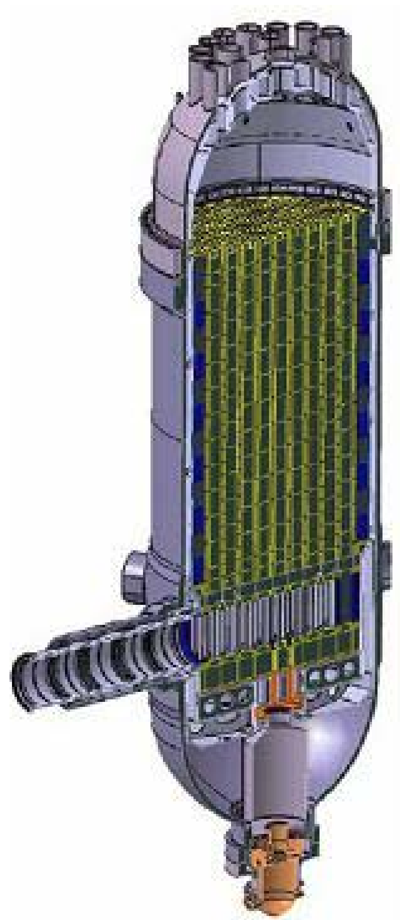


Figure 2. AREVA prismatic reactor.

2.2.1 Reactor Pressure Vessel

The RPV is approximately 25 meters high, 7.5 meters in diameter, and 150 mm thick and made from modified 9Cr-1Mo. This steel is preferred because of its superior high temperature properties compared to SA 508/533 LWR steel. The material selection will minimize risk and uncertainty in the design process and maximizes operational margin. This is a developmental material for this application. An ASME code extension is needed, but AREVA does not believe qualification for this application will not be substantially more difficult than qualification for LWR steel, and the resulting margin during a transient situation would be greater. Issues with availability, fabricability, through thickness properties and post-weld heat treatment need to be resolved.

2.2.2 Core Barrel

The core barrel consists of a double wall, metallic structure between the reactor pressure vessel and the outer side reflector. The core barrel material is 800H.

2.2.3 Reactivity Control Rods

The control rods will be enclosed in C_f/C canisters (cladding).

2.2.4 Cross Vessel

Like GA, AREVA has defined cross-vessels (rather than pipes), with a concentrically arranged primary hot gas duct that separates the hot (core exit) and the cold (core inlet) gas flow streams. The hot gas duct is insulated to reduce regenerative heat losses to the outer flow stream (core inlet cold gas).

2.2.5 Power Conversion and Intermediate Heat Exchanger

AREVA provided two plant configurations – a plant configuration that used a Brayton Cycle to generate electrical power, and a plant configuration that used steam to generate electricity by using a Rankine Cycle. The Brayton Cycle configuration is based upon the original ANTARES design. AREVA has recently concluded that the Rankine Cycle configuration may be more adaptable to NGNP requirements, and therefore the more favorable design. These configurations are briefly described below.

The Combined Brayton Cycle configuration employs an indirect power conversion unit and an indirect hydrogen heat transport loop. The secondary loop used to provide energy to the Brayton Cycle turbines uses a 20% He/80% N gas mixture at 5 MPa. Heat recovery steam generators would be deployed following the Brayton Cycle turbines to supply a bottoming Rankine Cycle. The hydrogen heat transport loop would be designed to carry up to 60 MWt to the hydrogen plant. Long term material behavior in the coolant environments requires specific qualification, including helium impurities and nitriding environment of the gas loop at high temperatures. AREVA recommends three primary to secondary heat transfer loops using metallic shell-and-tube heat exchangers to supply gas to the power conversion loop. They believe the use of shell and tube heat exchanger designs for the full power conditions is less developmental than use of compact heat exchanger designs to meet the schedule. AREVA does recommend use of a smaller (60 MWt) compact metallic heat exchanger in a fourth loop to supply the hydrogen process. The indirect power conversion loop and indirect hydrogen heat transport loop are linked to the primary cooling loop in parallel. This design shows a potential for very high efficiency, 47 % or more, while minimizing technological risk. In addition, the indirect combined-cycle is quite flexible to accommodate the simultaneous generation of electricity and industrial heat at any level between 800°C and ambient. This specific characteristic of the indirect-cycle HTR opens new markets to nuclear power.

The Rankine Cycle configuration differs from the Brayton Cycle configuration in that shell-and-tube steam generators are directly coupled to the primary cooling loop in order to drive the steam turbines. The steam generators are arranged in parallel with the hydrogen heat transport loop. The steam system would operate at a temperature of approximately 550ºC. No recommendation of steam system pressure was provided. The direct-cycle concept has little flexibility to adapt to cogeneration because when operating parameters change, efficiency drops and operation may become difficult. AREVA concluded that the Steam-Rankine cycle (possibly supercritical) is the best fit for near term applications because it provides high efficiency electricity production and can readily service near term process heat markets.

2.3 PBMR Concept

This reactor is being developed in South Africa by PBMR (Pty) Ltd. through a world-wide development effort. [4, 5, 6, 7, 8, 9] The program includes testing of mechanical systems and components, a comprehensive fuel development effort and a testing and verification program to support the licensing process. A full-sized demonstration PBMR reactor will be built at the Koeberg nuclear reactor site (owned by Eskom, the South African national utility) near Capetown, South Africa. Westinghouse recommended a pebble-bed reactor over a prismatic reactor design based on the fuel and fueling system demonstrated in Germany (AVR and THTR), minimal development costs and risks because of progress in

South Africa, higher capacity leading to higher performance capability, lower fuel temperatures, and a strong vendor/supplier infrastructure.

The PBMR utilizes 450,000 graphite-based spherical fuel elements, called pebbles, which are approximately 6 centimeters in diameter. The design of these pebbles, based on the German HTR, are located in an annular cavity in the reactor vessel, between a cylindrical inner graphite reflector and an annular outer graphite reflector. Pebbles proceed vertically downward until they are removed at the bottom. On removal they are checked, and if they are intact and not past the burnup limit, they are circulated to the input queue again. Otherwise, they are replaced with fresh pebbles. This on-line refueling feature makes refueling shutdowns unnecessary, and it also allows the reactor to operate with almost no excess reactivity, which confers advantages in safety, economy, and resistance to nuclear weapons proliferation. A schematic of the PBMR reactor is shown in Figure 3.

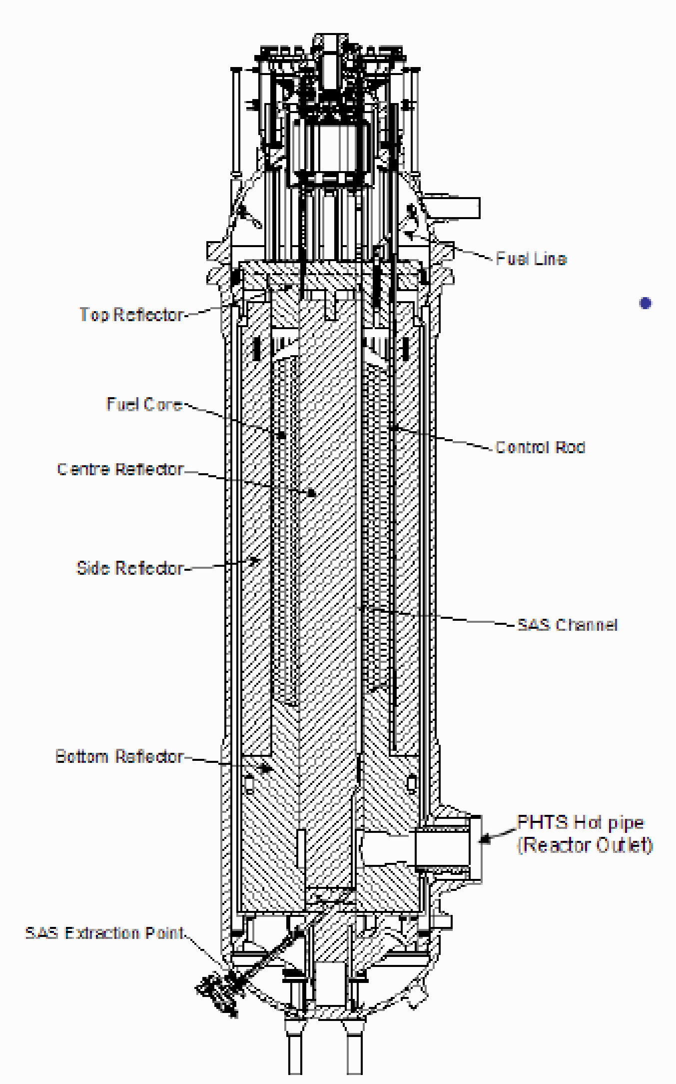


Figure 3. Schematic of the Pebble Bed Modular Reactor (PBMR) annular pebble bed reactor.

The building design for a single PBMR module consists of a reinforced concrete confinement structure, called the citadel, which houses the power conversion unit, which is located inside a more conventional concrete building that houses all of the auxiliary equipment. The function of the citadel is as a confinement structure to protect the nuclear components of the power conversion unit from external missiles and to retain the vast majority of fission products that might be released in the event of a reactor accident. The limited total core power allows the reactor to be designed for passive heat conduction from the core, thermal radiation and convection from the vessel and conduction to the confinement structure keeping temperatures low enough to prevent core or fuel damage.

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

2.3.1 Reactor Pressure Vessel

The vessel design consists of a welded cylindrical shell welded to the bottom head. The top head, containing numerous penetrations for fuel handling and reactor control systems, will be bolted to the cylindrical section. The dimensions are 6.8 m in diameter x 30 m high.

The RPV design configuration is such that its normal operating temperature range is from 300-350°C. The selection is SA508 Grade 3 Class 1 (forgings), SA533 Type B Class 1 (plates) and SA504 Grade 24B Class 3 (bolts) steels. A separate stream of helium actively cools the RPV, however during postulated severe accident conditions calculations indicate that the proposed RPV material temperature may be in the creep range. However these LWR pressure vessel steels provide the following benefits:

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

2. There is welding experience with these materials

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

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

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

The IHX vessels, made from the same steel as the RPV, connect the primary system to the secondary heat transport loop, and contain the heat exchangers.

2.3.2 Core Barrel

The core barrel is a welded internal metallic vessel that supports the graphitic reflector and the fuel spheres. The core barrel is directly supported by the RPV. It further separates the gas flow through the core from the gas flow on the inside of the RPV. The core barrel material is constructed from SA240 Type 316H plate material. The H-grade is specified to provide enhanced creep resistance when the core barrel is exposed to temperatures above 427°C. There is also extensive industrial experience and inclusion in ASME Code Case N-201 for core support structures for Type 316H material. The core barrel is shown in Figure 4.

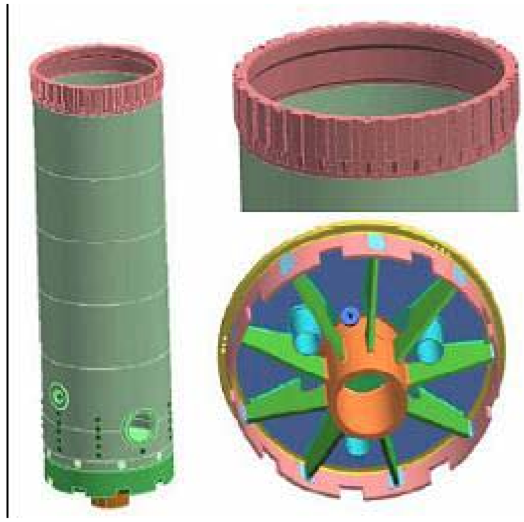


Figure 4. Core Barrel & Support Structure.

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

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

1. Extensive qualification and industrial use.

2. Significant resistance to irradiation effects due to fast neutron fluence ($E > 1$ MeV).

3. Relatively cost-effective in comparison to the other available candidate materials.

2.3.3 Reactivity Control Rods

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

There are 24 control rods which are located in the side graphite reflector blocks. Half of the rods are used for control and the other half are used for shutdown. The shutdown rods are longer, running the length of the reflector blocks, while the control rods only run in the upper half of the reflector blocks. The rods consist of a B_4C rings between two coaxial cladding tubes. Although the clad material was not specified in the pre-conceptual design report, previous documents list Incoloy 800H as the most suitable material for the control rods for the following reasons:

1. Adequate high-temperature strength at the normal operating temperature of 700°C
2. Creep resistance sufficiently qualified for long-term operation at 700°C.
3. Limited operation at 850°C under abnormal events is allowed as per available data.
4. Irradiation response has been characterized to high levels of fast fluence.
5. Extensive qualification of Incoloy 800H control rods in previous German HTR programs.

2.3.4 Core Outlet Pipe Liner

The arrangement of the RPV core outlet pipe liner is shown in Figure 5. The liner forms an integral part of the insulation held between the liner and the outer pressure boundary material. The insulation in the core outlet piping is a necessary component of the insulation system required for keeping the outer pressure boundary (ferritic steel) temperatures within operational limits.

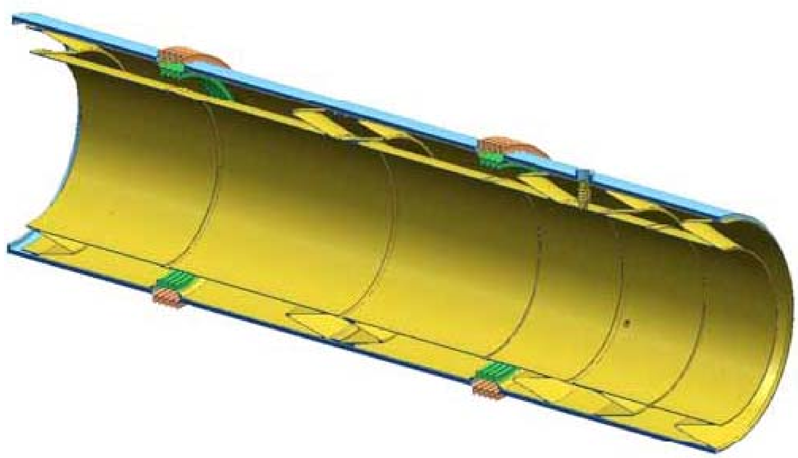


Figure 5. Core Outlet Pipe Liner.

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

1. Adequate high-temperature strength and creep resistance
2. Extensive fabrication experience in large diameter pipe sections

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

2.3.5 Power Conversion and Intermediate Heat Exchanger

Westinghouse recommends the use of an indirect power conversion cycle and an indirect hydrogen heat transport loop arranged in a serial fashion. The intermediate heat exchanger for the hydrogen heat transport loop would be placed first in the series in order to obtain the highest temperature gas from the nuclear reactor. Section 1 (IHX A), made of metal or ceramic, is expected to operate at temperatures between 710 and 900°C and is expected to be replaceable. Section 2 (IHX B), made of metal, would be expected to operate at temperatures below 710°C and would be designed for a 60-year lifetime. Both intermediate heat exchanger sections are envisioned as compact heat exchanger modules, as it is believed that tubular heat exchangers would be too large and costly to be economical. The pressure of the primary loop was assumed to be 9 MPa, and the secondary loop was assumed to operate at a pressure between 8.1 and 8.5 MPa. The power conversion cycle uses steam generators and a traditional Rankine Cycle to generate electricity, and would be designed to receive the full power of the reactor.

3. Background

A number of reports have been produced that provide useful background for the NGNP materials program and provide direction for research planning. The first section contains background information on nuclear grade graphite. Three lengthy reports were prepared in FY2006 outlining the issues relevant to material selection for the RPV, IHX and control rods. These reports are summarized in sections 3.2-3.4 respectively. In addition, reports detailing the general issues of environmental effects, standards and codes, and high temperature design methodology are summarized in sections 3.5 - 3.6. Finally, a report summarizing a review of the NGNP materials program performed by NERAC (Nuclear Energy Research Advisory Committee) and an analysis of the materials related risk issues performed by the ITRG (Independent Technical Review Group) are presented.

3.1 Nuclear Grade Graphite

Nuclear grade graphite is a specially developed composite material 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. Figure 6 shows the major processing steps in the manufacturing of nuclear graphite. 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 with the entire process taking 6-9 months.

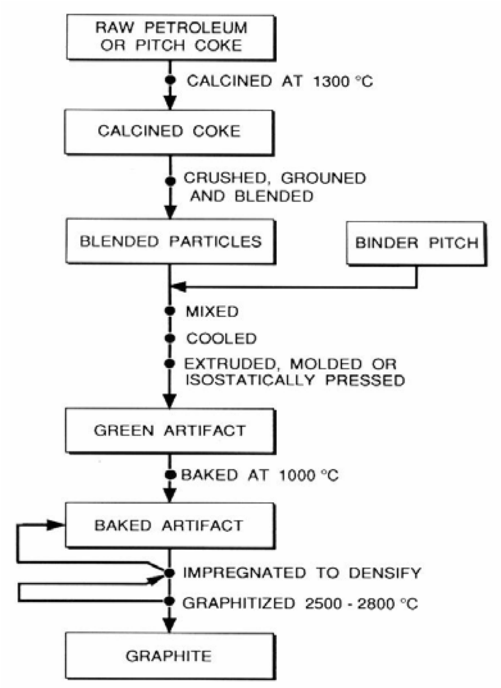


Figure 6. Typical process steps in the manufacturing of nuclear graphite.

Nuclear grade graphite has been specially developed to meet reactor design requirements. Attributes required for modern nuclear grade graphite are:

- Dimensional change - Near isotropic graphite
- High purity - Low elemental contamination especially boron
- Fabrication – Ability to machine large graphite components
- Irradiation - Must possess irradiation design database

While these are minimum attributes necessary to achieve acceptable component lifetimes necessary for use within an irradiation environment, they may not be sufficient to demonstrate adequate structural integrity for all design configurations. It is known that individual "nuclear grade" graphites will have distinctly different responses to the irradiated environments based upon the extent of anisotropy, grain size, microstructural defects, microstructure orientation, purity, and fabrication method. As an example, for reasons not fully understood, orthotropic Magnox reactor graphite components show no evidence for cracking, whereas isotropic AGR graphite components show extensive cracking. Thus, the response of each graphite type must be verified for use as a structural component within the NGNP.

The nuclear graphite previously used in the USA for HTR applications is no longer available (H-451). New types have been developed and are currently being considered as candidates for the NGNP but a qualified properties database on these new candidate grades of graphite must be developed to support the design of graphite core components within the specific reactor service conditions of the NGNP. Non-irradiated and irradiated data are required for the physical, mechanical (including radiation induced creep) and oxidation properties of the new graphite. To meet these requirements a radiation effects database must be developed for the currently available graphite materials. Much more detailed information on graphite fabrication, properties, and the acquisition of bulk material is discussed within the "NGNP Graphite Selection and Acquisition Strategy" report, ORNL/TM-2007/153.

Component lifetime calculations using new graphite types will be determined from both the initial non-irradiated, "as-received" material properties and the property changes that will occur due to radiation damage or environmental degradation to the graphite during operation. The non-irradiated mechanical and material property values will be used as baseline data for initial reactor startup and operation. The "as-received" property values of the graphite components will be used to calculate the initial core thermal properties (conductivity, specific heat, etc.), and physical response (applied stresses, dimensional tolerances, etc.).

The evolution of these property changes is dependent upon a number of factors including temperature, fluence/dose, graphite microstructure/orientation, chemical purity, and applied stresses during operation. Obviously, those components located physically closer to the fueled region of the core will experience higher temperatures and doses than components on the edge of the reactor and a faster rate of change is expected. The extent of property changes include physical changes to the component (i.e. dimensional changes), changes in the thermomechanical properties especially irradiation-induced creep, and changes to thermophysical properties such as thermal conductivity, coefficient of thermal expansion etc. All of these will affect the prediction of graphite lifetime.

3.2 Reactor Pressure Vessel

A report entitled "Preliminary Materials Selection issues for the next Generation Nuclear Plant Reactor Pressure Vessel" (Natesan et al., September 2006)[10] reviewed the available information on candidate materials for the construction of the RPV and made a preliminary assessment of several factors relevant to judicious material selection. Some of the factors addressed in this report were, baseline mechanical properties, including the effects of thermal aging, impure helium and thick sections; availability of commercial alloys, with a global assessment of potential RPV vendors; welding issues; and ASME code compliance, including the status of candidate alloys in the ASME Codes for nuclear service and an evaluation of suitability during steady state and depressurized conduction cool-down (DCC) conditions. Three primary candidate alloys were considered for the RPV: low alloy steel SA508 (UNS K12042), Fe-2.25Cr-1Mo-0.25V steel (UNS K31835), and modified 9Cr-1Mo steel (UNS K90901). Technology gaps in our knowledge base were highlighted.

3.2.1 Baseline Mechanical Properties

There is a sufficient database available for the mechanical properties of SA-508 steel. The data available on the thermal aging effects on the mechanical properties is promising but additional information is needed on the long-term aging effects. At present no data is available on the effects of impure helium on the long-term corrosion and mechanical properties of the material.

There is adequate tensile data on the Fe-2.25Cr-1Mo-0.25V in the temperature range of interest in NGNP RPV. There is only limited creep data available for the steel and additional data are needed, especially at elevated temperatures that encompass depressurized conduction cooldown conditions. Further investigation is required into the thermal aging affects, which are of particular concern because of the risk of temper embrittlement in this alloy caused by segregation of impurities to the grain boundaries. In addition, substantial data on the creep fatigue properties, performance characteristics in impure helium, and properties of thick section material are needed prior to selecting Fe-2.25Cr-1Mo-0.25V the for NGNP RPV.

A substantial database on the baseline mechanical properties is currently available for the modified 9Cr-1Mo steel. Sufficient data are also available on the long-term thermal aging effects on the mechanical properties for this steel. However, additional data are needed for the mechanical properties of thick sections, where there is the possibility of retained ferrite in this martensitic steel which can lead to embrittlement. As with the other alloys under consideration, properties in impure helium must also be explored.

3.2.2 Procurement Issues

The current schedule for the NGNP plant requires that the conceptual and preliminary designs and the application for the construction permit from NRC be completed by the middle of calendar year 2010.

The selection of material for the NGNP RPV is one of the critical items to meet the schedule. In order to fabricate the huge RPV, vendors are needed who can produce seamless rings (forged) or plates (forged or rolled), achieving uniform through-thickness properties with the candidate materials.

At present, several vendors around the world have substantial experience in fabrication of RPV's from SA-508. Procurement of a vessel of this material may depend primarily on the availability of a vendor to meet the schedule and not on the technical issues with the material.

Fe-2.25Cr-1Mo-V steel is extensively used in the fossil industries and hydrogenation reactor pressure vessels. There is no available information on existing pressure vessels having dimensions similar to that required in the NGNP RPV; however because of the large scale use of this material in fossil, petrochemical industries, etc. it can be concluded that there is some experience in forging/rolling thick-sections of this material.

Modified 9Cr-1Mo steel has overall superior mechanical properties among the three candidate materials that would enable manufacture of an RPV with thinner walls, thereby reducing thermally induced stresses and minimizing eventual thermal fatigue and making it a primary candidate for use in the NGNP RPV. However, information is lacking on thick-section properties and fabrication experience of this material. An assessment of the potential vendors from all over the world showed that capability and experience to fabricate a modified 9Cr-1Mo vessel of the size required for NGNP are severely lacking (Table 2). None of the vendors have the capability, at present, to forge thick-section large diameter rings of modified 9Cr-1Mo steel. It was clear that none of the vendors was willing to upgrade their existing facility to facilitate forging of this steel unless an incentive is offered to them (in terms of assured market/customers to order RPV of the modified steel, or in some other form). Ring forging of RPV using modified 9Cr-1Mo steel does not appear to be a feasible option at present; therefore axial welding of plates/ring segments is the alternate choice. However, none of the vendors has experience in manufacturing thick-section plates either, although Saarschmeide of Germany is confident they can (about 55 plates would be required to construct the RPV). This procurement limitation may preclude meeting the NGNP schedule and eliminate this material as a candidate for the RPV.

3.2.3 Welding Issues

The RPV will be much larger than the current Light Water Reactor vessels and requires field welding of either ring forgings or plates of the selected material. While ring forgings are preferred since they would result in fewer welds (no longitudinal welds) to assemble the RPV, this may not be possible. Depending on the material selected, welding procedures may include pre- and post-weld heat treatment in the field.

Pressure vessels of low alloy steels have been fabricated and used in U.S. LWRs and there is substantial experience in welding of both plates and rings to form the vessels. Vessels with wall thicknesses varying between 203 to 254 mm (8 to 10 in.) and diameter-to-thickness ratios of ≈20 have been fabricated for the Pressurized Water Reactors (PWR). In contrast, Boiling Water Reactor (BWR) vessels with much larger diameter and a wall thickness of 152-mm (6-in.) have been fabricated. Pressure vessel materials have performed well in service, and no fatigue-driven cracks have been found in any PWR vessels. The only materials-related variable that appears to affect fatigue is the sulfur content (and distribution as sulfides) in the steel. Low alloy steels with average to high sulfur levels generally exhibit higher crack growth rates in the laboratory than low sulfur steels (<0.010 wt.%).

2.25Cr-1Mo steels were developed for the petrochemical refinery industry service at high temperatures and high hydrogen pressures. A vanadium-modified version of these steels was developed so that components having wall thickness in excess of 10 in., and diameters and lengths on the order of up to 20 and 200 ft, respectively, can be fabricated. These steels offer the fabricability and toughness of bainitic microstructures without the difficulties of welding and heat treatment of high chromium martensitic materials like modified 9Cr-1Mo steel. Testing has shown that filler metal, base metal, and HAZ were comparable in strength and within ASME's base metal scatter band. However, for design temperatures in excess of 468°C (875°F), ASME requires performance testing of weldments, so additional long-term test data are needed to qualify the welded components for application in the NGNP RPV. The susceptibility of Fe-2.25Cr-1Mo-0.25V to temper embrittlement is also concern when welding in addition to the aging effects mentioned above. If the grains are insufficiently tempered, the effect is the greatest. Test results for base metal, HAZ, and weld metal exposed for 20000 h at 482°C (900°F) have shown that acceptable toughness is retained for the materials, provided impurity levels were maintained within specific limits.

Modified 9Cr-1Mo steel is a leading candidate for NGNP RPV; however the superior mechanical properties of the 9Cr-1Mo weldment strongly depend on creation of a precise microstructure and maintaining it throughout the service life of the welded component. Welding procedure and post-weld heat treatment play critical roles in creating the desired microstructure and producing a stress-free weld. The most significant problem with welding of modified 9Cr-1Mo steel is its propensity to Type IV cracking in the heat affected zone. Material which has exceeded the minimum transformation temperature during the welding process can partially reaustenitize and coarsen, resulting in substantially reduced creep-rupture strength and leading to cracking at relatively low operating temperatures and early component lifetimes. Boron addition seems to reduce cracking susceptibility but additional data are needed to quantify the effect over the long term. Over-tempering, under-tempering, cold-work, dissimilar metal welds and stress corrosion cracking are also potential problems encountered in modified 9Cr-1Mo weldments. Welding this steel requires more care in fabrication procedure and joint design than lower alloy steels being sensitive to temperature variations both during welding and post-weld heat treatment.

Creep-fatigue interaction could be more severe in modified 9Cr-1Mo weldments compared to 2.25Cr-1Mo weldments. The creep-fatigue data show that the number of cycles to failure decreases with the introduction of hold time, and the effect is more sever for the weldment than for the base metal. Significant additional data are needed to quantify this effect and establish the maximum reduction in life, if any.

3.2.4 RPV Requirements

The NGNP RPV will have dimensions of 8-9 m in diameter and probably 200-300 mm in thickness. The maximum operating temperature of the RPV will depend on the NGNP design (pebble bed or prismatic block), the outlet gas temperature and power level selected. The PBMR design in South Africa has calculated an RPV nominal operating range of 260-300°C by utilizing an independent cooling stream. The prismatic GT-MHR design estimated a vessel operating temperature of 495°C. Analysis by Gougar and Davis (2006)[11] using RELAP5 calculated temperatures within these extremes: 360°C for the pebble bed design and 427°C for the prismatic design. The radiation exposure expected for the NGNP RPV is very low (<0.1dpa) and the effect of irradiation on mechanical properties is expected to be negligible.

3.2.5 ASME Code Compliance

The NGNP RPV needs to be designed using the ASME Section III Code rules. If the RPV wall temperature can be maintained at a sufficiently low temperature (≤ 371°C=700°F) with only limited excursions as defined under Code Case N 499, Subsection NB of the Code can be used. Otherwise, creep strains are not negligible and Subsection NH must be applied. Figure 7 shows the allowable primary membrane stress intensities vs. temperature for each material and the maximum calculated primary stress intensity/maximum wall temperature data points for the two RPV designs. However, the maximum design lifetime data provided in Subsection NH is ≈ 34 yrs (300,000 h) for the steel, which is less than the NGNP design lifetime of 60 yrs.

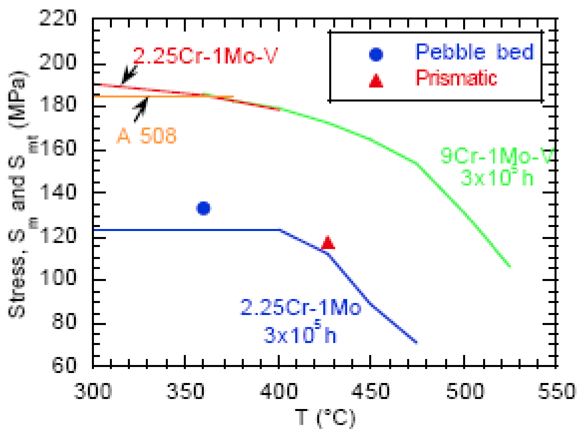


Figure 7. Variation of primary membrane stress intensity and allowable primary membrane stress intensities as functions of temperature and time.

SA508/SA533 steels are ASME Code approved for Class 1 nuclear components and Subsection NB rules are applicable. SA-508 forging can be a potential candidate for the pebble bed RPV design since the peak temperature as calculated by RELAP5 is ≤371°C; however, temperatures >371°C were calculated when a pebble-bed-specific code was used. SA-508 steel is unacceptable for the prismatic core RPV design because the calculated temperatures during normal operation exceed 371°C. Active cooling will be mandatory, if this steel is selected for the prismatic, and perhaps even for the pebble bed RPV design.

Figure 7 indicates that while Fe-2.25Cr-1Mo (without V) does not have adequate strength for either NGNP RPV design, Fe-2.25Cr-1Mo-V has sufficient strength up to 400°C and limited creep-rupture data in the literature indicates it may have adequate high temperature strength for either design. However, at present, Fe-2.25Cr-1Mo-V steel is approved under ASME Code Section VIII (non-nuclear applications) but not approved under Section III for nuclear service.

Modified 9Cr-1Mo steel is approved in Section III of the ASME Code for nuclear applications; however, the creep-fatigue limits for the steel in the code is highly conservative and it may preclude its selection for the NGNP RPV application. Calculations performed for the modified 9Cr-1Mo steel showed that the peak membrane stress for the pebble bed design RPV is within the ASME Code Subsection NB allowable for the steel (Figure 7). The peak membrane stress for the prismatic design RPV is within the ASME Code Subsection NH allowable. Stress analysis of the depressurized conduction cooldown condition for both pebble bed and prismatic designs showed the peak temperatures to be in the creep range for the steel, but the stresses are too low to cause any significant creep deformation ($<10^{-6}$).

Table 2. Forging capability of modified 9Cr-1Mo for NGNP RPV (≈8 m dia. x 24 m high with thickness of 100-300 mm)

Manufacturer	Current Ring Forging Capability	Future/Upgrade Plans	*Viability to forge Modified 9Cr-1Mo*
Japan Steel Works, Japan	8 m OD	May be inclined to try 2.25Cr-Mo steel but not modified 9Cr-1Mo steel	*RINGS - NO* *PLATES - NO*
Bruck Forgings, Germany	5.2 m OD (max)	8m rings in 2-3 years*	*RINGS - NO* *PLATES - NO*
Saarschmeide, Germany	< 5 m	Probable investment in large forging press by 2009*	*RINGS - NO* *PLATES - YES***
Scot Forge, IL	6 m OD (max)	None	*RINGS - NO* *PLATES - NO*
Doosan Heavy Industries (DHI), South Korea	*Experience with modified 9Cr-1Mo for non-nuclear applications.*	*KAERI*** in talks with DHI to fabricate thick section vessel using modified 9Cr-1Mo.*	*?*

*Not necessarily for modified 9Cr-1Mo

**≈ 6 m x 2.5 m plates, but no experience in manufacturing modified 9Cr-1Mo or 2.25Cr-1Mo-V.

***Korean Atomic Energy Research Institute (KAERI) is interested in investing/funding DHI for this project.

3.3 Intermediate Heat Exchanger

In the indirect cycle system, an IHX is used to transfer the heat from primary helium from the core to the secondary fluid, which can be helium, a nitrogen/helium mixture, or a molten salt. A recent report (K. Natesan, A. Moisseytsev, S. Majumdar, and P. S. Shankar, 2006)[12] made a preliminary assessment of the issues pertaining to IHX for the NGNP. Two IHX designs: namely, shell-and-tube and compact printed circuit heat exchangers (PCHE) were considered. Since the publication of this report it has been concluded that plate-fin and printed circuit style compact heat exchangers have potential size, weight and efficiency advantages over more traditional shell and tube style heat exchangers for application as the intermediate heat exchanger for HTGR.

3.3.1 Materials under consideration

The material considerations for the two heat exchanger designs are essentially similar, with the exception of some fabrication issues. The majority of materials research and development programs in support of high temperature gas reactors (HTGRs) were conducted in the 1960s to early 1980s. The thrust of these programs was to develop a database on materials for application in steam-cycle and process-nuclear-heat based HTGRs. Less work has been done on materials with emphasis on direct and/or indirect gas-turbine-based HTGRs. The available material property data was reviewed in detail and an assessment of relevant factors was made including tensile, creep, fatigue, creep-fatigue, and toughness properties for the candidate alloys, thermal aging effects on the mechanical properties, ASME Code compliance information, and performance of the alloys in helium containing a wide range of impurity concentrations. The assessment included four primary candidate alloys for the IHX: Alloy 617 (UNS N06617), Alloy 230 (UNS N06230), Alloy 800H (UNS N08810), and Alloy X (UNS N06002). A brief summary of the properties, an analysis of the status of the materials database, and the ASME code status for each alloy follows.

3.3.1.1 Alloy 617 (52Ni-22Cr-13Co-9Mo)

This alloy is the primary candidate for construction of the NGNP IHX, and the existing mechanical property database is extensive. It has exceptional creep strength at temperatures above 870°C, good cyclic oxidation and carburization resistance, good weldability and lower thermal expansion than most austenitic stainless steels. It retains toughness after long-time exposure at elevated temperatures and does not form embrittling phases. Preliminary testing indicates Alloy 617 has the best carburization resistance of the four alloys. Composition refinement is necessary to achieve consistent mechanical properties and for precise evaluation of microstructural effects on mechanical performance based on variation in properties for different heats of the alloy.

Alloy 617 is not currently allowed in ASME Code Section III, although it is allowed in Section VIII, Division 1 (non-nuclear service). A draft code case for Alloy 617 has been developed but is not approved by the ASME Code. Significant R&D will be needed to qualify Alloy 617 under ASME Code Section III for high temperature applications, even though some design-relevant material properties are given in the draft code case for temperatures up to 982°C. The ratcheting rules, that were limited to <650°C in the draft code case, need to be expanded to allow higher temperatures. Additional mechanical property data is needed on creep-fatigue behavior with and without hold time. The analysis showed that the damage under creep-fatigue conditions is much more than predictions based on the linear rule; and additional data are needed to develop a predictive capability on creep-fatigue damage in the alloy, especially in helium purity levels typical of gas-turbine-based HTGRs. There is also a need to establish the corrosion regimes in helium with impurity levels anticipated in a gas-turbine based reactor system; and, it is necessary to validate the effect of system pressure on the corrosion performance of the alloy in impure helium. Microstructural and mechanical property characterizations are needed for thin section materials, especially for use in PCHEs. Tests are needed to verify that the tensile, creep, and creep-fatigue strengths of the diffusion bonded Alloy 617 joints used in the PCHE are adequate.

3.3.1.2 Alloy 230 (57Ni-22Cr-14W-2Mo-La)

Alloy 230 has outstanding resistance to oxidizing environments, good weldability and fabricability. It has a lower thermal expansion coefficient than Alloy 617. Alloy 230 has higher tensile strength than Alloy 617 up to 800°C, but above that the difference is insignificant. It appears Alloy 617 has slightly better creep properties than Alloy 230, although both exceed the ASME requirement. Alloy 230 has better thermal fatigue crack initiation resistance but worse thermal cycling resistance.

Like Alloy 617, Alloy 230 is not currently allowed in ASME Code Section III, although it is allowed in Section VIII, Division 1 (non-nuclear service). At present, there is insufficient database for Alloy 230 to develop a code case for elevated temperature application. Limited data are currently available on the mechanical properties for the alloy. Data on long term aging effects on the mechanical properties need to be generated. There is lack of information on the long-term corrosion performance of the alloy in helium of relevant impurity levels. The effect of system pressure on the corrosion performance of the alloy in impure helium needs evaluation. Microstructural and mechanical property characterizations are needed for thin section materials.

3.3.1.3 Hastelloy X (47Ni-22Cr-9Mo-18Fe)

Above 700°C Hastelloy X can form embrittling phases that result in property degradation. Hastelloy X has the best oxidation resistance of the four alloys, although its carburization resistance is the worst. The creep rupture strength is not as good as Alloy 617 or 230.

A limited database exists for Hastelloy X, but the high temperature scaling in Hastelloy X has not been acceptable. As a result, a modified version, Hastelloy XR, has been developed in Japan; however, the U.S. has little access to XR either for evaluation or for ASME Code qualification.

3.3.1.4 800H (42Fe-33Ni-21Cr)

This alloy is the only iron based alloy under consideration, although it has a solid-solution strengthened austenitic structure like the other three alloys. However, upon aging precipitates can form and reduce the tensile and creep ductility. 800H has the lowest creep rupture strength and the worst resistance to oxidation of the four alloys.

Among the four candidate materials, alloy 800H is the only one code certified for use in nuclear systems, but for temperatures only up to 760°C and therefore the temperature limit is not high enough for application of this alloy in NGNP IHX. Only very limited data are currently available on the mechanical properties of this alloy beyond 800°C, especially in impure helium environments.

Figures 8, 9, 10, 11 compare the creep rupture strength, oxidation behavior, carburization behavior, and allowable stress for the four alloys, respectively.

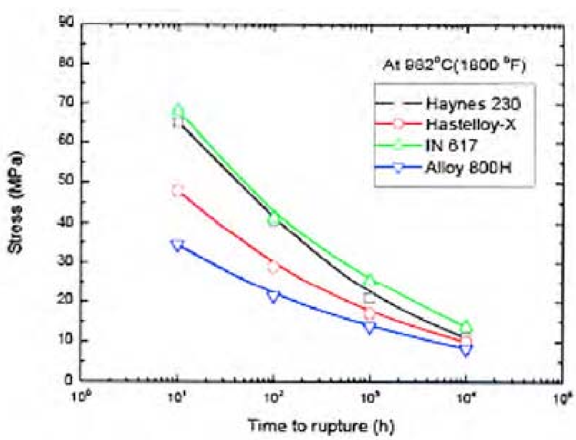


Figure 8. Creep rupture strength at 962°C in air.

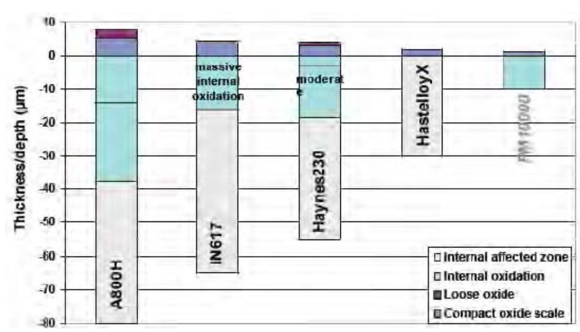


Figure 9. Isothermal oxidation behavior after 800 h exposure at 950°C in helium environment.

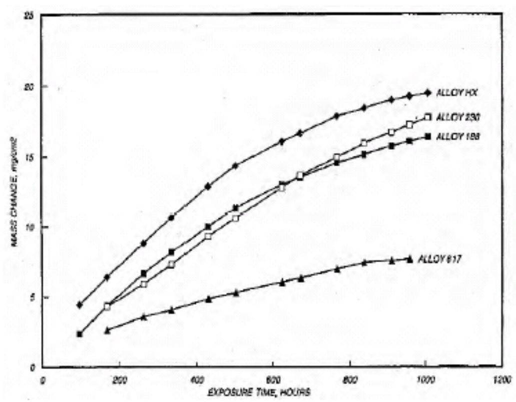


Figure 10. Mass change as a function of time in H2-5.5%CH4-4.5%CO2 carburizing environment at 1000°C.

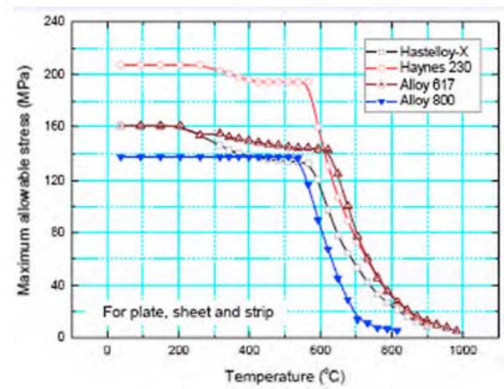


Figure 11. Allowable stress for heat exchanger materials for plate, sheet and strip forms.

3.3.2 Thermal hydraulic analysis

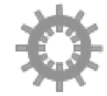
In addition to analyzing the materials under consideration, the ANL report performed a thermal hydraulic analysis for helium-to-helium, helium-to-helium/nitrogen, and helium-to-salt HXs. The model was used to determine the sensitivity of numerous HX parameters as well as the required size of the HX. Additional thermal conduction and stress analyses were used to perform ASME code compliance calculations and estimate maximum allowable design lives. Since the IHX is a high temperature component where creep is important, the draft code case for Alloy 617 patterned on ASME Code Section III, Subsection NH was used, assuming that the IHX will be classified as a class 1 component. From these analyses, the following were concluded.

- If the IHX is classified as a Class 2 or Class 3 component, new code cases need to be developed for high temperature applications.

- Design rules of Subsection NH of the ASME Code, which were developed for shell like structures where one dimension (thickness) is much smaller than the other two (such as shell-and-tube HX), are not strictly applicable to the compact three dimensional honeycomb structures of PCHE. Thin sections and diffusion bonding must also be considered. New design rules and analysis tools are needed for this type of structure.

- Calculations were made to compare the performance of a PCHE with a shell-and-tube heat exchanger. The results (Table 3) showed large reductions in PCHE volume compared to a shell-and-tube HX, especially without ID and OD fins. Also, the PCHE had substantial increases in the pressure drop for the hot side and even larger pressure drop increases for the cold side compared to the shell-and-tube HX. The results also showed that for an IHX heat duty of ≈45 MWt, ≈20 units of PCHE would be required based on the fabrication size limit used in the calculations.

Table 3. Comparison of PCHE and shell-and-tube IHX designs.

Heat Exchanger Type	PCHE	Shell-and-Tube 	Shell-and-Tube 
Unit dimensions	0.6 m (L) x 1.5 m (W) x 0.6 m (H)	6 m (L) x 8.17 m (D)	6 m (L) x 3.31 m (D)
Number of units	19.408	1	1
Total HX volume, m^3	10.5	314.5	51.6
Pressure drop, hot/cold sides, kPa	25.6 43.7	0.05 0.05	3.2 3.7

• For a reactor outlet temperature and pressure of 900°C and 7 MPa, respectively, and a secondary side inlet temperature and pressure of 575°C and 1.95 MPa, respectively, the allowable design life (based on in-air tensile and creep rupture strengths of Alloy 617) of the PCHE is 80,000 h and that of the shell and tube IHX (for the tube size selected) is 20,000 h. These design lifetimes are based on analyses of the IHX core and further reduction in life may result from interaction of the core region with the header region. The design lifetimes may also decrease, if thin section creep properties (which are generally less than those of thicker material) are considered. In addition, the effect of impure helium on creep properties needs to be incorporated in the design data and in the lifetime calculations.

• Replacing Alloy 617 with Alloy 230 will not lead to longer design lifetime for the IHX unless the performance of Alloy 230 in NGNP helium is significantly superior to that of Alloy 617.

3.4 Control Rods

A report (Wright and Lloyd, 2006)[13] studied the potential materials that may be selected for the control rod sleeves and the issues associated with the selection of each material including how these issues relate to the reactor designs. The report examined past reactor designs and control rod performance, as well as the two reactor design concepts under consideration for NGNP in detail as they pertain to control rod design and performance. In addition to evaluating the material requirements, the report reviewed the irradiation behavior, testing, codification, cost, availability, and manufacturing issues of three materials: the two composites currently under consideration and alloy 800H.

3.4.1 Design requirements

Composite materials (carbon fiber/carbon [C_f/C] and silicon carbide fiber/silicon carbide [SiC_f/SiC]) have been considered the only viable choice for control rod sleeves based on the assumption that the control rods would see a temperature (1250°C) and irradiation environment (30 dpa lifetime dose) similar to the fuel in the active core. In actuality, the control rod requirements are much less demanding than previously thought. In part, this is because the outlet temperature has been lowered from 1000°C to between 850 and 950°C, but primarily because of a misunderstanding that the shutdown rods in the prismatic design would be inserted in the core during normal operation and off-normal events rather than suspended above the core. Also, liners were mistakenly thought to be required inside the control rod channels to keep graphite reflector blocks aligned. As a result, temperature requirements are about half of the original requirement, and radiation dose requirements are less than one-third of the original assumptions, which were based on modeling results for the control rods residing in the inner reflectors. The requirements are fairly similar for the pre-conceptual prismatic and Pebble Bed Reactor (PBR) designs. Although the PBR has a slightly lower fluence value, it would not impact material selection. However the prismatic design has potentially higher operating temperatures under certain conditions and also a higher maximum DCC temperature, which could be significant in discriminating between control rod sheathing materials. Table 4 summarizes the current estimated operating conditions.

Table 4. Current estimated operating conditions for NGNP control rods

	Normal temp	Max DCC temp	Lifetime dose
Prismatic	732°C	1153°C	8.4 dpa
Pebble bed	546°C	1017°C	7.9 dpa

Based on the reduced requirements for materials, the study determined Alloy 800H is certainly worthy of consideration. The reactivity control rods for both the prismatic and pebble bed design could arguably operate under the ASME code for 800H, as it currently stands, with a maximum use temperature of 760°C. Although (like all austenitic alloys) 800H experiences radiation embrittlement, the alloy is being used in high-temperature gas reactors (HTGRs) and the embrittlement can be accounted for in the component design. The primary problem with using 800H control rods is planning for high temperature off-normal events such as DCC. Either procedural solutions such as delayed scram or temperature testing will be required. The testing would determine if the part would deform to the point it could not be extracted from the outer reflector after exposure at 1150°C for 1500 hours. 800H will also require additional testing, including irradiation testing to fill out the database, and perhaps elevated temperature testing to extend the use temperature specified in the ASME code. However, these needs appear to be less than what will be required to codify composite materials.

3.4.2 Codification and licensing

Any new material or engineered material system to be used in the NGNP reactor design will require acceptance by the NRC prior to implementation in the NGNP. The necessary activities would include development of ASTM International testing standards, ASME performance specifications, ASME design rules, and ASME inspection requirements. These items would provide acceptable evidence of the material's suitability for the application to the NRC. Such an application may involve many years of development activities before submittal of a topical report to NRC for consideration.

The composite materials are non-homogeneous and anisotropic, and therefore require unique test methods to measure their properties. Most test methods that exist for these material types restricted to room temperature evaluation. Tests at the temperatures of use are necessary for performance assessment. Standardized test methods to measure all properties of interest of new ceramic composite systems at high temperatures, do not exist or are still in development. Test methods to evaluate these composite materials at elevated temperature, including irradiation effects, will need to be developed and formalized using ASTM International standards development procedures. This can be a lengthy process, depending on the complexity of the test method proposed, and the difficulty in obtaining the trial test data. Such a trial is usually provided by at least six interested laboratories on a volunteer basis, which is necessary in assessing the viability and reliability of the method. This would be the first step in the NRC acceptance process.

Following development of standardized test methods, the material(s) of interest would have to go through the ASME codification process to achieve a standard performance specification. This step also has the potential to be a lengthy process. There are no ASME specifications for composite materials to be used in nuclear applications. Entirely different design and performance philosophies must be adopted for composite use. The types of properties or performance criteria to be measured must be determined, then appropriate methods must be devised to make the measurements in a repeatable way. A subcommittee of interested parties within an appropriate ASTM Section/Division will typically develop a draft standard; It is important that there are enough members interested in developing such a standard and to have the subcommittee actively pursue its development. Once approved at various committee levels, a round-robin test program will typically be organized to assess the standard. If analysis of results obtained in the round-robin test proves acceptability of the standard, it will be presented to the membership for a vote. If approved by general vote, it will be incorporated into the following year's edition of the ASTM Standards.

Existing ASME design and assessment rationale within Section III are guided by homogeneous material characteristics. The system of design and assessment, as well as specification, will have to be revised to account for the differing performance modalities presented by fiber composite systems. It can be argued that test specimens must be cut from prototype components subjected to appropriate thermal and irradiation conditions, to test actual structures. This may provide useful design information for the particular component of interest but very little detail for comparative purposes, or a different material system. This also creates substantial costs, both time and fiscal, as a test method and procedure developed according to this approach may only be suited for a narrow range of articles and material thickness. Furthermore, not only must the individual constituents of the final material be qualified and approved, but also the method of assembly.

The federal government has mandated cooperative development of new licensing-strategy documents that utilize modern approaches to reactor design and operational risk assessment; and the ASME Board is actively working with NRC towards development of a probabilistic risk assessment-based regulatory structure for future reactors. This approach for future approval of designs and materials should provide a shorter path from material identification to final approval. However, the time when such an approach is developed and implemented for actual use is uncertain. There is, at least, the perception that the NRC is adopting a more progressive approach to its approval methodologies and processes, but the approval process for a new type of material will still be a long process.

3.4.3 Fabricability and technological maturity

Manufacturing the basic shapes required to make the control rod sheathing poses no particular problem for any of the materials. SiC_f/SiC is still produced in small quantities, although that may change by the time the actual control rods are required. The necessary equipment, capacity, and expertise are available for 800H and C_f/C composites. Machining is not particularly difficult, but each material poses its own design challenges. 800H will be operating in a temperature regime where creep could be significant, so care must be taken with stress concentrations. Composites must be designed so that threads are either avoided or properly supported by fibers, with consideration not only towards the integrity of the part, but the difficulty in qualifying the component design. SiC_f/SiC requires additional effort to design near net shape parts because of the very high cost of the high quality SiC fibers required for irradiation resistance. Both processing time and material availability could be issues if composites are chosen. Only two companies currently produce SiC fibers; both are Japanese and production is at the laboratory scale. Carbon fibers are mass produced by both domestic and foreign suppliers, but the industry is currently capacity constrained and certain types of fiber are very difficult to obtain. This may not be the case when material is needed for production of control rods, but would affect development.

The cost of developing ceramic fiber/ceramic composites for control rods is expected to be significantly greater than the cost of developing an alloy such as 800H. Raw material costs are trivial except for the composite fibers; the SiC fibers are 15 to 30 times more expensive than carbon fibers. Processing for all the composites is expensive compared to the metal because of the long times and high temperatures involved. Material and component testing is the largest cost, and that is also larger for composites because of the extra testing that would be required to standardize and qualify a new class of material. Compared to the development costs, the individual part cost of a component should be low, particularly if it is from one of the more mature industries with competitive bidding. Therefore, the lifetime of the component is of minimal significance, providing it is reasonable and the component can be changed during scheduled maintenance shutdowns.

The development of ceramic composite control rods should continue, even though they may not make the NGNP program deadlines due to challenges involved with achieving codification. In light of the revised design requirements of ~8×10^{21} n/cm^2 for the 60-year reactor lifetime, C_f/C composite material has sufficient radiation properties (up to ~8 dpa) for the NGNP control rod. The properties of SiC_f/SiC composite material far exceed the needs of this application, and in light of the expense, limited availability, and less-mature technology relative to C_f/C composites, and the expense of developing two composites, continued research on SiC_f/SiC composites is difficult to justify. Although the primary heat exchanger is currently limiting the outlet gas temperature to a maximum of 950°C, a composite control rod would also be required to push the reactor temperature, and thus the efficiency, higher. At this point it would be premature to narrow the control rod material choices to one metal with significant issues to resolve; continuing composites research will help to mitigate technical risk. Continuing the composites development may also allow the NGNP program to leverage work being performed by the French, South Africans, and Japanese. The programs in these nations have expressed interest in the NGNP composites work and may collaborate in the efforts to include the composites into the ASME code and to obtain Nuclear Regulatory Commission approval so that they can market reactors in the U.S.

3.5 Environmental Degradation

A report (Wright 2006)[14] was produced to examine what is known about the composition of the helium in reactor environments and the impact of long exposure of metallic alloys to these atmospheres. The influence of the environment of both microstructure and mechanical properties was discussed. Current programs within the international Generation IV research program were described and suggestions for additional work were discussed.

3.5.1 Mechanisms of degradation

Depending on the impurity concentration and the temperature, high temperature alloys can undergo oxidation, carburization, or decarburization. Carburization is associated with low temperature embrittlement and decarburization is linked to reduced creep rupture strength. The optimum coolant chemistry for long-term stability of high temperature alloys is slightly oxidizing and results in formation of a tenacious and protective Cr_2O_3 scale. The concentrations of H_2O and CO are of particular interest because they essentially control the oxygen partial pressure and carbon activity, respectively. Model chemistries used in many of the controlled studies tend to have higher levels of some impurities than those found in operating reactors.

The most critical metallic component of the NGNP is the heat exchanger. Inconel 617 is the primary candidate alloy for this application because of its superior creep resistance combined with high chromium content for environmental resistance. The mechanisms of environmental interaction between this alloy and prototype HTGR helium chemistries have been extensively studied. A modified type of Ellingham diagram that maps the ranges of carbon activity and oxygen partial pressure that result in each of the degradation mechanisms has been developed (Figure 12). Zone III in the Ellingham stability diagram is preferred for optimal chromia layer protection against corrosion. In addition, there is a maximum service temperature above about 950 °C for each alloy determined by the microclimate reaction or volatilization of Cr_2O_3.

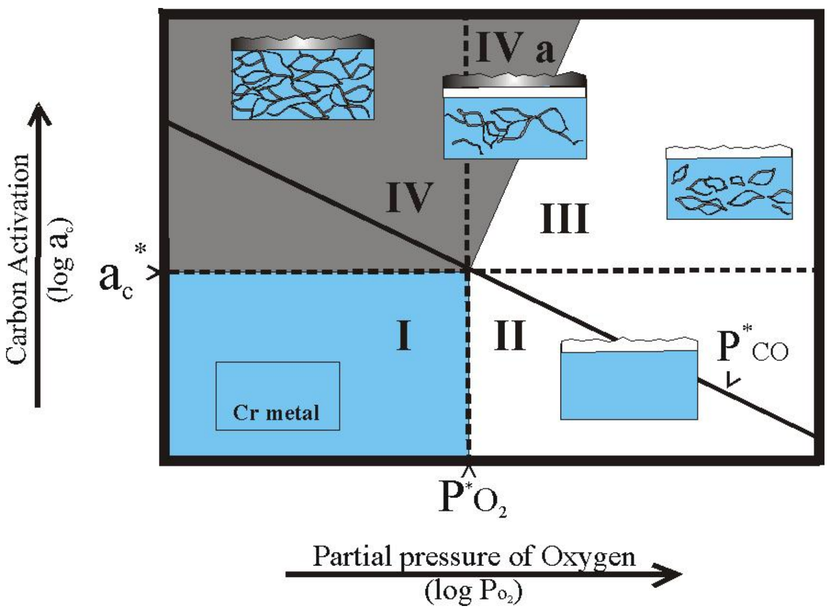


Figure 12. The modified Ellingham stability diagram for Alloy 617.

While the mechanisms of degradation are relatively well-known, the kinetics of the reactions for reactor conditions of high helium pressure and velocity are not well characterized. Most laboratory studies have been at very low flow rates to approach thermodynamic equilibrium for fundamental studies of corrosion mechanisms. Similarly, typical test protocols have employed a pressure of 2 atm, while the NGNP will operate at approximately 60 atm.

A number of very high temperature helium-cooled reactors have been built and operated for extended periods. The helium coolant in the primary circuit has been found to contain low levels of impurities after steady-state operation that can lead to an environmental degradation of the high temperature alloys used for internals and heat exchangers. All reactors which have operated to date have had similar impurity levels, and there have been no reported problems with failure of components on the primary side associated with environmental effects. The impurity levels in the coolant for HTGRs that have operated, perhaps fortuitously, were in the stable oxidizing condition (zone III on Figure 12) for chromia forming alloys. The massive amounts of graphite in the reactor play a major role in maintaining the gas chemistry.

3.5.2 NGNP research

The NGNP materials program has extended previous studies on environmental effects of prototype impure helium on Inconel 617 by increasing temperatures and using test coupons that incorporate fusion welds in controlled chemistry test loops. In addition, parallel studies have been initiated with a less well-characterized alloy, Haynes 230 which may suffer less internal oxidation. The objective of this work is to determine the effect of the H_2O and CO concentrations on the corrosion mechanism, and the range of gas chemistries that gives rise to stable oxide formation for these alloys at temperatures potentially up to 1000°C. This will experimentally verify the Ellingham diagram. The ultimate goal is to be able to predict the corrosion mechanism in effect for a particular gas chemistry at a given temperature for the selected alloy.

A closed-circuit, low flow velocity test loop has been designed and assembled at Idaho National Laboratory. It has the potential to continuously getter excess impurities and add necessary trace impurities based on mass spectroscopy measurements in a closed loop system. The system volume is about 20 liters, can be heated to 1000°C, evacuated to a pressure of 10^{-6} torr and subjected to a flow rate of 40 l/m. All impurities are added to the system as individual gases using a precisely controlled automated valve set-up. Much of the current effort has focused on control of the gas chemistry, including water, over sustained periods of time.

3.5.3 Generation IV research

Through an INERI agreement with the French Generation IV materials program, the upper temperature limit determined by the so-called "microclimate reaction," or Cr_2O_3 volatility, will be determined. The focus and approach of the two systems is somewhat different so that they are complementary rather than duplicative. The system makes dynamic weight gain measurements on a single test coupon subjected to a premixed gas composition. The French system measures the level of moisture, but does not attempt to control it. The French program has been carrying out experimental work for several years and preliminary results have been published.

The current US and French programs will contribute little to the two significant issues that remain to be investigated with respect to possible environmental interactions in the NGNP: environmental interaction at reactor pressure and the influence of very high velocity gas on the environmental interaction. While, it is anticipated that the reaction mechanism determined using low pressure test loops will also be observed at high pressure, there is some concern that the kinetics of reaction will be altered. Although particulate erosion is not expected to be of concern, it is possible that the steady-state gas chemistry at the specimen surface could be considerably different with high velocity flow.

3.6 Standards and Codes

A report was produced for the NRC by Shah et al. (2003).[15] The objective of the report was to review and evaluate currently available national and international codes and procedures to be used in design of high-temperature gas-cooled reactors (HTGRs) including, but not limited to, the PBMR and the GT-MHR designs. The approach included evaluation of the applicability of the codes, standards, and procedures to the materials that have been used or recommended for HTGRs, taking into account the HTGR operating environments.

Major findings of the evaluation were as follows:

1. Most of the materials needed for HTGR are not included in the code cases. New code cases are needed for these materials.

2. The maximum temperature permitted by the codes and code cases for the materials acceptable for HTGR components is lower (760°C) than the maximum temperature (≥850°C) that these

components may experience during operation. The scope of the code and code cases needs to be expanded to include the materials with allowable temperatures of 850°C and higher.

3. The codes and code cases do not provide specific guidelines for environmental effects, especially the effect of impure helium on the high temperature behavior (e.g., fatigue, creep, and creep-fatigue) of the materials considered. High-temperature fatigue life may be influenced more by environment than by creep damage for some materials. Carburization or decarburization in impure gaseous He environment may have an effect on high temperature fatigue life. This phenomenon needs to be explored by tests and the effects incorporated in a life predictive model.

3.6.1 ASME Code, Subsection NB

The ASME Boiler and Pressure Vessel Code (ASME B&PV Code) Subsection NB of Section III of the Code provides design rules for Class 1 components at relatively low temperature [≤371°C (≤700°F)] for ferritic steels and ≤427°C (≤800°F) for austenitic stainless steels and high-nickel alloys. The design of HTGR components needs to satisfy these low temperature design rules in addition to the high-temperature design rules discussed below.

The structural of a component below the creep range is assured by providing design margins against plastic distortion and/or instability under a single application of the maximum anticipated load; buckling due to compressive loading; incremental collapse, ratcheting or fatigue under cyclic loading or fast fracture instigated by a defect.

3.6.2 ASME Code, Subsection NH

Subsection NH provides the high temperature design rules for out-of-core nuclear structures. The rules were developed in support of the U.S. liquid metal fast breeder reactor (LMFBR) program. Therefore, the available material choice is limited to five candidates - Types 304 and 316 austenitic stainless steels (up to 816°C), 2.25Cr-1Mo steel (up to 593°C), Alloy 800H (up to 760°C) for applications other than bolt, and Alloy 718 (up to 566°C) for bolt. In addition to providing design rules to protect against low temperature damages such as plastic collapse, plastic instability, buckling, cyclic ratcheting, and fatigue, Subsection NH also provides design rules to account for potential high temperature damages such as creep rupture, excessive creep deformation, creep buckling, cyclic creep ratcheting, and creep fatigue. Except for ferritic steels, which exhibit ductile-to-brittle transition behavior, the code does not require design to protect against brittle fracture using fracture mechanics analysis. In view of the high ductility and toughness of the permissible materials, this is reasonable. The creep-fatigue evaluation procedure does not account for environment effects, and the use of the bilinear creep and fatigue damage summation rule may not be universally applicable to all materials.

To carry out high-temperature design, certain mechanical properties as functions of temperature are needed, from which the design allowables are derived after applying appropriate safety factors. These include tensile properties such as the modulus of elasticity and Poisson's ratio, yield strength, ultimate tensile strength with the stress-strain curves; creep properties such as stress vs. creep rupture time for base metals and their weldments, stress vs. time to 1% total strain, stress vs. time to onset of tertiary creep, constitutive equations for conducting time- and temperature-dependent stress-strain analysis and the isochronous stress-strain curves; and continuously cycling fatigue life as a function of strain range at a fast strain rate and creep-fatigue cyclic life involving cycles with various strain ranges and hold times.

Design rules were developed by the ASME Section III code committees on a consensus basis, and for the most part, these are followed by other countries as well. The two main difficulties of using

Subsection NH to the design of HTGRs are the rather limited choice of materials, and the code allowable temperatures are not sufficiently high for HTGR application. To expand the material choice for HTGR applications, high-temperature design data for other potential candidates, including their weldments, have to be generated. Therefore, the scope of this code needs to be expanded to include the materials with allowable temperature ≥850°C.

3.6.3 ASME Code Case N-499-1

Code Case N-499-1 may be applicable to GT-MHR reactor pressure vessel and connecting pipe made of low-alloy steels (SA 508 Cl 3 forgings, SA 533 Grade B. Cl 1 plates) with a normal operating temperature of 290°C, provided the requirements on allowable cumulative time and maximum temperature for upset, emergency, and faulted conditions are satisfied. The normal operating temperature requirement is satisfied, if the vessel is insulated or cooled by helium returning from the turbine. The scope of the code case should be expanded to include 9Cr-1Mo-V, which is specified for an uninsulated reactor vessel with an operating temperature of about 500°C. The code case does not address the effect of the helium environment on these materials. The effect of helium environment (including impurities) on elevated temperature (538°C) fatigue design curves, isochronous stress-strain curves, stress ruptures curves, and the creep-fatigue damage envelope needs to be evaluated to further assess the applicability of this code case to the low alloy steels used as pressure vessel materials.

3.6.4 ASME Code Case N-201-4

This code case provides rules for construction of core support structures in two parts. Part A extends the rules for construction of subsection NG for restricted service at elevated temperature without explicit consideration of creep and stress-rupture, whereas Part B alters the rules for service at elevated temperature by accounting for both creep and stress rupture effects. As a result, the highest temperatures permitted in Part A are lower than those in Part B. The rules are similar to those for Subsection NH. Code Case 201-4 does not provide specific guidelines for environmental effects, but states that the combined effects of exposure to elevated temperature, contacting fluid, and nuclear radiation on material properties shall also be considered. Therefore, the effects of helium coolant (with impurities) on the material properties need to be evaluated.

Code Case N-201-4 provides design rules for five materials: ferritic steels 1 Cr-0.5 Mo-V and 2.25 Cr-1 Mo, Type 304 and 316 stainless steel (SS), and Alloy 800H. Some of these materials are not suitable for the high temperatures anticipated for core support structures in HTGRs. Ferritic steel, 1Cr-0.5Mo-V, is not suitable because the maximum allowable metal temperature in Part A is low (538°C), and its use is not permitted by Part B of the code case. Austenitic stainless steels (types 304 and 316 SS) have commonly been used for high-temperature steam application due to their excellent strength retention at high temperatures. However, a sufficient database is not available for the use of these alloys at temperatures of 600°C and higher in the impure helium environment of HTGRs. Furthermore, the austenitic stainless steels exhibit high thermal expansion rates and low thermal conductivity. This results in the development of high thermally induced stresses during heating and cooling, which can cause thermal fatigue and creep. Ferritic steel, 2.25 Cr-1 Mo, may be used for the reactor core support structures and internals located in the upper portion of the GT-MHR vessel, where the metal temperatures may be less than 500°C because of the low core inlet temperature of the coolant. Alloy 800H may be used to fabricate the support structures and internals that are exposed to the maximum temperature of 760°C.

Since the GT-MHR core support structures, especially those located in the lower portion of the reactor pressure vessel, may experience temperatures ≥850°C, the scope of the code case needs to be expanded to include the materials with higher allowable temperatures. The candidate materials for core support structures and vessel internals are Alloy 617 at high temperatures and 9Cr-lMo-V steel for lower temperatures. A draft code case discussed later provides design rules for Alloy 617. Code RCC-MR provides design rules for 9Cr-lMo-V steel.

3.6.5 Draft ASME Code Case for Alloy 617

The draft code case for Alloy 617 provides design rules for very-high-temperature reactors (HTGRs) with reactor outlet temperatures of about 950°C. The original request to the ASME B&PV Code Committee for design rules for very-high temperature nuclear components came from the U.S. DOE and one of its contractors. An ad hoc task force of the ASME Code was established in 1983 to address the design of reactors operating at very high temperatures. The task force was organized under the jurisdiction of the Subgroup on Elevated Temperature Design of the Subcommittee on Design. The Task Force completed the draft code case in 1989 and submitted it to the Subgroup, which later approved the code case. No further work was done on the draft code case because of the lack of further interest from US DOE and its contractor.

The draft code case focused on Alloy 617 because it was a leading candidate of designers, and there was a significant material properties database at the temperature of interest. The code case focused on all the failure modes that are addressed by Subsections NB and NH, including non-ductile failure. The last failure mode was considered because of the significant loss of fracture toughness in Alloy 617 after long term exposure to high temperatures.

The draft code case was patterned after relevant portions of Code Case N-47 and was limited to 982°C (1800°F), and a maximum service life [total life at temperatures >427°C (>800°F)] of 100,000 h or less. The main reasons for this shorter design life than the 300,000 h specified in Subsection NH are (1) the uncertainties of data extrapolation at very high temperatures, and (2) at long times, the allowable stresses are less than 1 ksi at the highest temperature; there is lack of experience in designing reliably at such low allowable stresses.

Most of the design rules provided by the draft code case are similar to those provided by Subsection NH, although some are different because the draft code case considers higher temperature and different material. One of the main differences is that Alloy 617 exhibits unique behavior that includes (1) lack of clear distinction between time-independent and time-dependent behavior, (2) high dependence of flow stress on strain rate, and (3) softening with time, temperature, and strain. Therefore, design rules of Subsection NH that are time- and rate-independent, or strain hardening idealizations of material behavior, required careful consideration in the draft case. For example, the case specifies that inelastic design analyses for temperatures above 649°C (1200°F) must be based on unified constitutive equations, which do not distinguish between time-independent plasticity and time-dependent creep.

Extended exposure at elevated temperature may cause a significant reduction in fracture toughness of Alloy 617. The draft case requires a fracture mechanics analysis to justify the ability of the component to withstand the expected service conditions, especially when the component cools down to lower temperatures. Because of this concern for potential loss of fracture toughness, Alloy 617 bolting is excluded from the draft case. In addition, exposure of cold worked material to very high temperatures results in recrystallization. Therefore, cold worked Alloy 617 is also excluded from the draft case.

The draft code case is based on a limited database and minimal service experience. Further development of the draft code case is needed before the alloy could be satisfactorily and reliably applied. The recommendations for further development are presented below in three categories:

Actions Required to Complete the Draft Case

- Alloy 617 must be added to the low-temperature rules of Section III.
- Weldment stress rupture factors must be added.
- Thermal expansion coefficient must be added.
- Additional isochronous stress-strain curves at 427°C to 649°C must be added.

Material Data Needs

- Weldment fatigue data.
- More complete creep-fatigue database.
- The synergistic effects of aging, environment (including radiation), loading, and temperature.
- The effects of aging on toughness of materials.

Structural Design Methodology Needs

- Unified constitutive model are needed for developing isochronous stress-strain curves.
- Some very high-temperature, time-dependent tests of Alloy 617 structural models are needed to (1) provide better understanding of structural behavior and failure modes, and (2) validate inelastic analysis methods and failure modes.
- Simplified ratcheting evaluation procedures need to be developed for temperatures above 649°C.
- The use of linear damage fractions as the basis of the creep-fatigue rules is probably the biggest shortcoming of the draft case. A basic effort is needed to identify and experimentally validate a more fitting damage theory.

3.6.6 French Code RCC-MR

RCC-MR was developed in France as a high-temperature extension to RCC-M, for the French breeder reactor program. The basic rules in RCC-MR are similar to those in ASME Code Subsection NH. RCC-MR provides more detailed instructions for fatigue and creep-fatigue design analysis than given in the ASME Codes. It also uses a somewhat different approach for analysis of creep ratcheting without the use of isochronous stress-strain curves. But the basic safety factors used in the generation of design curves in the two codes are comparable. The choice of materials in RCC-MR is also limited; however, modified 9Cr-1Mo is codified in RCC-MR.

3.6.7 British Procedure R5

Procedure R5 is a comprehensive assessment procedure for the high temperature response of structures. The procedure addresses several aspects of high temperature behavior, including creep and creep fatigue crack growth, and both similar and dissimilar metal welds. The main objective of the procedure is to ensure that failure of both defect-free and defective components by creep rupture is avoided. The calculation of the expected lifetime is performed by conservative approximations based on Reference Stress Techniques. Procedure R5 permits use of ductility exhaustion for creep-fatigue evaluation, if the material data are available. If appropriate data are not available, linear damage summation, similar to that recommended by ASME Code Subsection NH, is used.

Procedure R5, which is a guideline and not a code, considers creep cracking explicitly. Neither the ASME Code nor the RCC-MR Code addresses the subject because creep cracking is more of an issue in residual life assessment than design. Therefore, the use of this procedure in the design of HTGR components is limited. However, the procedure presents the use of ductility exhaustion as an alternative to the linear damage rule for calculating the creep component of damage. The use of ductility exhaustion instead of linear damage rule may be more suitable for estimating creep damage in the HTGR materials and should be evaluated for incorporation in Subsection NH.

3.6.8 Life-Predictive Models for High Temperature Fatigue

The linear damage rule (D-diagram) approach, which is used by many design codes including the ASME Code, was originally developed from tests on austenitic stainless steels for which tensile creep damage (intergranular cavitation) rather than environmental effects plays a major role in determining creep-fatigue life. There are other life predictive models, such as Damage Rate Equations, Strain Range Partitioning Equations, the Frequency Separation Equation, and the Ductility Exhaustion Equation, which are potentially more accurate than the linear damage rule and need to be explored for application to HTGR materials. Because of its simplicity of use, the linear damage rule has been force-fitted for other materials (e.g., ferritic steels) whose high-temperature fatigue life is influenced more by surface cracking (environmental effect) than by bulk creep damage. However, extrapolation beyond the database using linear damage rule cannot be justified. Specialized models for predicting surface cracking due to environmental effects need to be developed for such purposes. Carburization or decarburization in impure gaseous He environment may have an effect on high-temperature fatigue life. This phenomenon needs to be explored by tests and the effects incorporated into a life predictive model.

3.7 High Temperature Design Methodology

The essence of this section also appeared in last year's program plan. It presents the data and models required by ASME Code groups to formulate time-dependent and independent failure criteria and rules that will assure adequate life for components fabricated from materials identified for the NGNP. The generation and development of design data needed to quantify the failure criteria and design rules (e.g., uniaxial creep-rupture data, are not included in this task); this data is specific to the High Temperature Materials and Pressure Vessel sections – although the activities are integrally linked. Experimentally based constitutive models that are the foundation of inelastic design analyses specifically required by the ASME code are discussed in this section, and appropriate simplified design procedures provided as necessary for use in conceptual and preliminary design phases, as well as possible final design applications. All of the key parts of a high-temperature design methodology - failure criteria, inelastic design analysis methods, and alternate & simplified design procedures - require experiments verified by an assortment of representative structural tests. Efforts will be focused on materials for the reactor pressure vessel (508/533 steel and Gr91 steel) and the intermediate heat exchanger (Alloy 617, 230, X/XR).

The primary roles of the HTDM task, which is an integral and inseparable part of the overall NGNP materials program, are as follows:

1. Provide the data and models required by ASME Code groups to formulate and verify time-dependent failure criteria that will assure adequate life for components fabricated from the selected NGNP materials.

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

3. Review and provide appropriate alternate and simplified design procedures required for the conceptual and preliminary phases of design and useful for many less-critical regions in final design.

4. Assist in the development of a “Guideline for Inelastic Analysis and Life Prediction” to replace or update the DOE-NE (DOE office of Nuclear Energy) guideline F9-5T that was developed specifically for high temperature design of LMFBR components. The guideline will be effective in obtaining acceptance from the NRC for licensing of the NGNP.

5. Assist in identifying existing NRC concerns and any remaining or unresolved issues following a request from DOE to obtain a license from the NRC to construct and operate the NGNP. Develop and execute R&D plans to resolve such concerns.

3.7.1 Alternative Primary Load Design Methods

Currently, the analysis techniques permitted in ASME-NH for “Load Controlled Stress Criteria”, design criteria, only permit the use of elastic analysis (NH-3214.1). The methodology in the ASME code requires that the stresses resulting from analytical or computer aided elastic analysis to be classified into primary, secondary, and peak stresses. Many of the tools available to engineers today are not capable of classifying stresses as required by ASME code, or the effects of geometry, such as a stepped cylinder, make classification of stresses very difficult or impossible. Proper classification of primary stresses is an essential step in the design process. Primary stresses are applied to design rules and compared to allowable stress limits to avoid various failure modes, particularly time-independent failure from short term loading events or time-dependent failure resulting from creep damage or creep rupture. Furthermore, a compact IHX structure does not lend itself to implementation of current ASME design procedures. An alternative method, other than full inelastic analysis, is desired.

Recent efforts within the HTDM task at ORNL have focused on adoption and adaptation of the reference stress approach as a permissible inelastic analysis method to satisfy load controlled stress criteria in ASME-NH.[16, 17, 18] These stress criteria address both time-independent and time-dependent load controlled failure of structures, i.e. plastic collapse and creep/creep-rupture respectively. While the reference stress approach is extensively used in nuclear and non-nuclear industries in Europe, adoption of the approach by subcommittee members in NH has not occurred. The foundation of stress classification in ASME is based upon limit analysis; the reference stress approach is an extension of limit analysis to time dependent failure.

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

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

NH is incapable of addressing primary load limits for structures with inherent complexities such as in a compact IHX because the structure does not resemble a typical pressure boundary; specifically, the great difficulty and/or inability to classify stresses properly and to implement stress intensities with the design criteria of NH-3222. Similarly, the current NH load controlled stress criteria do not address discontinuity effects such as notches or intersections of nozzles with cylinders or spheres – clearly, a compact IHX contains numerous discontinuities. On the other hand, the reference stress approach is capable and useful in conducting conceptual, preliminary, and in many instances final analysis for selection and design of structures with discontinuities. Furthermore, the reference stress approach does not require a sophisticated material model, nor does it require a sophisticated creep database and a creep exponent – only the geometry and loading need to be defined along with an arbitrary yield strength. Comparison of reference stress approaches with coupon samples, structural feature-like test samples, and small scale versions of a compact IHX will permit extensive direct comparisons of the materials and design of an IHX; correlation and verification will be necessary for regulatory acceptance. This activity will directly address some concerns of "unverified analytical tools" and "limited supporting technology and research" documented in the 1994 NRC review of DOE's submittal of GE's PSID for PRISM,[19] as well as concerns brought out in the NRC/ACRS review of the CRBR project.[20]

3.7.2 Simplified Methods and Criteria:

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

Simplified design analysis methods were developed during the LMFBR programs in an attempt to conservatively predict the effects of cyclic loading on creep strain accumulation, and to prevent ratcheting.* The very high temperatures of the NGNP reactor raised concerns regarding the validity of the simplified methods for HTGR applications. In fact, the Inconel 617 draft code case does not permit the use of simplified methods above 649°C.[21] Verification of the validity of the simplified methods for HTGR applications has been the focus of ongoing work at ORNL.[16, 17, 18, 22] Unpublished results of various load histories reveal that one of the simplified methods, the B-1 Test, is not conservative when applied to a thin tube of Alloy 617 under constant pressure subjected to cyclic linear thermal gradients; a conservative solution is being developed. The results have significant implications with respect to the current methodology, and will likely impact other simplified methods in Appendix T as well. No component should pass any simplified analysis and design criteria that would otherwise be rejected by a more detailed criteria and inelastic design analysis route. While the simplified methods are not intended to be rigorous solutions with great accuracy, the intent is to provide a conservative, simple, and efficient screening tool for conceptual and preliminary designs, and in many instances for final design stages of components that will not experience demanding or challenging loading. This approach leaves more time for designers to focus efforts on more demanding component design issues in final design stages.

The applicability of the simplified methods to general structures and loading, other than the simplified geometry and loading for which the methods were developed, is a concern – especially in light of recent unpublished results. The NRC has raised concerns regarding the use of “unverified analytical tools” and “limited supporting technology and research”, as stated earlier. Their concerns also apply to cyclic loading effects.

3.7.3 Failure Models for Design Criteria- Creep-fatigue

Cyclic loading introduces a failure mechanism that must be considered in structural components at elevated temperatures: creep-fatigue. Code developers and researchers worldwide generally recognize that the current linear damage accumulation rule for creep-fatigue used in Appendix T and many other design codes has significant shortcomings, particularly at higher temperatures and longer times. Various improvements, such as those based on ductility exhaustion and damage rate concepts, have been proposed, but none have been backed by sufficient R&D to allow their adoption as a replacement for linear damage in Appendix T. These shortcomings should be remedied for the NGNP, particularly for use of Gr91 steel at elevated temperatures and the IHX material (Alloy 617 or 230). Creep-fatigue has been a technical and safety issue that the NRC has raised concerns about consistently. The effect of the environment, i.e. impure helium, may have significant effects on the creep-fatigue performance of materials; while not addressed by the ASME code, such effects must be demonstrated as insignificant or appropriate means taken to account for differences.

The creep and fatigue performance of materials will certainly change significantly with variations in product form and accompanying grain size, e.g. plate vs. sheet vs. foil with large or small grains. Data and understanding of such variations for both Alloy 617 and 230 for the IHX must be generated and obtained, for both air and impure He environments. Similar test programs will be required if Gr91 steel is to be used for reactor or core internals, and possibly the reactor pressure vessel if the operating temperature of the RPV is considered high enough that creep is significant. Air testing provides a baseline that is required and used for the ASME code; environmental effects are required by the NRC for licensing in addition to the baseline. Conducting tests of this nature is not trivial; appropriate test facilities are rare if they exist at all, especially for controlled impure He facilities. Refurbishing or

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

modification of existing equipment requires careful planning, time, and resources. Such activities will take at a minimum 1 year to establish capabilities and work out significant issues; additional time and resources will be required to expand upon such capabilities and will be driven by available resources. Efforts at University of Illinois, carried out under a NERI award, will be very beneficial to this effort.

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

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

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

3.7.4 Extrapolation of Creep Data for Very Long Service Applications

The 60-year plant life for the NGNP is unprecedented. No creep database contains test durations of 525,000 hours. Consequently, extrapolation of creep data is required. Typically the longest creep test durations in the U.S. are limited to several tens of thousands of hours, with limited data approaching 100,000 hours. A nickel based super alloy IHX will not operate for 60 years, but must be replaced – likely several times. Assuming an IHX will operate for 20 years, testing will need to be conducted at various stress levels and times to verify what deformation mechanisms will be operative during service to determine if extrapolation of data is justified, to what extent data can be extrapolated, and how to extrapolate.

Extensive databases in terms of data and test durations on the order of 100,000 hours or more exist for materials such as Alloy 800H that have decades of R&D and industrial experience. However, the U.S. and ASME database for Alloy 617 is not as large nor does it include as extensive test durations. Furthermore, various compact IHX designs entertain a wide range of product thicknesses from 0.2 mm (8 mils) to 2 mm (80 mils). Assuming that a minimum of 10 grains would be desired across the thickness, grain sizes would vary from maximum sizes of about 20 μm (ASTM G.S. 8) to about 200 μm (ASTM G.S. of 1.5). The current draft code case for Alloy 617 would only support grain sizes 127-360 μm (ASTM G.S. 0 to 3); selection of smaller grain size product forms (thin section or thin product forms) for a compact IHX will require additional testing and R&D. Extrapolation of creep data used for the Inconel 617 draft code case is a significant issue for the periods required for NGNP applications and will be a major NRC concern; similar concerns will arise for material with smaller grains as the database is less extensive. Unfortunately, Special Metals reports that the raw creep data for Alloy 617 no longer exists – the data only exist in processed and tabulated form. This has significant ramification in terms of generating test data to support constitutive modeling efforts, apparent errors in tabulating or processing some data, and possibly extrapolation of test data.

Alloy 230 has a less extensive database compared to Alloy 617, and has similar issues with respect to various grain sizes and extrapolation of data for NGNP needs and compact IHX designs. Haynes has provided creep curves for Alloy 230 to ORNL for use in ASME code needs; this likely will significantly reduce testing needs for codifying this material although efforts are needed to assemble, analyze, and summarize the data.

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

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

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

3. A material must be selected for fabricating an IHX – Alloy 617 or 230. The selection of an IHX material will clearly depend upon limited interim mechanical property data, aging, and environmental studies. The option of using Hastelloy XR may still be considered a viable option

if the U.S. can reach an agreement with Japan to acquire their database of Hastelloy XR to expedite an ASME code case or acceptance by NH; Hastelloy XR has been accepted by the Japanese nuclear code and used in the Japanese HTTR at temperatures of 850°C-950°C. The limitations of this alloy will be similar to the current draft code case for Inconel 617 in terms of grain size and product form and limitations on service time.

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

3.7.5 Weldments and Discontinuities

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

A review of the existing weld reduction factors for materials currently in NH will be required to begin to address the NRC concerns.[21] Additional testing may be required, especially demonstration of the welding process of thick sections of Gr91 steel, filler metal, thick section properties, and heat treatment. Collaboration with AREVA will be crucial in order to complete this activity successfully and in the timeframe for the NGNP. Extensive R&D will be required to demonstrate the joining process, filler metal, and any necessary heat treatment for a to be determined material, grain size and product form, and joining process for the to determined IHX. Even if a grain size covered by the Alloy 617 draft code case is used, weld metal creep rupture data is sparse. Overall, collaboration with AREVA and/or GIF nations would assist greatly in the costs, time, and scope of this activity.

Like metallurgical discontinuities, geometric discontinuities (i.e., notches and other local structural discontinuities) are sources of component failure initiation. The adequacy of the design methodology to handle such discontinuities is a reliability and licensing issue, particularly when heat-to-heat variability, strain hardening/softening, triaxiality, aging and environmental effects on ductility and creep ductility, and cyclic loadings are considered. This was the second unresolved issue (after weldments) in the CRBRP licensing hearings, and again a multi-year development program was required by the NRC. Reviewers felt that the effects of stress gradients were not reflected in creep-fatigue design limits and that general notch weakening and loss of ductility under long-term cyclic loadings were not well understood. Notches

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

will need to receive particular attention in the development of the required high-temperature structural design technology for the NGNP and other Gen IV reactors. Selection of materials for the NGNP will permit focused efforts as material behavior can play a significant role in adequately resolving this issue.

3.7.6 Inelastic Design Analysis Methods

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

Experimentally based constitutive equations must be developed for NGNP materials where distinguishing between plastic and creep behavior is not possible: specifically nickel based super alloys such as Inconel 617 and Haynes 230. AREVA is generating data to support such development for Alloy 230. AREVA indicates that the RPV will experience much lower temperatures than originally anticipated. The issue of indistinguishable creep and plasticity no longer applies for Gr91 steel. However, AREVA is pursuing approval for ASME to raise the insignificant creep temperature for Gr91 steel from 371°C to approximately 400-450°C. Depending upon if and how Subsection NH modifies the definition of insignificant creep, and available data to support the criterion, a constitutive equation will likely be required to capture the effects of any microstructural changes over long periods of time and at temperature on subsequent material behavior in terms of strain accumulation.

Recent modeling efforts on Alloy 617 includes rough first order predictions based on Ashby deformation mechanism equations for various grain sizes. AREVA has access and/or is generating relevant data that would be useful in developing constitutive models. The results indicate that smaller grains sizes can result in the change of the deformation controlling mechanism from power law creep to diffusion dominated mechanisms for times, temperatures, and stress levels applicable to the NGNP IHX. Consequently, the maximum constant stress permitted over a 100,000 hour period can be significantly lower than that in the larger grain material currently in the draft code case for Alloy 617; the difference obviously increases with temperature. Since the U.S. has no experimental data to support modeling efforts for different grain sizes at this point, establishing a formal collaboration mechanism would be of great value to this subtask.

Experimental data are required to first validate the model, and then make improvements. For conceptual and preliminary design needs, the model, preferably experimentally based, should be incorporated into a suitable structural analysis program and coupled with thermal analysis to investigate a variety of thermal and mechanical load histories, design configurations, and so forth. Full inelastic simulation of an IHX would be very challenging given the complexity of the structure, requirements of mesh size, and so forth. Incorporation of simplified design methods that utilize elastic-plastic finite element methods and lend themselves to complex structures will be of great value. Isochronous stress strain curves, generated by even a rough model, permit designers to investigate design issues and gain experience in identifying difficulties and issues associated with various design configurations, material proper limitations, and so on. Once initial and sufficient mechanical property data (e.g., tensile, creep, fatigue, and relaxation testing) have been compiled, the experimental data can be used to develop an experimentally based constitutive model.

When a selection of material is made, more extensive testing and modeling development will be required. A final iteration on the equations for the final NGNP materials will likely be required. To be of value, this must be completed early in the final design phase. This final iteration will correct shortcomings identified from the results of structural tests and from design analysis experience. While every effort will be made to utilize information from the data generation tasks, experience shows that an extensive test program of uniaxial and biaxial exploratory tests will be required to establish key response features resulting from various mechanical and thermal load histories, and biaxial loading paths. Establishing test matrices for such tests a priori is not easy; tests must be planned and carried out in concert with model development efforts, and subsequent tests depend on the findings from the previous tests.

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

These tests are very demanding in terms of test equipment, since they involve precise control of load history and conditions. Suitable test facilities, controllers, and extensometers must be designed, procured, and built. Often potential surfaces (akin to yield surfaces or loading surfaces in classical plasticity) must be established in biaxial stress space. One example of this in past unified equation development efforts was the determination of surfaces of constant inelastic strain rate. In addition to uniaxial specimens, thin-walled tubular specimens subjected to combined internal pressure, axial loading, and torsional loading will be employed. Also, structural tests are recommended against which inelastic analysis predictions can be benchmarked.

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

3.7.7 Confirmatory Structural Tests and Analyses

In the case of most of the current Subsection NH materials and design methodology, confirmatory time-dependent structural tests have provided data that either

1. validated the high-temperature design methodology (inelastic design analysis methods, simplified design methods, alternate design methods, and design criteria) or

2. led to changes in inelastic design analysis guidelines or ASME Code rules. The role of structural tests will be even more important for the NGNP because of the lack of long-term service experience.

This need was recognized by the developers of the draft Alloy 617 Code case. The need for very-high-temperature, time-dependent tests of Alloy 617 structural models was identified to

1. provide a better understanding of structural behavior and failure modes,

2. validate inelastic analysis methods, and

3. provide some applications feedback to the Code.

Even in the case of Gr91 steel, applicable experience within ASME is limited as the material was recently approved for use in Section III and Subsection NH, and while structural tests were planned under the LMFBR program, they were never carried out.

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

3.7.8 Impact & Risk of Scope of Activities and Schedule

The objective of the HTDM task is to provide technology and R&D in support of ASME code needs for a vendor to obtain a construction and operating license from the NRC for the NGNP. The HTDM encompasses materials and design methods & criteria for the RPV and the IHX; therefore, success of this task is dependent at least upon adequate funding and activities in support of these.

3.7.8.1 Very High Temperature Alloys

The definition of design and material performance requirements for the IHX is a critical starting point. In order for a decision to be made with respect to a design and material selection, material data is required. Presently, the selection of Inconel 617 or Alloy 230 has yet to be made. Furthermore, selection of a product form with a small grain size or large grain size has yet to be made as data is lacking to support such a decision. Selection of a material inherently includes a feasible joining process with acceptable weldment properties. The scope of these activities is immense, particularly if options of Alloys 617 and 230 along with grain size options are all considered. Funding requirements follow suit. Many of the subtasks discussed above depend upon selection of a material and product form (grain size), e.g. creep-fatigue, inelastic design analysis methods, failure models, confirmatory structural tests, and to some extent even simplified methods. Selection of Inconel 617 in the product form and grain size supported by the draft code case will relieve some requirements for R&D; however, not by an order of magnitude, and it remains to be determined if an IHX made of large grain size Alloy 617 can be designed to meet performance requirements and be economical. Acquisition of German Inconel 617 data for large grain sizes could greatly expedite addressing the extrapolation issue; the database is available, but acquisition remains stalled due to legal issues. If a thinner product form is required which contains smaller grains, Alloys 617 and 230 would be on nearly equal footing as each would require significant data generation, Alloy 617 perhaps having a slight advantage if existing studies of environmental effects for large grain Inconel 617 apply directly to smaller grain product forms.

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

1. Select Alloy 617 in the grain size supported by the draft code case.
   Pros:
   - This will reduce data generation requirements, assisting in with schedule constraint.

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

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

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

   Cons:
   - Requires some additional preliminary material property data.
   - Requires joining process for thin section product.
   - Vendor material selection may differ; joining process may differ significantly as well. Focus of DOE supported R&D would need to change – less gain in early selection of Alloy 617.

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

3. Pursue Alloy 617 and 230 in product form and grain sizes for which data exist today.
   Pros:
   - Recent acquisition of Alloy 230 creep data from Haynes permits this option.
   - Grain size selection reduces data generation requirements, assisting with schedule constraint.

   Cons:
   - Requires some additional preliminary material property data, e.g. creep-fatigue of Alloy 230.
   - Requires joining process for thin section product, both Alloy 617 and 230.
   - Vendor grain size selection may differ to support different IHX design concept; joining process may differ significantly as well. Focus of DOE supported R&D would need to change – no gain in early selection of grain size.

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

4. Pursue Alloys 617 and 230 in all product form and grain sizes.
   Pros:
   - Recent acquisition of Alloy 230 creep data from Haynes.
   - Vendor material and grain size selection encompassed.

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

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

5. Pursue Alloys 617 and 230 in thin product form (small grain sizes).
   Pros:
   - Data generation provides needed material properties for selection of material and section size.

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

   Risk:
   - Vendor will not select small grain size product form.

6. In addition to scenarios1-5, the option of using Hastelloy XR exists.
   Pros:
   - Requires no action or commitment on the part of DOE, until and if a vendor selects Hastelloy XR.
   - Japanese currently using Hastelloy XR in heat exchanger in the HTTR – experience in HTGR.
   - Material is codified in Japan for nuclear use; accelerates code acceptance in ASME.
   - Extensive environmental database and experience exists.

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

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

3.7.8.2 Reactor Pressure Alloys (Grade 91 Steel SA 508/533 Steel).

The RPV materials being considered are Grade 91 steel (AREVA) and SA508/533 steel (South Africa). The use of Gr91 steel is not limited to the RPV in the AREVA design, but will also be used in internals; as such, creep-fatigue of Gr91 steel is an issue in the AREVA design – regardless of the RPV.

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

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

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

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

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

3.7.9 ASME Codification & NRC Licensing

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

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

3.7.9.1 Selection of materials to reduce the time and effort required R&D.

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

3.7.9.2 Pursue issues related to Grade 91 and R&D to support thin section Inconel 617 or Haynes 230 for use in the IHX.

This will be the highest risk approach. Essentially, a new material would need to be tested and approved by ASME code. With full funding, the timeline could extend another 2-3 years. The probability that ASME code approval for Inconel 617 or Haynes 230 thin section material may not be obtained prior to submitting the FSAR exists; as such, the FSAR would need to be submitted prior to ASME approval. This is a licensing risk that would need to be considered. If unrestricted funding is not provided, a realistic timeline for construction and operation of the NGNP will extend as well.

3.8 NERAC Report

A report summarizing a review performed by NERAC of the NGNP Project as required by the Energy Policy Act was issued in January, 2006.

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

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

Table 5. ITRG recommendations, actions and observations.

ITRG Recommendation	DOE Action	NERAC Comment
A) ITRG recommendations were based on a NGNP schedule for initial operation in 2017	EPACT legislation provided a schedule with two phases; i.e., 2005-2011 - R&D and 2011-2021 - construction and licensing	DOE should re-examine and modify its detailed R&D program plan to meet the target timetable for technology selections
B) For the two HTGR reactor concepts, neither is more likely to be successful for the NGNP	DOE has modified its R&D plan to provide a more balanced effort for prismatic & pebble HTGR	No specific comments here.
C) It is impractical for a molten-salt-cooled reactor development effort to be successful for NGNP	DOE has adjusted its R&D program with only small MSR exploratory feasibility studies	There has been an appropriate alignment between DOE plans and ITRG recommendations
D) Use of molten-salt in a heat transfer loop in the NGNP may be a desirable design concept	DOE has adjusted its R&D program and has started a design activity in this technical area	There has been an appropriate alignment between DOE plans and ITRG recommendations
E) Consider alternatives for licensing and purchase of viable technology from offshore vendor.	This alternative is not consistent with EPACT 2005 and DOE plan	No specific comments here
F) NGNP fuel development should focus on processes that have most successful worldwide experience base (e.g. UO_2 kernel)	DOE agrees that worldwide fuel experience base be considered, but technically disagrees with ITRG focus only on UO_2 kernel	NERAC subcommittee examined this item in detail (F and G), and recommends that the fuel R&D program be reconsidered (VIII.7)
G) NGNP fuel development plan should incorporate UO_2 & UCO kernels in R&D to determine the influence of fuel manufacturing processes on fuel quality	DOE agrees and has incorporated this approach in AGR in-pile tests with UO_2 & UCO kernels; it has modifications planned for 2011 technology selection goals	NERAC subcommittee recommends that the fuel R&D program reconsider how AGR in-pile tests can be optimized for 2011 technology selection target
H) Fuel development R&D plan should be consistent with overall NGNP R&D plan and schedule	EPACT 2005 has provided an overall NGNP schedule and DOE is aligning its R&D schedule to it	The overall R&D plan needs to be aligned with the NGNP reassessed mission and associated schedule
I) ITRG views need to achieve a high outlet temperature in NGNP be justified, and suggested a reactor outlet value of 900-950C	DOE agrees and aims to set NGNP reactor outlet temperature by hydrogen production needs and the material capabilities	DOE approach is consistent with ITRG recommendation; NERAC recommends as wide a range as possible given tech. constraints

ITRG Recommendation	DOE Action	NERAC Comment
J) An indirect cycle power conversion concept fulfills the high-level functional objectives	DOE agrees this minimizes risk; this approach is needed for hydrogen production, but may be premature for electricity now.	DOE plan is generally consistent with ITRG recommendation, but subcommittee recommends the NGNP mission be reassessed
K) The development of a high-temperature hydrogen production capability should be accelerated	DOE agrees and notes that there has been greater R&D activity in FY-05 and will be in FY-06	NERAC is pleased with the R&D plan and research activity for hydrogen production
L) Resource intensive R&D can benefit from direct international and industrial participation	The GIF provides international input to NGNP and DOE plan is developing to involve industry	NERAC subcommittee has some detailed suggestions for industrial input (Section VIII.)
M) ITRG noted that design uncertainties (IHX, RPV) be addressed with focused R&D	DOE has identified these and other items for R&D plan for revision to meet 2011 target	No specific comments here
N) ITRG concern: Electricity, Hydrogen or a dual mission	EPACT 2005 has given guidance on this issue	NERAC subcommittee considers this a key decision (Section VI.)

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

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

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

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

- Trade study results to select the NGNP's mission

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

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

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

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

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

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

3.9 ITRG Materials Issues and NGNP Materials Program Response

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

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

3.9.1 Upper Temperature Limit

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

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

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

3.9.2 Pressure Boundary Time-Dependent Deformation

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

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

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

3.9.3 Fabrication, Welding, Inspection, and Monitoring

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

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

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

3.9.4 ASME Codes and Standards

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

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

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

3.9.5 Corrosion and Oxidation

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

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

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

3.9.6 Microstructural Stability

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

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

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

3.9.7 Graphite

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

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

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

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

3.9.8 Extrapolation of Limited Data

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

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

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

3.9.9 Advanced Materials Development

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

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

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

3.10 DOE Initiative to Address ASME Code Issues

3.10.1 Introduction

The DOE Generation IV Nuclear Energy Systems Program includes six reactor systems: the Supercritical Water Reactor (SCWR), the HTGR, the Lead-cooled Fast Reactor (LFR), the Gas-cooled Fast Reactor (GFR), the Sodium-cooled Fast Reactor (SFR), and the Molten Salt Reactor (MSR).[23] The NGNP is a helium-cooled HTGR reactor concept with an outlet temperature in the range of 850-950°C, and is of primary interest to DOE. Nuclear structural component construction in the U.S. complies with Section III of the ASME Boiler and Pressure Vessel Code,[24] although licensing is granted by the NRC. The extensive use of graphite, particularly its use as a structural material in the NGNP is novel. A number of additional technical topics were identified by DOE, ORNL, INL, and ASME to have particular value with respect to the ASME Code. A three-year collaboration between DOE and ASME was established that addresses twelve topics in support of an industrial stake holder's application for licensing of a Generation IV nuclear reactor. Efforts to address the first five tasks are currently initiated. The majority of these tasks are relevant to action items within ASME Section III Subsection NH, and the nature of the topics inherently include significant overlap, and in some cases parallel activities on the same issue. These tasks are summarized below.

3.10.2 Task 1: Verification of Allowable Stresses

The task will require formal access and use rights of the various materials databases. The original database needs to be assembled and reviewed, including methods used to set the time-dependent allowables. Comparison of European and Japanese databases and the methods and procedures used by these sources to set allowable stresses are needed to provide guidance on how to set allowable stresses for ASME Section III Subsection NH. For Alloy 800H, the U.S. database needs to include existing data produced up to 925°C, including both creep and stress rupture data. An updated compilation of the creep and rupture data for Grade 91 needs to be assembled, especially for up to 300 mm thick plate, forgings, and heavy wall piping. Consideration of post-weld heat-treatment (PWHT) effects needs to be made. Assessment of alternate procedures for describing creep and stress-rupture for the 60 year plant life of the NGNP is also required; procedures developed by the Pressure Vessel Research Council need to be considered as well. The current allowables need to be compared to the results of the reassessment, and a recommended course of action made with respect to ASME Section II-Part D and III-NH, including identification of supplementary testing required to address the needs outlined in the High Temperature Metallic Materials Test Plan for Generation IV Nuclear Reactors.[25]

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

The NRC licensing review of the CRBRP in the 1970's and 1980's identified a number of concerns, including but not limited to weldment safety evaluation, notch weakening, and creep-fatigue evaluation[20]. The major fundamental regulatory safety need was improvement of the criteria to prevent creep cracking. The need to build confidence in the regulatory community that the designs will have adequate safety margins is critical. Compilation and storage of reports describing confirmatory program plans that were jointly developed by the NRC and the CRBRP are needed. A review of all the safety issues relevant to Subsection NH is required, including the generation of a historical record that includes a detailed description of how Subsection NH currently addresses these issues or not. Identification of additional safety concerns within NH for application to very high temperature service is needed. The review will serve as a foundation to initiate communications with the NRC on these issues, and facilitate future consultation with the NRC in improving, developing, and confirming design and fabrication procedures, strain limits and material design curves.

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

The current approaches available to define negligible creep need to be reviewed and their applicability verified for use to Grade 91 steel. Material data available in France and the U.S are needed. The methodology, data, and additional tests required to support the definition of negligible creep conditions for Grade 91 steel need to be identified. A critical comparison of ASME Section III Subsection NH and RCC-MR creep-fatigue procedures is needed. Comparisons are desired on the basis of experimental test results available from Japan, France and the U.S.; particular attention is required in the definition of safety factors and creep-fatigue damage envelope procedures. Assessment of whether or not the material data presently available in nuclear codes are thought to be sufficient and valid is required, including recommended improvements to existing procedures and definition of a test program to validate the improved procedures.

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

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

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

Task 3 and 7 include intentional parallel activity in evaluation of creep-fatigue procedures for Grade 91 (Task 3) and Alloy 617 & 230 (Task 7). The intent in this task is to compare procedures used on two different classes of alloys, and develop modified or new procedures for application to universal creep-fatigue modeling and alloy specific procedures. The task will require that formal access and use rights of the various materials databases be secured, followed by an evaluation of the state of existing data to determine if more data are necessary. Creep-fatigue criteria need to be summarized based on existing international nuclear codes, e.g. Subsection NH, RCC-MR, and Monju Design Codes. A comparison of creep-fatigue damage evaluation procedures is required, including assessment methods of strain range, initial stress and relaxation behavior, formulation of creep damage, strain rate and hold time effects, methods used in partitioning plastic, elastic, and creep strain, environment effects, and wave shape effects. A recommended testing program to generate required supplemental data needs, and model verification is needed.

3.10.7 Task 6: Graphite and Ceramic Code Development

Several new graphite and ceramic materials are being identified in new nuclear power plant designs because of high temperature operating environments[23]. Requirements must be established, and environmental effects must be considered. Support for development of ASME Code requirements for qualification of non-metallic components, including composites and ceramics required. In ASME Section III, the Project Team on Graphite Core Components is currently drafting design rules for the graphite core internal for a HTGR, and will eventually develop design rules for the composite and ceramic core internals. An independent peer review of the draft code rules, including a framework of probabilistic design approaches, fracture mechanics assessment methodologies, qualification of components, material specification/traceability, examination or inspection methods and acceptability criteria, and irradiation considerations in design rules are needed. Workshops are needed to inform ASME staff and committee members of the technical and background information required regarding the behavior and properties of brittle materials, including the effects of neutron damage on their properties. Codification of such brittle materials within ASME will require a review of the adequacy of the current data base, recommendations regarding further data needs to support the design rules, and a thorough review and endorsement of test plans and programs, both within the U.S. and internationally.

3.10.8 Task 7: NH Evaluation and Simplified Methods

With the demise of new nuclear programs in the late 1980's after the 1979 Three Mile Island incident, activity and development of elevated temperature code within ASME Section III virtually ceased. Nations other than the U.S. continued R&D activities and gained valuable experience in elevated temperature design. As such, a review and comparison of current design methods in US, Japanese and other codes are needed. Topics should include a range of design and analysis methods consisting of various levels of simplified analysis techniques: elastic analysis, limit load analysis, simplified inelastic shakedown and ratcheting analysis, and full inelastic analysis. Methods that either eliminate stress classification or provide a consistent and acceptable methodology for properly classifying stresses are desired. Code developers and researchers worldwide generally recognize that the current linear damage accumulation rule for creep-fatigue has significant shortcomings, particularly at higher temperatures and longer times. Various improvements, such as those based on ductility exhaustion and damage rate concepts, have been proposed, but none have been backed by sufficient R&D to allow their adoption as a replacement for linear damage in ASME Section III Subsection NH. Advances in thermo-mechanical fatigue life prediction methods in the last two decades have resulted from addressing mechanisms of damage nucleation and propagation; alternative creep-fatigue models should consider aging effects, formation and repeated fracture of protective oxides, internal oxidation, crack growth, and strain-based methods. Application to Grade 91 steel and Alloy 617/230/800H materials is required.

3.10.9 Task 8: Identification of Testing Needed to Validate Elevated Temperature Design Procedures for the HTGR

ASME Section III Subsection NH Code requires a combination of component testing, testing of coupons samples, and testing of basic structures with component like features in order to validate design procedures. A critical review of HTGR design features is needed, which summarizes necessary design procedures that require validation. The focus should be on major components such as reactor vessel, internal structures, piping, etc. The review should include international accomplishments in validation of design procedures and related activities for which results would contribute to the validation.

Recommendations on future tests required for validation are needed, and should address all known failure modes addressed in NH and any additional failure modes identified from the review. A brief summary of the extent and type of data currently required by ASME for acceptance of a material for a code case or adoption follows[25].

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

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

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

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

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

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

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

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

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

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

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

3.10.10 Task 9: Environmental and Neutron Fluence Effects

The ASME Code currently does not address environmental and neutron fluence effects; these effects must be addressed in any nuclear design. Inclusion in relevant design procedures must be deemed acceptable by the NRC in order to obtain construction and operating licenses. Historically, industry and the DOE have addressed these issues outside of the ASME framework; as such, the approach taken for liquid metal fast breeder reactor components may provide a basis for supplementing the rules that are now in ASME Section III Subsection NH. Supplemental design criteria, including materials of construction, ranges of expected temperatures, gas compositions, neutron fluences, capabilities for component replacement, and the like must be developed. The review should include assessment of the current U.S., European, and Japanese materials selection criteria, design codes, and conceptual designs of reactor pressure vessels and metallic core internal structures for gas-cooled reactor applications. A summary of recommendations should be provided, including an assessment of the adequacy of the U.S. Generation IV materials program to develop the materials performance characteristics needed to formulate supplemental rules, and if applicable, a supplemental testing programs to support design criteria that address environmental and neutron effects.

3.10.11 Task 10: ASME Code Rules for Intermediate Heat Exchangers

Several versions of HTGR reactors include an IHX[26] to transfer energy to a secondary plant dedicated to the production of hydrogen and/or electricity. This task will determine how and where within ASME codes and standards the IHX, safety valve, and similar components should be addressed. Many technical questions need to be addressed to determine how the function of such components affects the plants, safety, etc. The specific type of IHX has yet to be selected; all aspects of possible IHX concepts must be identified, including materials, design, fabrication, examination, testing, overpressure protection and in-service inspections that are used in the construction and operation of representative heat exchanger pressure boundaries and internals designs. Emphasis needs to be placed on (a) design criteria including methods, if any, for evaluation of cyclic life, (b) construction codes of record and designated pressure boundaries, (c) qualification of materials and fabrication techniques for the intended service, and (d) environmental effects. Recommend changes and additions to current construction codes or features of a new construction code must be identified.

3.10.12 Task 11: Flaw Assessment and Leak Before Break (LBB)

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

3.10.13 Task 12: Improved NDE Methods for Metals

Identification of new construction and in-service non-destructive examination (NDE) methods for examination of metallic materials (e.g., acoustic emission, ultrasonic) is desired. This will include (1) defining maximum acceptable flaw types and sizes based on the Load and Resistance Factor Design (LRFD) approach and material properties of candidate materials that are available, and (2) defining nondestructive examination methods needed to detect sub-critical flaws of the size and type defined in (1) in pressure equipment both during initial construction and periodically during the life of the equipment. The need for new methods is anticipated to reliably detect smaller discontinuities than those of concern for the current generation of pressure equipment. The methods will include the characterization of uncertainties in a manner that is suitable for reliability-based LRFD development for overall plant design. Some methods to be considered include but are not limited to (1) ultrasonic time-of-flight-diffraction, and (2) ultrasonic phased arrays. This task will provide technical information and background to resolve concerns and assist codes & standards committees and jurisdictional authorities in adopting improved NDE methods into ASME codes and standards.

3.10.14 Conclusions

Numerous activities of value to the DOE Generation IV Reactor Systems Program have been identified. A collaborative agreement between DOE and ASME has been established to facilitate technical review of code allowables, material databases, regulatory needs, and design procedures, expansion of ASME Section III Code, and an independent critical review is also planned. Initial efforts are also planned to consider environmental and neutron fluence effects and intermediate heat exchangers in appropriate ASME Code for nuclear applications.

4. Research Needs

Based upon the parameters discussed previously the technical areas for research and development are outlined below.

4.1 Graphite

Material property values within three primary areas (physical, thermal, and mechanical properties) will be required before any graphite can be used in NGNP. Specific material properties within each area are identified and the reasoning for obtaining this data is defined for each property.

4.1.1 Physical

Generally, physical material properties are concerned with characterizing the microstructure and the effects of microstructure on the macroscopic response of the material (i.e. dimensional changes).

4.1.1.1 Microstructural Characterization

Determination of grain size, morphology/anisotropy, and pore size/distribution within the graphite microstructure is critical to determining the macroscopic physical, thermal, and mechanical properties. These parameters must be determined before an accurate analysis of the graphite performance can be made.

In addition, a key technological deficiency is the inability to determine microstructural features within a graphite material, specifically with non-destructive techniques. This is important for determining not only the evolution in test specimen microstructures as a function of irradiation but also for determining defects within the large graphite billets. Inspecting billets without damage to ensure proper microstructural development is one of the largest problems facing any QA program for purchasing of nuclear grade graphite. The implementation of non-destructive techniques will be necessary for accurate quality assurance.

4.1.1.2 Mass and Dimensional Measurement

Dimensional change is one of the key parameters defining a nuclear grade graphite. Determination of volumetric and density changes as a function of temperature and dose will be necessary to understand critical performance measures such as turnaround, irradiation creep, and internal stresses imposed upon graphite components.

4.1.2 Thermal

Thermal material properties are critical for protecting the fuel particles during off-normal events as well as predicting thermally induced stress states within solid graphite components (i.e. reflector blocks). Degradation in thermal properties - conductivity, specific heat, and coefficient of thermal expansion (CTE) - will significantly impact the ability of the graphite to both absorb energy and transfer the heat load out of the core region during an off-normal event. Without adequate removal of the heat, fuel particle centerline temperatures will exceed the design limits resulting in unacceptable numbers of particle failures and radiation release levels. In addition, thermally induced stresses can be exacerbated between and within graphite blocks with significantly altered thermal properties. Elevated stress levels can exceed the structural strength of the graphite blocks resulting in cracking, spallation, and structural instability.

4.1.2.1 Thermal Expansion

The CTE for graphite components is critical for determining the dimensional changes that occur as a result of temperature increases. Localized external stresses can be imposed upon mechanically interlocked graphite core components as the individual pieces suffer differential expansion. Internal stresses can occur within larger graphite components if there is a temperature gradient causing differential expansion within the piece (i.e. one side has a higher temperature than the other). Finally, the thermal expansion is greatly dependent upon the graphite microstructure such as orientation/anisotropy, pore size and distribution, and crystallinity.

Irradiation damage can alter graphite CTE values significantly and changes must be quantified to determine the extent of change to the CTE. A reduction or increase in CTE can significantly affect the stresses (internally and externally) imposed upon the graphite components within a reactor core and will directly affect component lifetime. Determining the changes to the CTE as a function of irradiation dose and temperature will be a key parameter needed for reliable calculation of stress states within graphite components, volumetric changes, and irradiation creep rates.

4.1.2.2 Thermal Conductivity

The ability to conduct heat through the graphite core is critical to the passive safety of a graphite reactor. Reduction of the thermal conductivity within graphite can significantly affect the passive heat removal rate and thus the peak temperature that the core and subsequently the fuel particles will experience during off-normal events. Determining changes to the conductivity as a function of irradiation dose and temperature is important for the safety analysis for the HTGR.

4.1.2.3 Oxidation

he oxidation rate of graphite during an off-normal, air-ingress event is required to determine the effect of oxidation on the specific graphite properties as well as the entire core performance. There are two primary concerns; failure of individual graphite blocks (due to strength and thermal conductivity reduction as a result of pore formation and growth) and general core geometry configuration issues (the entire core fails due to acute oxidation and catastrophic graphite failure). Kinetic models resulting from experimental data will be required to predict weight loss in specific areas of the core. It is expected that the damage will be limited and that core geometry remains intact however some data will be required to confirm this assessment.

Additionally, based upon regulatory requirements thermal and mechanical testing of previously oxidized material will need to be performed to determine the chronic effects oxidation may have upon graphite material properties. Mechanical and thermal properties will be investigated from both acute and chronic oxidized material. The affects due to chemical and physical (pores) differences for each graphite type will be required.

4.1.2.4 Emissivity

Emissivity values for graphite must be high enough to allow heat energy to pass across the gap between the core and pressure vessel walls. Graphite emissivity is primarily a function of surface conditions for the graphite components (i.e. roughness, porosity, etc.). The as-received graphite is assumed to have high enough emissivity values to meet the heat conduction values. Confirmation that emissivity values do not degrade extensively due to oxidation and/or irradiation is needed.

4.1.2.5 Specific Heat

There are concerns that energy stored within the graphite microstructure as a consequence of irradiation damage can be released if graphite is raised to a high temperature (the Wigner energy release phenomenon). If there is an off-normal event where the graphite is undergoing air oxidation this additional stored energy, along with the heat generated from graphite oxidation, may exceed the specific heat value and produce a run away reaction. One study suggests that very high energy peaks may occur at the high operating temperatures of the NGNP.

It is generally understood that irradiation damage energy (accumulation of Frenkel defects) is only available when it is “frozen” in the microstructure at low irradiation temperatures (< 300ºC). When graphite is irradiated at temperatures higher than 300 ºC the increased point defect mobility affectively anneals out this accumulated damage within the graphite microstructure as fast as defect pairs can occur. Thus, at higher irradiation temperature it is assumed that Wigner internal energy release will be minimal but some testing to determine the stored energy levels within graphite will be required.

4.1.2.6 Electrical Resistivity/Conductivity

Electrical conductivity values for graphite are not specifically required. Electrical conductivity is used as a rapid, simple means to determine grain orientation, structure, and crystallinity of the graphite. In conjunction with optical microscopy it can be used to determine the microstructural texture of the graphite components without a great deal of sample preparation work.

4.1.3 Mechanical

The graphite single crystal is highly anisotropic due to the nature of its bonding (strong covalent bonds between the carbon atoms in the basal in the plane and weak van der Waals bonds between the basal planes). This anisotropy is transferred to the filler coke particles and also to the crystalline regions in the binder phase. Thus, the mechanical and physical properties of graphite vary within a billet due to texture introduced during forming and thermal processing. Moreover, there is statistical variability in the properties between billets within the same lots, between lots, and between batches due to variations on raw materials, formulations, and processing conditions. Since the mechanical properties are fundamental to determining the induced and applied stresses to the graphite components. Determining the resulting stresses in (and on) the components from exposure to a reactor environment is necessary to calculate the ability of the graphite to withstand the imposed loads and continued service conditions during operation. Therefore it is necessary to develop a statistical data base of the properties for a given graphite grade.

4.1.3.1 Irradiation Creep

Strain relief of induced stresses (i.e. irradiation creep) within irradiated graphite microstructures allows the graphite to withstand irradiation damage. However, graphite will continue to suffer from irradiation creep even after initial internal stresses are relieved resulting in continued dimensional changes. The resulting macroscopic behavior is similar to the changes in CTE discussed above. Thermal creep does not occur at expected NGNP operating temperatures.

Because irradiation creep can alter the underlying microstructure, it can affect the material properties in nuclear graphite during long term exposure. The graphite performance and changes to the material microstructure and properties during long term exposure must be characterized and understood to validate the design and establish accurate lifetimes for new graphite types.

Finally, since irradiation creep specimens are physically large it is relatively simple to irradiate a large number of specimens simultaneously inside an irradiation creep experiment. Therefore, while investigating irradiation creep rates all other irradiated material property values can also be determined utilizing both the creep samples and piggy-back irradiation samples within an irradiation capsule.

4.1.3.2 Elastic Constants and Stress-Stain Curve

The mechanical properties of graphite are necessary to determine the structural integrity of graphitic components. These properties are vital to determine the viability of the structural strength and integrity of the reactor core. The as-received and irradiated values are needed for whole core models which will be used for the graphite design code. This is discussed in later sections. Specific material properties required for the whole core modeling are listed in Table 6:

Table 6. Mechanical properties required for whole core modeling.

Static and dynamic elastic modulus
Shear modulus
Poisson's ratio
Creep poisson's ratio
Strength values : (σ_{flex}, $\sigma_{tensile}$, $\sigma_{compression}$)
Strain to failure
Fracture toughness : (K_{Ic}, G_{Ic}, σ_f)
Multi-axial strength[a]

[a] Multi-axial stresses are anticipated to be induced in the front reflector blocks which experience the largest dose and temperature levels. Gradients in dose and temperature within the block lead to tensile stresses on the front of the block gradually changing to compressive stress states on the back of the block which induce high multi-axial stresses resulting in potential failure within the block.

4.1.3.3 Tribology (wear/friction)

This is primarily a concern with Pebble Bed designs. The concern is that wear on the pebbles during movement can generate dust which will act as a means for transporting fission products during loss of coolant. To determine the amount of dust to be generated the tribological properties of the graphite must be determined.

4.2 High Temperature Materials

Key material issues were discussed during a facilitated meeting of program PIs and managers, vendor representatives and additional technical experts held in Salt Lake City on June 21 and 22, 2006 at the Sheraton City Center Hotel. The following sections on the RPV and High Temperature Alloys (primarily for the IHX) are condensed from the notes produced from this meeting.

Discussions underscored a key difficulty with the NGNP project (and many others). A definition of the materials design and performance requirements is needed for material research needs to be defined and for correct material selections. In other words, the material researchers want the reactor design and operating parameters determined. However, design engineers want to know (or assume) material capabilities now and delay design decisions as other pressing technical and political issues dictate. Of course, from the materials research point of view, nothing is more pressing, if an adequate material cannot be found for a critical part; or at least a codified material in the required time frame.

As such it was suggested that working groups of laboratory and industry/vendor personnel be formed to define the temperature, size, pressures, flow velocities, pressure boundaries, etc. as relevant to critical areas discussed below. It was commented that it is unclear what DOE really wants to demonstrate with the NGNP and that research can appear to be driven by continuity of work for the labs, so material selection remains unresolved. Identifying the tentative operating parameters is most important. Testing needs, including design of component test facilities is also required. Pre-conceptual designs have recently been developed.[1] The availability and ability to fabricate some major components to achieve the full 950°C gas outlet design temperature was identified as one of the primary technical factors to impact schedule and cost risk. The principal concerns pertain to the RPV and the IHX. Qualification of materials and methods to support design, ASME / ASTM code acceptance and NRC licensing activities was also identified as a significant risk factor. This is addressed in the design methodology section. It is critical that decisions be made about which and how many alloys will be used and kept for development to decrease research requirements.

4.2.1 Reactor Pressure Vessel

The AREVA and GA designs have high reactor inlet gas temperatures in the range of 500°C. The current designs for these plants expose the large vessels to this temperature during normal operating conditions. The RPV is also exposed to higher temperatures for some period of time during the postulated loss of coolant conduction cool down design basis accidents. This requires use of material with an acceptable strength and creep resistance at these temperatures. The Westinghouse PBMR design uses a lower inlet temperature that does not require the use of higher alloy material, and a more common SA-508/533 steel is used for these vessels.

Several assumptions are inherent in the following RPV issues. The new 2.25Cr-1Mo-V alloy; not the standard 2.25Cr-1Mo that is already approved in Subsection NH of Section III is of interest because vanadium is required for thick sections. The PWR industry will maintain alloy SA508/SA533 (508 is forging and 533 is plate). Material will be available any time and material data needs will be a limited set. Furthermore, the infrastructure for supply of pressure vessels (SA508/SA533) will grow and become available for application to gas reactors in parallel with program needs via PWR resurgence. However, the infrastructure for Grade 91 vessels must be developed.

Although multiple development risks must be addressed for Grade 91 steel, this material selection may be dictated by performance requirements. The data base of SA508 must be ensured to be complete, defensible and properly codified to establish a well defined design and licensing basis. The 2.25Cr-1Mo-V appears to be a less likely choice and should arguably be dropped to avoid wasted effort, however Areva is interested in this alloy.

While licensing issues are critical, they are being addressed primarily by other organization so are not mentioned specifically here. Similarly, the transportation of the vessel is a critical materials issue primarily as it affects final fabrication decision. Actual transport issues will not be addressed here.

1. Codify materials
   - Grade 91– in parts of code up to 649°C, but for NH pressure vessels up to 371°C.
     - Define insignificant creep.
     - Address heavy section products.
     - Determine weld reduction factors.
     - Define allowable excursion conditions.
   - A508 – in code case N-499-1 for general operating temperatures below 371°C
     - Define allowable excursion conditions.
     - May be pursued by Westinghouse and PBMR. Collaboration in this area should be possible.
   - 2.25Cr-1Mo-V – not code qualified at all (w/ V - needed for thick sections).
     - Define insignificant creep.
     - Address heavy section products.
     - Determine weld reduction factors.
     - Define allowable excursion conditions.

2. High Temperature Design Methodology
   - No issues because elastic.

3. Determine joining and fabrication methods
   - Weldability studies
     - Grade 91
       - One year in collaboration with AREVA; four years without.
       - Significant work has been done in other DOE projects (ex. Fossil) and collaboration can be cost and time efficient.
       - Particular challenge welding thick sections compared to others.
     - A508 not necessary.
     - 2.25Cr-1Mo-V one - two years.
       - Assumes availability of AREVA's recent welding developments in a useful time frame.
   - Establish fabrication process and details and select a vendor for RPV.
     - Grade 91 - who can make it?
     - Can sufficient ingot size be produced for such a large forging?
     - Finned tubes?
       - Not currently in production.
       - Should be easy to manufacture.
       - Fins are currently fabricated on the outside of the tube, but not on the inside.
     - Ring forging where, when, how?
     - Determine penetration limitations.

4. Characterize the environmental and aging effects
   - Grade 91
     - Why study long-term aging of Grade 91 when it is already code-approved for durations out to 300,000 hr? Will it be aged for longer than that?
   - A508
     - Is long-term aging information for A508 needed when it is only applied below 371°C?
   - Where are the data from the HTGR's that have been operated?
   - Define environment, temperature range and type of data needed.
     - Long term aging on mechanical properties.
     - Long term environmental effect.
     - Short term accident behavior.
     - Irradiation?
       - A508 done: Prototypic material, temperature, flux, and fluence irradiations will be required by NRC for all materials other than A508/A533.
       - Low-flux irradiations and PIE for any new material: 5 years for irradiation campaign + several years to make case to NRC.
   - Determine development of data time frame vs. time required to codify.

5. Develop inspection requirements and process for the vessel
   - Assume: PWR basis with special equipment development needs.
   - Common to three steels - as long as not in creep range.
   - High level requirements.

6. Testing components at model scale
   - Insulated pipes for heat transfer to H plant may be made from these steels.

7. Ensure vessel emissivity
- Verify need for emissivity testing given modified RPV temperatures.
- Determine target emissivity number. 0.7, 0.8, 0.85?
- Develop techniques suitable for the particular materials.
- Collaborate with AREVA on emissivity testing.

4.2.2 Very High Temperature Alloys

The pre-conceptual designs all recommended using a phased approach to operate the plant at lower temperature in the startup and early operation of the plant to provide more design margin for the available materials of construction. Both Inconel 617 and Haynes 230 alloys are qualified for operation at 850°C so construction of an IHX could proceed with minimal materials R&D. This approach would also provide more time for expanding high-temperature material databases and for finding suitable higher-temperature replacement materials (e.g., oxide dispersion-strengthened [ODS] alloys, ceramics, etc.).

Indirect cycles are assumed for the for high-temperature alloys. The primary coolant is He; molten salt as an intermediate loop heat transfer medium *is not* considered. Trade studies are being performed to decide if salt with be used instead of or in addition to helium. This will be a critical materials issue *if* salt is chosen. Again, licensing issues are critical but being addressed primarily by other organization so are not mentioned specifically here.

1. Codify materials
- Inconel 617
  - IHX alloy
  - Draft code case exists with allowable temperatures up to 982C, 100,000h.
  - Further develop draft case and apply to become code.
    - Additional material data needs
    - Weldment fatigue and stress-rupture factors needed
- 800H
  - Control Rods, Internals
  - On-going effort to extend temperature range above 760C.
  - Operating temperature definition may include short time, very high temperature excursions.
- Haynes 230
  - IHX alloy - lower probability.
  - Code development needs would be greater than In-617.
- Has the German code for Alloy 800HT (and possible for Alloy 617) been ratified or is it still in draft?
- Include bonding techniques (welding, diffusion bonding and brazing).

2. Develop high temperature design methodology
- Develop the equivalent to DOE NE F9-5T for guidance on inelastic design analysis methods for nuclear components.
  - Development of constitutive equations for alloy(s) for generation of isochronous curves.
  - Full finite element analysis.
  - Damage/life prediction methods - dependent upon ability to generate data for failure criteria.

- Develop creep-fatigue life prediction methodology for base and weld metals and overall weldments corresponding to the selected joining processes for IHX fabrication. Consider environmental effects as well.
- Conduct analysis for guidance and design of test parameters, geometry, loading and test facility requirements to develop a simplified methodology and produce relevant data for verification of design criteria for the complex geometry IHX.
- Develop analysis and design methods for complex 3-dimensional structures like PCHE.
- Expand to other components (pumps, valves, piping)?

3. Determine joining and fabrication methods
- IHX
  - o Address various product forms (thick/plate, sheet, foil).
    - ▪ Test different product form for creep, tensile, and aging.
      - Preliminary guidance and input to design and performance requirements.
      - Codification requirements.
    - ▪ Assess effects of grain size for thin sheet product
    - ▪ Address effect of bonding process (thermal cycle) on material properties.
  - o Physical design (shape, size, and distribution of flow channels) may have large constraint on joining process.
  - o Address material selection and performance of filler metals for brazing or similar processes.
- Core Barrel
  - o Check on difficulties of fabricating and transporting.
  - o On-site final fabrication required?
- Collaborate with other DOE programs where significant efforts have been made or are in progress on joining issues of these materials.

4. Characterize environmental and aging effects
- Consider erosion due to particle-laden (spalled oxide) gas stream.
- Environmental effect – particularly for thin sections.
- Test facilities for mechanical tests (creep-fatigue, etc.) in HTR environment.
- Define environments.
  - o Helium impurities helium on the primary side.
  - o Composition of helium on secondary side.
  - o Molten salt (composition?) on the secondary side?
  - o Temperature gradients, especially on the secondary side if molten salt is used?
- Determine actions on secondary side as well.

5. Develop inspection requirements and process
- IHX: Requirements for pre-service, In-Service Inspection (ISI), and continuous condition monitoring of the IHX will be heavily influenced by whether this is a class 1 or class 2 component. Given the very large issues and efforts for steam generator ISI and performance in LWRs, input from NRC and vendors will be essential in developing this technology.
- Determine need for defect-tolerance and flaw assessment data as part of NRC review/license application. Fracture mechanics data required?

6. Testing components at model scale
    - Establish formal international collaborative agreements to share test loop design concepts, test data, etc.
    - Design test loop.
        - Decision on secondary fluid required.
        - Determine materials test needs (temperature, pressure, flow rates, gas chemistry, duration, sample size, etc.).
    - Component development testing may be necessary before helium loop testing as part of IHX preliminary and final design process (before building a prototype IHX or before testing of a complete component).
    - Investigate feasibility and design/acquisition of 'test coupons' and sections of IHX for testing.
    - May need to look at triple walled high-temperature piping designs and materials proposed by PBMR.
    - Pumps and other components need to be examined as well.

7. Irradiate high-temp alloys for internals
    - Consider small effort to confirm He embrittlement of high Ni alloys is not an issue in core barrel and internals.
    - Control Rods
        - Fluence <0.15E21 n/cm^2/yr fast, <2.5E21 n/cm^2/yr thermal
        - Literature data primarily alloy 800 rather than 800H.
        - Determine need/parameters for in-situ irradiation creep.
        - Post-irradiation studies
            - Creep rate and rupture time: T/t limits for off-normal conditions.
            - Low-cycle fatigue: allowable number of scram events (replace control rods in event of scram?).
            - Tensile: extent of embrittlement is primary concern.

4.2.3 High Temperature Design Methodology

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

1. Alternative primary load design methods
    - Inelastic analysis method - reference stress design approach
        - Progress made developing methods and criteria equivalent to the intent of the current ASME-NH code.
        - Continue efforts to establish and obtain approval, including modifications as alternative methodology used in designing for load controlled stresses within Subsection NH.
        - Compare approach to coupon samples, structural feature-like samples, and small-scale IHX
    - Review available test conditions, materials, and test types to support any proposed design criteria modification
    - Additional testing specific to IHX material and design

2. Simplified methods and criteria
    - Evaluations of simplified analysis methods to general structures and loading
        - non-axisymmetric structures

  - non-axisymmetric loading
  - 2-3 years
- Develop unified constitutive models for inelastic design analysis, but difficult
  - without a material selection made
  - lack of raw data for Alloy 617 used to generate draft code case
- address cyclic primary loads and effects of the history of variable loading
  - additional simplified methods
  - full inelastic analysis

3. Failure models for design criteria
   - Continue efforts with creep-fatigue testing in air of Alloys 617 and 230 at ORNL and INL
   - Continue DOE-ASME collaboration (GenIV) creep-fatigue of Gr91 steel and Hastelloy XR
     - make modifications to the existing methodology for predicting creep-fatigue damage for very short term implementation into existing ASME code
     - likely be 3-5 years
   - Establish testing capabilities at very high temperature in impure He
   - Determine grain size effects

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

4. Inelastic design analysis methods
   - Modeling
     - Ashby deformation mechanism equations for Inconel 617
   - Experiments
     - Model validation
     - Model improvements – define surfaces
       - Biaxial stress space
       - Thin-walled tubular specimens, etc.
     - Develop constitutive model – utilize dedicated software
   - Incorporate into design analysis program (eg. Finite Element Models [FEM]) coupled with thermal analysis

5. Confirmatory structural tests and analyses

   Three types of structural tests, and associated analyses, were successfully employed in developing and validating methods and criteria for the liquid-metal reactor. Experience has shown that with these tests most of the key behavioral features exhibited by actual systems and components can be modeled and evaluated. The resulting data will provide a check of the most important features of the methods and criteria. These same types of tests are envisioned for the NGNP reactor.

- Basic structures
    - Axially-loaded notched bars and bars with a step change in diameter (e.g., to generate data for the SMT method)
    - Axially-loaded flat plates with holes, axial/transverse welds
    - Two-bar shakedown and ratcheting tests
    - Pressurized tubes with step changes in wall thickness, built-in ends
- Simple structures
    - Beams in bending
    - Pressurized cylindrical shells with heads
    - Thick cylinders subjected to cyclic radial thermal gradients and axial loads on internal pressure
- Component configurations
    - Nozzle to spherical shell with internal pressure
    - Nozzle to spherical shell with internal pressure and non-axisymmetric loading of the nozzle
    - Representative pieces of a compact IHX under:
        - internal pressure loading
        - cyclic internal pressure
        - internal pressure with thermal gradients
        - other relevant loadings as deemed significant during various design stages

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

6. Safety / reliability assessments

7. Resolution of identified shortcomings, issues and regulatory concerns

5. Research and Technology Plan

Work for this fiscal year (FY-07) was dramatically reduced from the original plan. Limited funds were available and the majority of those were committed to graphite research and planning. The planned workscope for the remainder of the materials program is currently uncertain, but will be established sometime during FY-08.

5.1 Graphite

The scientific and engineering techniques described within this section encompass all the anticipated tests required to validate and qualify nuclear grade graphite for use within NGNP. The plan presented here represents the information needed for full operational license of a prismatic NGNP reactor design. The test matrixes could be limited to reduce the scope of the testing in support of a limited licensing strategy (i.e. demonstration plant license) if necessary to meet NGNP deployment schedules. For a PB HGTR design additional testing will be required to support the longer design life (i.e. high dose levels) of the front facing reflector blocks. In addition, the slightly lower inlet temperature (from prismatic design levels) may require changes to the test matrix parameters to ensure the tests bounds the operating envelope. The high dose irradiation experiment are included here for completeness but will be separated from prismatic cost baseline. Ultimately however, data from all tests will be required for commercialization of the HTGR technology in order to use a new graphite type within a HTGR.

Constitutive relationships and model development using the data acquired from this R&D program will be required for codification of the graphite. The appropriate role of model development and the extent of development is discussed as it pertains to perceived ASME and regulatory requirements.

5.1.1 Graphite Experimental Data

Since many graphite components will be exposed to the full neutron flux generated in the NGNP core any changes to pertinent material properties must be determined to understand the long-term behavior of the graphite in reactor. As a consequence, an extensive non-irradiated and irradiated material characterization program is planned and currently in progress.

The non-irradiated characterization program focuses on developing a statistically valid material database for each of the graphite types selected for irradiation testing. This will establish baseline values for material properties that can be used to determine the quantitative changes during irradiation. A large irradiated specimen population will then be exposed to the expected NGNP reactor environment (temperature and dose ranges). As stated previously this experiment is expected to yield pertinent information for all irradiated material property values necessary to qualify a new nuclear grade graphite. However, doses and temperatures experienced within the prismatic reactor design are expected to be different than the pebble bed design. Generally, the prismatic reactor operates at slightly higher temperatures while the pebble bed is expected to operate for long periods of time.

Specific descriptions of test sample preparation, non-irradiated material characterization, irradiation experiment descriptions, and material characterization comprising the experimental data needs are outlined below.

5.1.1.1 Test sample preparation

Before any material characterization testing (non-irradiated or irradiated) can be carried out an optimal method of machining the graphite samples from the bulk material must be developed to ensure representative samples can be obtained. The NGNP program has developed an extensive sample cutting and sectioning plan to guarantee not only statistically valid sample numbers but also spatial validity so that microstructural changes within the bulk material (i.e. billet) affecting material property changes are well characterized.[27] Particular attention has been given to the traceability of each specimen to its spatial location and orientation within a billet.

These graphite billet cutting plans were developed to promote a more complete or finer resolution material property “mapping” of material property changes within the billets. This was achieved by maximizing the number of test specimens that could be obtained from each billet. However, in order to provide statistically significant results from the various test methods a minimum of four samples are needed from each location/orientation within the same billet (per ASTM methodologies). Since this is physically impossible it is assumed that the billets have some level of symmetry in material properties throughout the entire structure which allows samples from different sections of the billet to effectively be “similar” with respect to material properties, Figure 13. Using samples from similar locations within each billet section will yield enough samples to provide for statistical validity within a single billet.

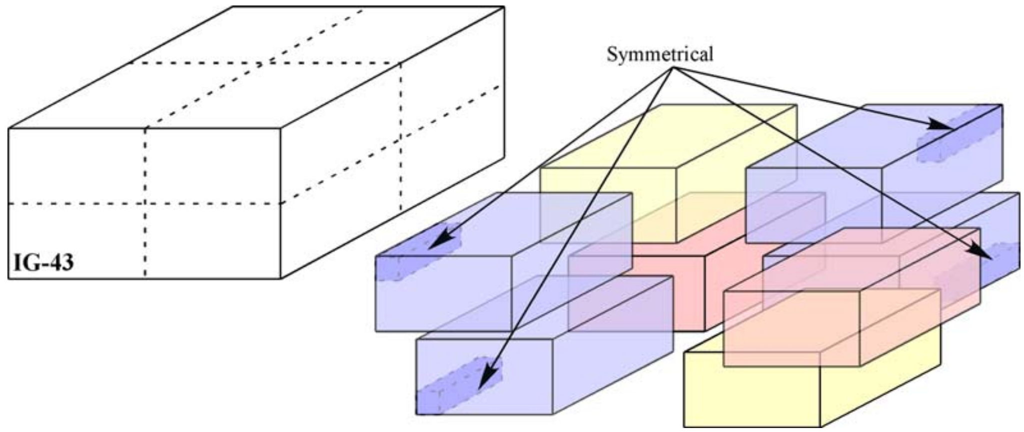


Figure 13. Example of symmetrical quadrant sections within a graphite billet

To facilitate machining the billets are cut into successively smaller sections designated as “slabs”. Each slab is sized to accommodate the proper grain orientation within the test specimens. All test specimen blanks machined for a given slab will have the same grain orientation (ag or wg) as the slab. The slabs will be further sectioned into sub-slabs to allow the rectangular test specimen blanks to be machined to the correct size. Finally, a tracking methodology is used that will account for every specimen machined from a graphite billet. A unique identification number will be assigned to each test specimen providing the exact location and orientation of the sample within the graphite billet. This identification system is based upon the cutting methodology to provide an easy and concise method for identifying the different samples. This methodology (and the corresponding assumptions) will be used for producing all test specimens for material characterization.

Since a large portion of the testing is with irradiated samples a minimum specimen size must also be considered (for volume restrictions within a materials test reactor). Each material test will depend upon the specific graphite’s grain size since ASTM test standards call for specimen sizes to have cross-sectional diameters of at least 5X the grain size across the stressed gauge section of the sample in order to provide representative and repeatable testing results. However, most graphite material tests use a minimum of 10X of the maximum grain size for the cross-sectional diameter across the gauge section. Thus, for a graphite with a 1mm grain size the minimum diameter in the test gauge area for a typical tensile specimen must be ~10mm. Fabricating smaller test specimens is not allowed since they will not provide representative material properties across such a few number of grains in the material microstructure. Other graphite types which have smaller grain sizes may use smaller gauge sections, again dependent upon the material’s grain size.

5.1.1.2 Non-irradiated Material Testing

Baseline, “as-received” material properties for each graphite type is needed to establish accurate thermal and mechanical response of the core. Since material properties are expected to vary throughout the rather large billets or blocks of graphite mapping the magnitude and spacial positions of variability is important in order to determine an individual component’s material properties. In order to enable credible core designs and to support the ongoing development of a probabilistic graphite design methodology the maximum variability within graphite components must be well characterized. For example, if the compressive strength is reduced significantly near the edges of the billets a graphite support column fabricated from a position near the edge may not possess sufficient strength to support the weight of the core blocks above. Thus, determining where the strength begins to be reduced within the larger billet and by how much is important for determining where to fabricate an individual graphite component to meet specific design requirements within the core.

A complicating factor is the variability not only within the individual billets but also from billet to billet and finally lot to lot. These within-billet, intra-billets, and lot-to-lot variations of the graphite must be accounted for in a statistical manner in order to determine the maximum range of material property variations expected for components machined from an "average" billet. Such a statistical material property database can only be obtained from extensive non-irradiation characterization of samples taken within billets and compared to sample between different billets and different graphite lots.

Physical, thermal, and mechanical property testing of multiple graphite samples from a large billet sample matrix is necessary for determining the proper statistical ranges of values. The appropriate sample matrix size, sample geometry, and sample dimensions as described previously will be important to establish statistical validity. All material tests to be used to build this material property database are described later in this section. Once the non-irradiated, "as-received" material properties have been determined the changes due to irradiation will be determined from post-irradiation examination and characterization studies on representative graphite types.

5.1.1.3 Irradiation experiments

To determine the graphite response under irradiation a series of irradiation experiments will be required. After the graphite sample matrix is chosen the irradiation conditions are determined based upon the expected reactor conditions as described in Section 3. Post irradiation examination and characterization entails performing the same physical, thermal, and mechanical tests as described for the non-irradiated materials, only this time with irradiated graphite samples.

As discussed previously, thermal creep of graphite is not expected at the temperatures experienced in the reactor core (< 1100ºC) so that determining the non-irradiation creep rate is not required. However, irradiation induced creep in graphite is expected at these temperatures and will play an important role in the irradiated behavior of the graphite during reactor service. Thus, irradiation creep experiments will form a significant part of all irradiation studies for graphite types in NGNP.

5.1.1.3.1 AGC Experiment

The AGC experiment is designed to provide irradiation creep rates for moderate doses and higher temperatures of leading graphite types that will be used in the NGNP reactor design. The experiment is designed to provide not only static irradiation material property changes but also determine irradiation creep parameters for actively stressed (i.e. compressively loaded) specimens during exposure to a neutron flux. The temperature and dose regimes covered by the AGC experiment are illustrated in Figure 14.

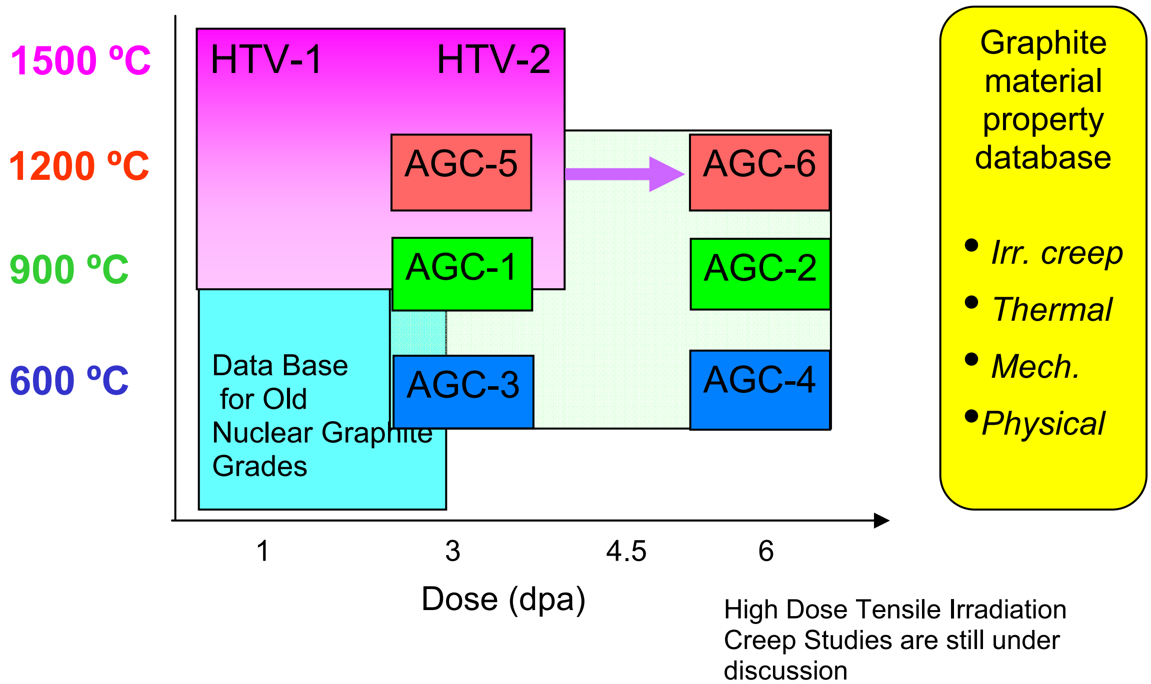


Figure 14. Schematic diagram illustrating dose and temperature ranges for AGC and HTV experiments.

As shown, the dose and temperature is bounding for the prismatic reactor design (dpa ~ 5-6 at 1100ºC) for both fuel and front facing reflector blocks. This dose limit is intentionally below the expected point of turnaround for the current NGNP graphite types at normal NGNP operating temperatures. Only AGC-6 experiment (6-7 dpa at 1200ºC) may approach expected turnaround limits for the selected NGNP graphite types.

To determine when (and if) turnaround will occur for the selected NGNP graphite during exposure in the AGC-6 capsule an additional experiment – HTV (High Temperature Vessel) has been postulated. The HTV1 & 2 capsules are simple "drop-in" capsules with the exposure parameters illustrated in Figure 14. As shown these experiments are operated at much higher temperatures (inducing faster turnaround) but at lower doses. As this is a simple dimensional change experiment to determine when turnaround may occur the graphite is not loaded during irradiation. A detailed description of the HTV 1&2 experiment is presented in ORNL-GEN4/LTR-06-019.[28]

Since the prismatic NGNP design estimates that reflector blocks can be replaced well before turnaround should occur at normal operating temperature (<5-6 dpa) and fuel blocks are replaced after only 2 cycles (<4-5 dpa) the AGC experiment should fully bound the graphite experience within a prismatic design. The dpa levels achieved in the AGC experiment will not fully bound the pebble bed NGNP design for high dose reflector blocks (see below). However, it will certainly provide preliminary data for the first 20-25% of the expected dpa levels for these graphite components.

Graphite components located farther from the core region will have correspondingly less dose and operate at much lower temperatures than the fuel region. As a consequence, turnaround and irradiation creep levels for these peripheral graphite components will be at significantly longer times and lower rates and should be fully bounded by the AGC data.

All six capsules comprising the AGC experiment will be irradiated in the South Flux trap of the Advanced Test Reactor (ATR). This will require sequential irradiation campaigns for each capsule. Each capsule will contain approximately 400 specimens - 90 large irradiation creep specimen pairs and over 300 “piggy-back” specimens. The smaller and non-stressed “piggy back” specimens are located in the center channel in the experiment. Other “piggy backs” are used in the lower non-stressed creep specimen channels as offset specimens to account for the slight asymmetry in the ATR flux profile. Each of the six columns contains stressed and non-stressed specimens. The symmetry of the flux buckling is used to irradiate each stressed and non-stressed specimens at the same fluence level. There are seven non-stressed specimens below core centerline and eight stressed specimens above core centerline. The creep measurement is made on the dimensional difference between the stressed and non-stressed specimens irradiated at the same fluence and temperature. Figure 15 shows the arrangement of the graphite specimens in the experiment.

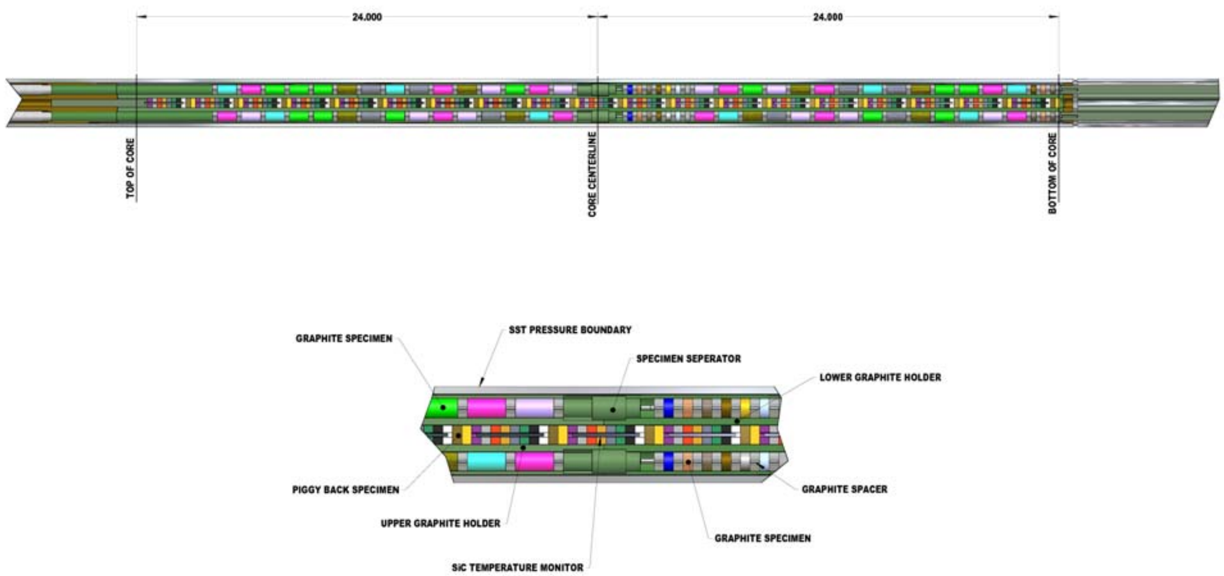


Figure 15. Internal configuration of AGC experiment.

All specimens are maintained at a constant temperature during exposure times of between 6 and 18 months depending upon required dose, Figure 14. Post Irradiation Examination (PIE) characterization is projected to take approximately 14-18 months for each capsule even though irradiated graphite samples can be contact handled after a short decay period (~ 6 months).

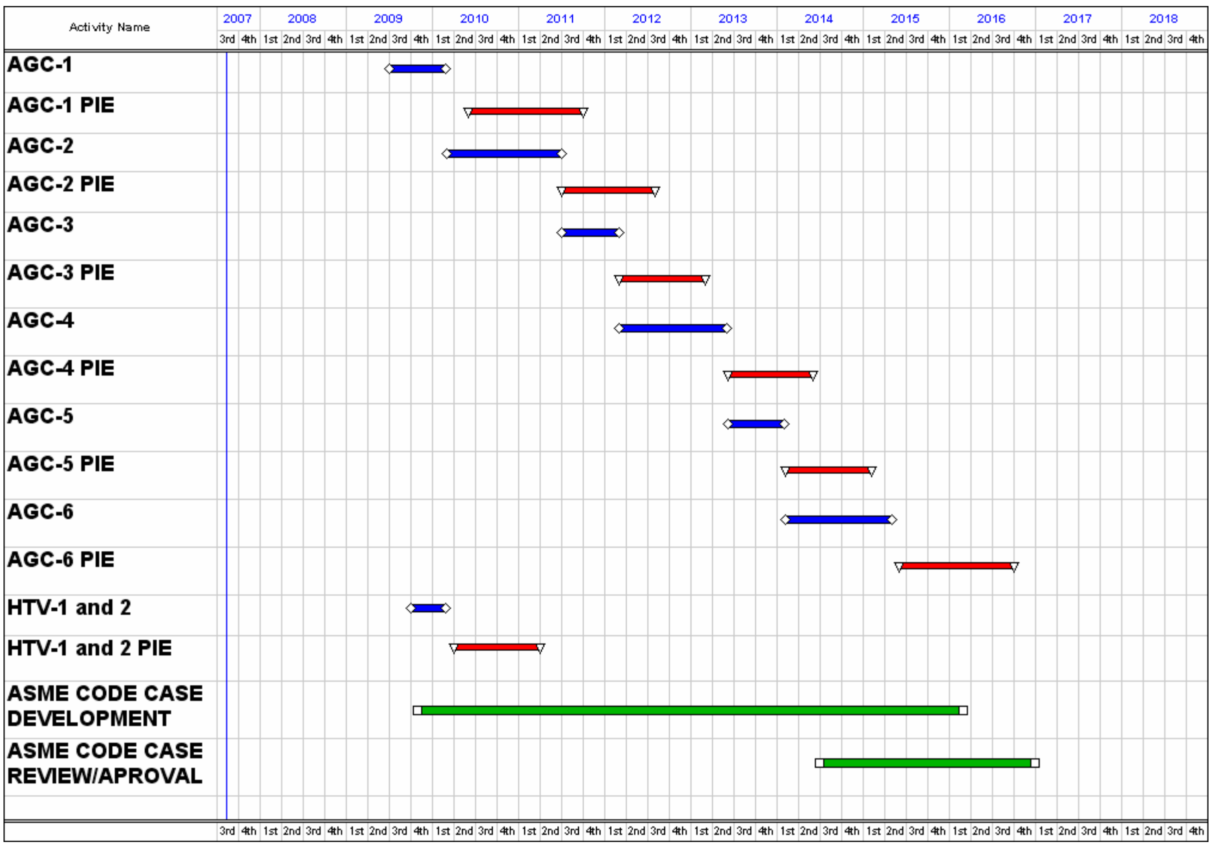


Figure 16. Estimated irradiation and PIE schedule for AGC experiment.

5.1.1.3.2 High Dose Irradiation Experiments

The high dose experiment is designed to provide irradiation exposure for very high doses and moderate temperatures. As noted previously the pebble bed design expects the facing reflector blocks (inner and outer reflector) to operate at much longer times and thus withstand a maximum of irradiation damage before the core is shutdown, de-fueled, and the blocks replaced. Current expectations are for the reflector blocks to operate approximately 20 – 25 years before replacement. At the higher end of the dose range noted previously for a NGNP pebble bed design this can correspond to as much as 25 dpa before change-out. While this appears to be a very large dose the expected temperature ranges are lower than for a prismatic design resulting in longer turnaround times and slower irradiation creep rate.

If a pebble bed design is selected the graphite material property changes for the higher expected dose levels will be required. A high dose creep experiment exposing selected graphite to much longer dose levels at moderate temperatures has been tentatively planned in support of this design selection. The temperature and dose regimes covered by this high dose experiment are illustrated in Figure 17.

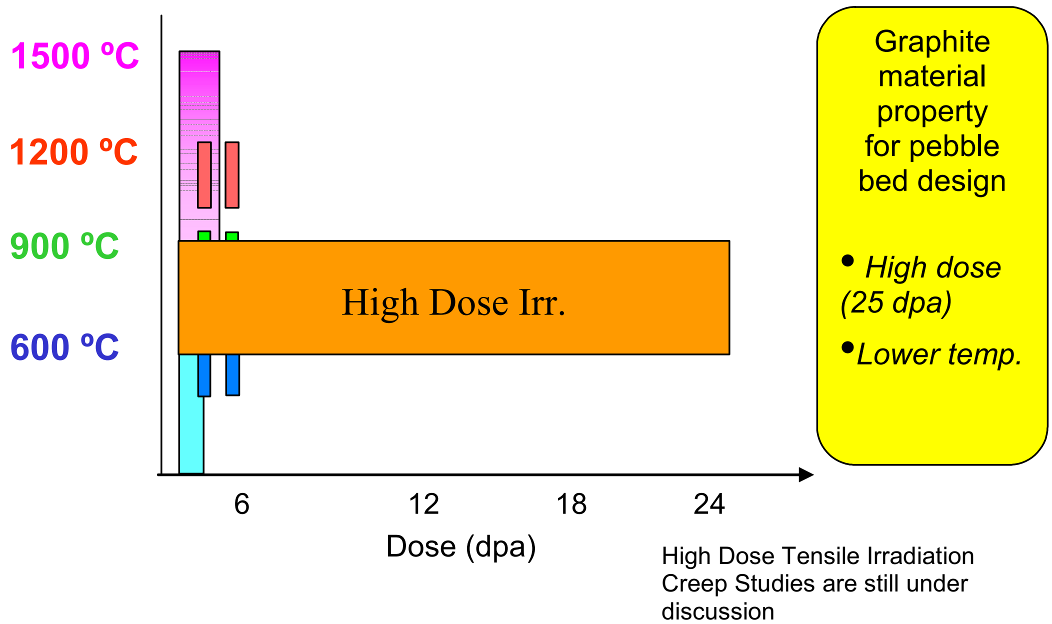


Figure 17. Schematic diagram illustrating dose and temperature ranges for high dose irradiation experiments.

As shown these samples will be exposed to dose levels considerably higher than expected turnaround dose levels even at the moderate exposure temperatures. In addition, the irradiation creep samples will be tensile loaded during exposure to assure optimal creep rates, see section 5.1.4 below. However, since these dose levels are expected after 25 years of service the high dose experiments are not needed for initial material property ranges specifically required for reactor licensing and startup operations. Results stemming from this experiment can (and most likely will) be delayed for a few years until after reactor startup since data from the AGC experiments will provide sufficient data to support operations for at least 6-7 FPY operation of the reactor. The data from the high dose experiment will be required if any graphite reflector block will be exposed to doses higher than 6-7 dpa.

5.1.1.4 Material Characterization

The following sections describe the material tests anticipated for all non-irradiated and irradiated examination and characterization studies. These material tests will be applied to both irradiated and "as-received" graphite samples to ascertain the changes to the material properties resulting from a neutron radiation field. Where possible, ASTM standard test methods will be employed. If no test standard exists some additional activity may be required to develop a test standard.

5.1.1.4.1 Physical testing

5.1.1.4.1.1 Microstructure Characterization

Optical microscopy measurements of grain size, morphology/anisotropy, and pore size/distribution will be used to determine the graphite microstructure. Adjacent optical samples will necessarily be taken as close to test specimens (with similar orientation) as possible from within the graphite billet and the microstructure will be inferred to the test sample microstructures.

In conjunction with the optical microscopy, non-destructive X-ray tomography (CT) will be investigated for its ability to ascertain the microstructure within individual test samples. X-ray techniques will allow samples to be analyzed before being tested using additional techniques (both non-destructive and destructive). Image analysis techniques will be used to enhance information such as internal pore sizes and pore structure throughout the microstructure.

Changes to the microstructure due to thermal, irradiation, and stress history will be compared to the original microstructure. Microstructural evolution and modifications as a result of exposure to reactor environments can then be ascertained.

Non-destructive methods for large scale analysis (i.e. manufacturing, ISI methods, etc.) will need to be developed. CT methods may be possible but have limited resolutions for these large component sizes. ultrasonic testing (UT), electrical resistivity/conductivity, impact echo, or other techniques will have to be developed in order to meet both in-service inspection requirements as well as billet characterization for manufacturing QA.

5.1.1.4.1.2 Mass and Dimensional Measurement

Precision measurements of all irradiation test samples will allow macroscopic dimensional changes and pore formation estimates to be determined. Volumetric and density changes will be calculated and compared to pre-irradiation values for each test sample.

5.1.1.4.2 Thermal testing

All thermal (and electrical) samples will be button samples having dimensions equal to or less than 12mm dia. x 6mm thickness. These small sample sizes allow for many specimens to be made available for both irradiated and non-irradiated testing. In addition, the small size also allows thermal samples to be machined from the ends of mechanical test specimens if needed. This assures spacial uniformity of measurements of relevant thermal, physical, and mechanical material properties within the graphite billet characterization. Additionally, "re-using" the same samples allows for larger sample batches within the irradiation test trains.

5.1.1.4.2.1 Thermal Expansion & Conductivity

Thermal expansion and conductivity values will be obtained from graphite button samples within a laser flash diffusivity analyzer to temperatures of 1600 ºC (off-normal maximum temperature). Non-irradiated and irradiated button samples will be prepared for testing at all temperature ranges of interest.

5.1.1.4.2.2 Oxidation

Currently there are no approved ASTM test methods for measuring the oxidation rate of graphite. The NGNP program is assisting in the development of this test standard. After development, the test will be used to ascertain the oxidation rate of selected graphite types for a variety of operating conditions. Results will be used to develop kinetic models to predict weight loss in specific areas of the core.

Additionally, based upon regulatory requirements thermal and mechanical testing of previously oxidized material will need to be performed to determine the effects oxidation may have upon graphite material properties. Core configuration issue as a result of air ingress—establishing that whatever mitigation techniques are selected for this event in NGNP an assessment that the damage is limited and that core geometry remains intact is important. In addition, mechanical and thermal properties will be investigated from both acute and chronic oxidized material.

5.1.1.4.2.3 Emissivity

Limited confirmatory measurements of emissivity values for graphite will be measured using standard techniques (i.e. infrared based, etc.). NRC PIRT requirements will demand some comparative studies to determine any changes in emissivity resulting from oxidation and/or irradiation.

5.1.1.4.2.4 Specific Heat

All thermal specimens will be subjected to analysis via Differential Scanning Calorimetry (DSC) to determine the specific heat for individual samples. Changes to the specific heat due to oxidation and/or irradiation will be compared to as-received values. In addition, previously irradiated samples from AGC capsules will be monitored to ascertain the potential reduction in specific heat due to the release of high temperature Wigner energy. These will be limited confirmatory studies to ascertain the potential for Wigner energy storage at the lower NGNP irradiation temperatures.

5.1.1.4.2.5 Electrical Resistivity/Conductivity

Electrical conductivity/resistivity values will be measured through sample button resistivity measurements. Microstructural characteristics will be compared to optical and X-ray tomography results. These tests will be performed as possible based upon the material geometry and size.

5.1.1.4.3 Mechanical Testing

Mechanical testing is the most extensive and complex part of the graphite test program. Strength, irradiation creep, fracture toughness, and multi-axial testing procedures utilize complex sample geometries, complicated testing techniques, and take a long time to perform. Therefore, the techniques and plans outlined for these mechanical tests, such as the irradiation creep tests, require careful consideration.

5.1.1.4.3.1 Irradiation Creep

An extensive irradiation creep program is needed to characterize graphite creep response as part of a larger irradiated materials characterization program. A large sample population (both irradiation creep and piggy-back specimens) will need to be exposed to the expected NGNP prismatic reactor design in the Advanced Graphite Capsule (AGC) experiment. If property changes within graphite for higher doses are required then a second irradiation experiment, high dose irradiation experiment, will be implemented.

Generally, at doses below turnaround (0~6 dpa for NGNP graphite grades) in the normal operating temperature regime expected for NGNP (~1000-1200 ºC) both compressive and tensile irradiation creep rates are similar. As a consequence, conducting irradiation creep with a compressive load should yield the same response as in a tensile stress. This assumption is true until turnaround occurs. Since turnaround is a function of both temperature and dose (dpa) those graphite types exposed to higher temperatures will experience turnaround at correspondingly lower doses.

After turnaround, graphite loaded in tension enters into a non-linear (tertiary) creep regime where the creep rate is significantly increased (c-axis growth and pore formation). Tensile stresses either promote or at the very least allow unhindered strain relief during irradiation providing a “worst-case” creep rate for the graphite types exposed to higher doses, see Figure 18. Compressive loads, after turnaround, will tend to retard the creep rate and effectively delay the tertiary creep regime. Therefore, once turnaround has been achieved graphite samples should be in a tensile stress state in order to determine the fastest rate of irradiation creep possible within the graphite.

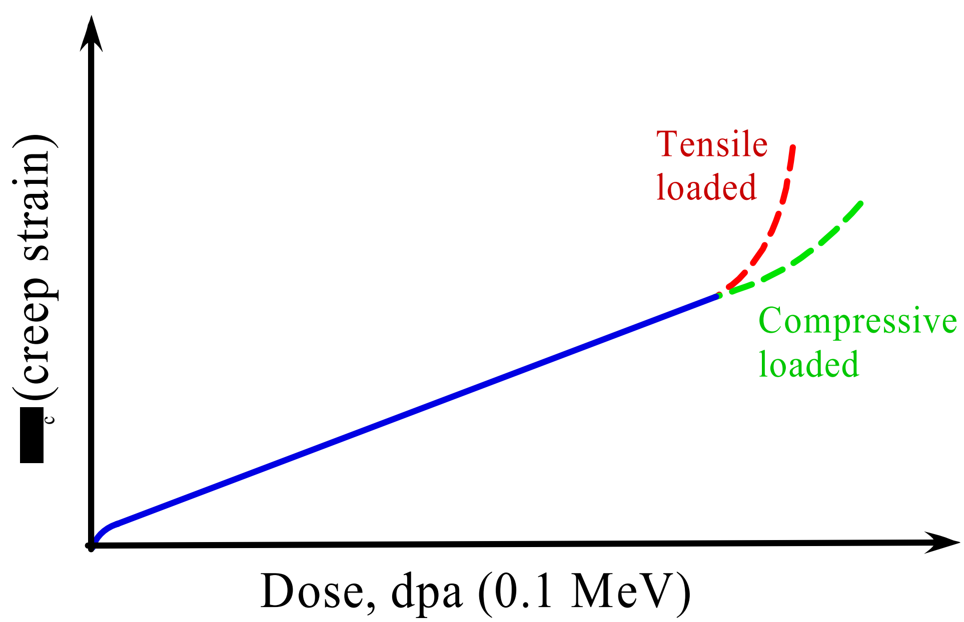


Figure 18. Schematic illustration of tensile and compressive loading on tertiary creep response of graphite.

5.1.1.4.3.1.1 AGC Experiment

The AGC experiment is designed to provide irradiation creep rates for moderate doses and higher temperatures of leading graphite types that will be used in the NGNP reactor design. The experiment is designed to induce irradiation creep within the secondary regime thus allowing the graphite to be compressively loaded during irradiation which simplifies the experiments considerably. Static compressive loads of 14.5 – 20 MPa (2-3 Ksi) are applied to the graphite during irradiation. The temperature and dose regimes covered by the AGC experiment are illustrated in Figure 14.

As shown, the dose is intentionally below (or possibly at) the point of turnaround within the graphite at the normal NGNP operating temperatures. Only AGC-6 experiment (6-7 dpa at 1200ºC) will approach expected turnaround limits for current NGNP graphite types. Since the AGC experiments are a comparison measurement between stressed and unstressed irradiated specimens if turnaround were to occur during AGC-6 exposure the creep rate results would be affected by the compressive loading state of the graphite.

Results from the High Temperature Vessel (HTV) experiment will provide both turnaround and high temperature irradiation data for all selected graphite types. Turnaround data from these experiments will be used to adjust the exposure, loading, and temperature limits for AGC-6 to extrapolate as much accurate information from it as possible.

5.1.1.4.3.1.2 High Dose Irradiation Creep

As noted previously the pebble bed design expects the facing reflector blocks (inner and outer reflector) to operate at much longer times and thus withstand a maximum of irradiation damage before the core is shutdown, de-fueled, and the blocks replaced. Current expectations are for the reflector blocks to operate approximately 20 – 25 years before replacement. At the higher end of the dose range noted previously for a NGNP pebble bed design this can correspond to as much as 25 dpa before change out of the reflector blocks. While this appears to be a very large dose the expected temperature ranges are lower than for a prismatic design resulting in longer turnaround times and slower irradiation creep rate.

If a pebble bed design is selected the creep rate and resultant strain from these higher doses must be determined for accurate lifetime predictions. This will require an extensive design development program to determine an optimal tensile loading configuration that can withstand long term exposure (i.e. 4 years within ATR or 2.5 years within High Flux Isotope Reactor [HFIR]). In addition, sample size, geometry, and matrix size, will need to be considered to determine the most advantageous Material Test Reactor (MTR) to use for this experiment.

One benefit is that only one graphite type will be required for these tests since the NGNP pebble bed design is currently interested in only a single graphite type. Thus, the sample matrix can be significantly reduced allowing multiple MTRs to be considered. However, similar to the AGC experiment the test temperatures, fluences, and tensile loads must be constant during the test.

5.1.1.4.3.2 Elastic Constants and Strength Testing

Standard strength testing techniques using stress-strain (σ-ε) curve relationships will provide the bulk of the mechanical material properties. Extensive testing programs for both non-irradiated and irradiated graphite samples will be necessary to (1) prove consistency between billets and lots of graphite, (2) provide baseline material property data, and (3) quantitatively demonstrate the material property changes as a result of exposure to a HTGR reactor environment.

ASTM test standards call for specimen sizes to have cross-sectional diameters of at least 5X the grain size across the stressed gauge section of the sample in order to provide representative and repeatable testing results. Traditional practices tend to use a minimum of 10X of the maximum grain size for the cross-sectional diameter of the gauge section. Thus, for typical large grained graphite (i.e. NBG-18) with a maximum grain size of 1.6 mm the test gauge section will need to be at least 16mm across to provide accurate mechanical property values. Most other graphite types have considerably smaller grain sizes and may use smaller gauge sections.

This imposes a minimum sample size in many cases which may be too large for most MTRs. The ATR facility can and will be used to accommodate the larger sized specimens but multiple reactors are anticipated to meet all the irradiation needs. This may force the use of much smaller specimens that are not included in standard ASTM testing methods. A program to develop miniature test specimens for irradiation testing will need to be developed for graphite. This is similar to the on-going miniature sample development for irradiated metal samples. In either case, careful determination of the optimal sample size is required for each mechanical test before irradiation.

Specialized grips are required for the testing of graphite as specified in the test standards. This is especially true for miniature test specimens that will require specialized fixtures to accommodate the small size. These specialized fixtures must be identified and developed prior to mechanical testing activities.

Finally, some elastic constants will be obtained using non-destructive UT methods. Standard testing procedures will be used to obtain dynamic elastic modulus values (tensile and compression) for specific samples.

Table 7. Mechanical properties and test standards.

Property	Test Standard
Static and dynamic elastic modulus	ASTM C747-05: Standard Test Method of Elasticity and Fundamental Frequencies of Carbon and Graphite Materials by Sonic Resonance
Poisson's ratio	ASTM C747-05: Standard Test Method of Elasticity and Fundamental Frequencies of Carbon and Graphite Materials by Sonic Resonance
Creep poisson's ratio	
Strength values σ_{flex}	ASTM C1161-02c: Standard Test Method for Flextural strength of Advanced Ceramics at Ambient Temperature
$\sigma_{tensile}$	ASTM C749-02: Standard Test Method for Tensile Stress-Strain of Carbon and Graphite
$\sigma_{compression}$	ASTM C695-05: Standard Test Method for Compressive Strength of Carbon and Graphite
Strain to failure	ASTM C565-93: Standard Test Method for Tension testing of Carbon and Graphite Mechanical Materials
Fracture toughness : (KIc, GIc, σ_f)	Under development
Multi-axial failure criteria	Under development

5.1.1.4.3.3 Tribology (wear/friction)

Standard "pin-on-wheel" wear testing procedures will be used to determine wear, friction, and dust generation values for selected grades of graphite. Additionally, previously irradiated and oxidized graphite will be subjected to similar tests to determine any changes. These will be limited studies focused on those graphite types of interest to pebble bed designs (i.e. NBG-18).

5.1.2 Multi-scale Model Development

Models are required to allow the designer to assess the condition of graphite components and core structure design margins at any point in the lifetime of the reactor. The models are needed to describe interactions between graphite components, specifically, the behavior of the stack of graphite bricks making up the core moderator and reflector. Specific models should be able to calculate external loads imposed upon the components, internal stresses resulting from radiation and temperature induced dimensional changes, movement of components (i.e. dimensional clearances for control rod insertion), and estimates of residual strength both with and without environmental attack (i.e. air-ingress during off-normal event).

Modeling the behavior of a graphite core is complex and will require some fundamental understanding of the graphite physical, thermal, and mechanical behavior as a function of irradiation temperature and neutron fluence. However, the primary objective of these models are to provide the ability to calculate in-service stresses and strains in graphite components and estimate the structural integrity of the core as a whole. Thus, understanding the fundamental and mechanistic material behavior during operation should be limited to its applicability in understanding the whole core response both during normal operation and during off-normal events (i.e. predict seismic behavior of the core). A physics based understanding of microstructural damage and its effects of materials structure and properties will provide an initial start to estimating the amount of changes to a graphite component but the degree of change is unique to the specific nuclear graphite grade used and these fundamental principles must be supplemented with actual experimental property data to provide a complete analysis of the core behavior.

For example, the existence of temperature and flux gradients within the core and individual components will generate differential changes in dimensions, and hence stress. Such stresses will creep out (relax) at the expected temperature and fluence levels experienced during normal operation. In addition, stresses arising due to thermal gradients will also creep out during operation, but will reappear in the opposite sense when the core cools during reactor shutdown. In order to model the core and component stresses during operation (and cool down) the change in properties of the graphite as a function of temperature and neutron dose must be known. Since the point to point flux and temperature data will not be available for all combinations of dose and temperature for the required properties, behavioral models are required to estimate the stress states for all components throughout the core. Thus, the whole core models must use a combination of experimentally derived material properties underpinned by an understanding of the fundamental physics to account for all variations possible within the graphite components of the core. Consequently, a major goal is the development and rigorous validation of multi-scale models for the behavior of graphite, core components and whole graphite cores for use in licensing and continued operational safety arguments.

5.1.2.1 Whole Graphite Core and Component Behavior Models

FEM is required to define the core condition at all times during core life. Such models will take core-physics and thermo-hydraulics inputs for point dose and temperature values and apply graphite material behavior models to calculate the changes in properties with neutron dose, temperature, and oxidative weight loss. Core and component scale models will allow designers to predict core and core block (e.g., reflector or fuel element) dimensional distortion, component stresses, residual strength, and probability of failure during normal or off-normal conditions.

Finite element based codes such as COMSOL™ and ABAQUAS™ offer platforms upon which the desired whole core/component behavioral models may be assembled. It is anticipated that reactor vendor's will have their own custom "bespoke" codes to describe and predict the behavior of the core within their particular design for NGNP. However, rigorous, independent validation of these whole core-scale models will be required to ensure the safety envelope of the core during normal and off-normal operating conditions. The NGNP R&D program will need to provide this independent validation capability to predict and assess the behavior and condition if the NGNP core and core components.

Finally, the development and utilization of such codes is an integral part of the design process and is recognized as such by the ASME graphite core components design code which is currently under preparation by a sub group of ASME B&PV Sect III (nuclear). The sub-group is currently bench-marking core component stress models against a standard set of problems (data sets). Additional validating data for the developed models will come from large multiaxial load specimen testing and ultimately from full-scale core components tests.

5.1.2.2 Macro-scale Materials Behavior Models

Materials behavior models are needed to predict the effects of temperature, neutron dose, oxidative weight-loss on key physical and mechanical properties. The material behavior results from these models are validated through an extensive program of non-irradiated and irradiation characterization experiments. The properties of interest include:

- Coefficient of thermal expansion & thermal conductivity, specific heat
- Strength (tensile, compressive, flexural)
- Fracture Behavior
- Elastic constants (Young's modulus, shear modulus, Poisson's ratio)
- Creep coefficient(s)

The material property models must also take into account the interaction of effects, e.g., neutron damage and weight loss, and the interdependency of effects, e.g, effects of stressed dimensional change (creep) on the physical properties of graphite. Materials models must be physically based, i.e., based on the materials structural changes, and should incorporate structural damage models and existing physics-based models (e.g., phonon conduction). Existing material property models (in some cases empirically derived models) must be evaluated and new or improved materials behavior models developed. Material property values needed for validation of these models will be obtained from the experimental characterization research as described above (i.e. AGC experiment, non-irradiated property characterization, and possibly high dose experiments).

Particular emphasis will be placed on this aspect of the multi-scale modeling as it is most directly applicable to the NGNP R&D program. Individual vendor designs are expected to significantly influence the whole core-scale modeling efforts and as such the majority of the development effort for whole core-scale models is expected to resided with the vendors. However, as illustrated above the material property models necessary for predicting graphite component and core behavior will be essential to developing and validating the whole core-scale models.

5.1.2.3 Micro/Nano-scale Models

Nano- and micro-scale modeling provides a fundamental understanding of material behavior. *Ab-Initio* models of the atomistic phenomena occurring on irradiation will allow prediction of the displacement damage that can occur and may shed light on the crystal deformation modes. Simulations (e.g. Density Function Theory) of defect structures for relevant combinations of dose and temperature can provide the basis of determining crystal strains. Understanding the physical interactions of the graphite crystallites, and the inherent porosity within and around the crystallites, are key to building microstructural models for the behavior of polycrystalline graphite. Similarly, the deformation processes that occur with in the crystallites when graphite is subjected to stress, either externally applied or those that develop within the graphite due to dose and temperature gradients, must be understood and modeled.

Surprisingly, after ~60 years of graphite use in reactors, the microstructural mechanism of irradiation creep and crystal deformation are still being questioned and are not fully elucidated. Recent fundamental studied by Heggie et al (University of Sussex Group, UK) have suggested displacement damage structures previously considered improbable are indeed energetically favorable, indicating the need for further study. Crystallite damage observations using TEM, coupled with SEM and x-ray tomography studies of irradiated graphite will provide mechanistic data for structural models.

As indicated above development of nano- and micro-scale models will underpin the macro-scale materials property models, as well as providing valuable input for experimental validation requirements. However, fundamental studies and micro-scale modeling should be supportive of the material property models to enable a basic understanding of the mechanisms driving the material property changes. While important, less direct emphasis will be placed on complete development of these nano- and micro-scale models than on the material property models discussed previously. Some support will be required in order to fully understand the underlying principles that induce changes to the material properties but the majority of the work will be left to long-range research funding sources.

5.2 High Temperature Materials

5.2.1 Program direction

The following issues are expected to have an impact on the direction of the program.

1. The selection of the steel to be used for the RPV will be made based primarily on the reactor design and vendor team selected for the NGNP.

2. A limited qualification program for two potential metallic alloys (Alloy 617 and Alloy 230) for very high temperature applications will be continued. These alloys are primarily directed at the

3. A limited program directed at the further qualification and codification of C/C composites for potential applications associated with the NGNP control rods may be warranted. It is envisioned that continued effort in this area will be strongly influenced by the reactor design and vendor team selected. It is, however, unlikely that C/C materials will initially be used in the core of the NGNP in critical application because of the lengthy process and time required for the development of appropriate ASTM standards and adoption of an ASME codification process for these materials. Based on prior reactor licensing experience, it is unlikely that these materials will

be accepted for use in the core of the NGNP by the NRC without completion of the ASME Section III process for the use of these materials. The current alternative to the use of C/C materials for control rod components is Alloy 800H, however, this material has temperature limitations and potential radiation issues.

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

5.2.2 FY-08 Materials Program Plan

The planned activities listed for FY-08 are dependent on NGNP Materials Program budget and other factors which are currently not clarified. The activities listed are those that have been previously submitted to DOE for planning purposes and are considered important activities to support the NGNP schedule. In some cases the activity will be broken down into individual tasks, some of which may be performed at a later date, depending on the FY-08 budget.

5.2.2.1 Reactor Pressure Vessel

5.2.2.1.1 Technology development and acquisition strategy

This plan will be developed in the first half of FY-08. It is critical that a RPV material be selected in FY-08.

5.2.2.1.2 Support early scoping tests and measurements of RPV emissivity.

These scoping tests are being supported through a NERI program at the University of Wisconsin. The NGNP materials program will closely monitor results of that project.

5.2.2.1.3 Heavy section metallurgy

The through-thickness hardenability will be investigated on section sizes representative of the RPV. Specific tasks will depend on the selection of the RPV alloy.

5.2.2.1.4 Define inspection needs

Specific definition of inspection criteria and methods need to be supplied. Definition of the criteria and methods of inspection for the major components, on-sight fabrication of the RPV and in-service condition of the RPV (as well as the graphite and heat exchanger) will all be initiated.

5.2.2.1.5 Aging effects on heavy sections.

The long-term metallurgical stability of heavy section pressure vessel steels at representative operating conditions of the NGNP will be characterized. The microstructure and mechanical properties of the base metal and the weldments will be investigated.

5.2.2.2 Very High Temperature Alloys

5.2.2.2.1 Technology development and acquisition strategy

The development and acquisition strategy for high-temperature alloys will be completed by February of FY-08. This includes consideration of heat exchanger design, mechanical properties and codification of Ni-based high temperature alloys, and consideration of alloys for other core internals.

5.2.2.2.2 Alloy 617/230 down select criteria studies

Performance characteristics of Alloys 617 and 230 will be investigated to clearly determine the possible superiority of one of the alloys for heat-exchanger applications. The potential differentiating characteristics to be investigated are creep-fatigue behavior, carborization and thin-section environmental effects.

5.2.2.2.3 Creep/fatigue characterization

Complete creep/fatigue scoping characterization of Inconel 617 and Haynes 230 plate and weldments. Perform constant temperature, constant load testing at INL. Expose specimens to impure He environment and test mechanical properties at ORNL.

5.2.2.2.4 Aging and environmental effects in impure He

Experimental validation of an environmental effects model in controlled chemistry He will be completed for In-617 and Haynes 230. The stability of microstructures and mechanical properties will be characterized for temperatures ranging from 800-1000°C, with impurity contents expected in the NGNP environment.

5.2.2.2.5 Define inspection needs of intermediate heat exchanger.

Non-destructive evaluation methods will be studied for complex geometries expected in the NGNP heat exchanger. Minimum detectable flaw size in diffusion bonded joints and weldments will be considered.

5.2.2.2.6 Characterization of thin product forms.

Microstructure and properties of thin sections will be characterized and compared to plate material.

5.2.2.2.7 Product form definition of thin sheet and thick section materials.

A study will be completed to determine the requirement for thin sheet characterization as part of the data package to support the code case.

5.2.2.2.8 Component definition (insulation, core barrel, control rods)

Materials requirement for other high temperature materials in addition to the IHX and RPV will be determined. Candidate materials for these applications will be identified.

5.2.2.3 High Temperature Design Methodology

5.2.2.3.1 Support high temperature material ASME code case

This will support NGNP participation in ASME Code Week meetings.

5.2.2.3.2 Develop high temperature materials design rules

Develop high temperature materials design rules that will be used for potential consideration of the eventual code case. This is a joint effort with INL, ORNL and ANL. ORNL will subcontract a portion of this activity. The focus of this activity is to develop constitutive models of macroscopic properties, simplified methods computer code for new models and to validate these models using structural features testing.

5.2.2.3.3 Develop high temperature material creep/fatigue model

Experimental characterization of creep/fatigue behavior of In-617 and Haynes 230 and their weldments will be used to develop a phenomenological constitutive model of their continuum behavior.

5.2.2.3.4 Support development of materials handbook

5.2.2.3.5 Develop a unified high temperature constitutive model

A constitutive model will be developed that will describe uniaxial behavior of In-617 and Haynes 230 for creep and low strain-rate plastic behavior.

5.2.2.4 Welding (and other joining)

5.2.2.4.1 Evaluate behavior of fusion weldments

Creep and creep/fatigue behavior of weldments in Inconel 617 and Haynes 230 joined using best current commercial practice will be evaluated.

5.2.2.4.2 Fusion welding of Ni-based alloys

Improved methods of joining Haynes 230 will be investigated.

5.2.2.4.3 Characterization of weldments and bonded regions

Microstructure and mechanical properties of thin section In-617 and Haynes 230 weldments and diffusion bonds will be characterized.

5.2.2.4.4 Fusion welding of steels

Microstructure and mechanical properties of heavy section pressure vessel steel weldments will be determined. Requirements for pre- and post-weld heat treatment will be evaluated.

5.2.2.4.5 Aging effects on weldments

The long-term metallurgical stability of fusion welds in Ni-based alloys will be examined.

5.2.2.4.6 Alloy fabricability of Ni-based alloys.

The potential for detrimental effects associated with high temperature thermal cycles during brazing and diffusion bonding will be determined.

5.2.2.5 NGNP Component Testing

Conceptual design for a facility to test pumps, valves and heat exchangers will be completed.

5.2.2.6 Evaluate R&D needs for composite development.

Requirements for composites considered for application in core internals will be evaluated.

5.2.2.7 University Programs

Continue collaboration with the University of Michigan NERI for NGNP alloys for high temperature service. Continue to monitor the progress of University of Wisconsin NERI evaluating emissivity of NGNP materials and University of Illinois NERI on constitutive modeling of Ni-based alloys.

5.2.2.8 Manage the materials program.

Cost, scope, schedule and deliverables will be managed commensurate with the needs of the NGNP program. Funding level will vary significantly depending on the number of tasks that can be supported by the FY-08 budget.

5.2.2.8.1 Revise the NGNP Materials Program Plan.

A more detailed plan for high temperature metals will be produced by February 2008.

5.2.2.8.2 Foreign travel for additional support of PMB Working Group.

Travel will be funded as necessary to support the materials activities of the PMB.

5.2.3 Materials Program Plan Beyond FY-08

The NGNP Materials Program workscope has not been determined in detail for FY-09 and beyond. It is assumed that the workscope will be strongly influenced by the results of a detailed materials acquisition strategy, NGNP trade studies, the NGNP vendor team selected, the NGNP design selected.

6. Program Cost and Schedule

This section contains projected costs for high temperature materials and graphite R&D activities. Graphite activities have been more thoroughly vetted than High Temperature Materials (e.g. the Graphite Technology Development Plan - INL/EXT-07-13165). A detailed technology development plan for the High Temperature Materials R&D requirements is planned for next fiscal year. However, some of the near-term HT Materials R&D activities are presented in Table 8 for completeness.

6.1 HT Materials

Table 8. Near-term HT Materials R&D activities.

Research Activity	**Comments**
Project Management	Includes funding for project management, quality assurance and records management activities for NQA level 1 program.
HTM and RPV tech development & acquisition strategy planning	Complete required activity list with total estimated costs
617/230 down select criteria studies	
HTM Creep-fatigue characterization	
Materials evaluation of HTM Aging and environmental effects in He	1)Aging effects on base metal 2) Aging effects on weldments 3) Aging effects on heavy sections
Develop high temperature materials design rules for potential consideration of ultimate code case:	1) Develop constitutive models of macroscopic properties 2) Develop simplified methods computer code for new models 3) Structural features testing - (review of historical data)
Evaluate behavior of fusion weldments	1) Ni-based alloys 2) Fusion welding of steels 3)Characterization of weldments and bonded regions
Develop HTM creep/fatigue model based on data	
Unified high temperature material constitutive model	
Development and validation of environmental effects model	
Component testing (pumps, valves, heat exchanger)	Component definition (insulation, core barrel, control rods)
Reactor pressure vessel	1) Heavy section metallurgy 2) Define inspection needs
Intermediate heat exchanger	1) Develop NDE and ISI inspection techniques 2) Product form definition with thin sheet and thick section materials 3)Characterization of thin product form
Brazing & Diffusion bonding of Ni-based alloys	1) Brazing for Ni-alloy fabricability 2) Diffusion bonding for Ni-alloy fabricability
Estimate of near-term HT Materials R&D costs	A more detailed and complete R&D activity list and the associated costs will be developed next fiscal year under the HT Materials Technology Development Plan

6.2 Graphite

Experimental testing and data collection are considered to be the largest costs for the graphite R&D program. As indicated in Sections 4 and 5 the list of required material properties are fairly extensive and the irradiation testing program rather long. The activities supporting licensing (development of whole core models, ASME code case development, and NRC licensing reviews) are assumed to be less time and cost intensive however the exact activities are less defined leading to uncertainty for appropriate budgets. As a consequence, the costs are broken into two areas; experimental data collection and licensing support.

6.2.1 Data Collection Costs

The costs for support activities such as quality assurance, sample procurement/fabrication, and pre-irradiation tasks in addition to the actual testing programs are discussed as well. The experimental work is further divided into non-irradiated and irradiated tasks to better reflect the development plan in Section 5.0. A brief description of the identified activities and the estimated costs are shown in Table 9.

Table 9. Estimated costs for graphite R&D

Activity	Estimated Costs	Comments
	Experimental testing	
Procurement of graphite lots for sample fabrication Statistical char. Irradiated testing	$250k per lot (10-15 billets) 4 lots per graphite $1M per graphite	PBMR was able to use some of the graphite billets from each lot to fabricate components for their first core. This cost saving occurs only for the graphite type selected for the NGNP reactor.
Source qualification	$1-2M	1. Will establish the requirements for source qualification (qualify other coke sources for additional graphite production) in future cores. 2. Will be required for design certification over lifetime of reactor if new graphite is used. 3. Actual qualification of new coke/graphite sources will have costs similar to the current NGNP graphite development program.
Statistical thermo-mechanical characterization	$3-4M per graphite	Non-irradiated material property database. Includes machining costs, all testing (including multi-axial), and data analysis.
AGC irradiation capsule design and review	$5M	It is assumed that the approval costs for future AGC capsules will be significantly reduced once the generic design for all AGC capsules has been approved in FY 08.
Preparation for PIE of irradiated graphite samples	$5M	To meet NGNP schedule requirements the INL will need to modify existing laboratories to facilitate PIE of graphite irradiation samples in parallel with ORNL.

Activity	Estimated Costs	Comments
AGC experiment – Irradiation	$4M per 3-dpa capsule $8M per 7-dpa capsule $36M total	Nominal review and approval of new capsule design, construction costs, and irradiation fees (neutron costs).
AGC- experiment – PIE	$3.5M per capsule $21M total	All irradiated physical, thermal, and mechanical testing to be performed for each graphite.
HTV 1&2 design and approval	$2M	Includes design, approval, and construction costs for these simpler “drop-in” capsules. Neutron costs will be minimal since these tests are in HFIR.
HTV 1&2 – PIE	$2M	Physical, thermal, and mechanical testing of these un-loaded specimens
Oxidation studies	$2M	Both development of ASTM test standards for oxidation testing of nuclear graphite as well as determining oxidation rates of non-irradiated and irradiated graphite.
Baseline experimental costs (prismatic reactor design)	$78-80M	1. This is the estimated experimental expense for qualifying a graphite type for use within a prismatic reactor design (the selected baseline design). 2. Additional costs for a pebble bed design are included at the end of this table.
Design validation		
Micro-scale modeling	$2M	1. As in all modeling efforts this activity can be very expensive. Careful selection of specific work focused on license approval will reduce the costs significantly. 2. It is assumed the primary funding source for micro-scale modeling will be NERI type awards. 3. Whole core models in direct support of ASME code case and NRC licensing approval will be funded significantly by reactor venders.
Macro-scale modeling	$6-10M	
Whole core modeling	$6M	
NDE Development Pre-irradiation – as-received In-service inspection (ISI)	$5M	NDE techniques capable of characterizing the as-received graphite components before emplacement within the reactor as well as in-service inspection tools to ensure the integrity of the graphite components within the core are expected to be needed for NRC licensing.
ASTM test standards development	$4M	ASTM committee duties, standard writing, and participation in Round Robin proof testing.

Activity	Estimated Costs	Comments
ASME code case support	$3M	ASME committee duties and participation in data collection 10 years participation 2 researchers from INL/ORNL
Project management	$8-10M	Includes funding for project management, quality assurance and records management activities for NQA level 1 program.
Baseline estimated design validation costs (prismatic design)	*$34-40M*	1. As stated above, modeling costs can vary dramatically. These costs are considered the minimum necessary for NRC licensing requirements. Whole core model costs will most likely be shared by vendors in support of licensing their design. 2. NDE costs are relatively unknown at this time since the scope is undefined. However, in-service inspection of graphite components is necessary for UK reactors and anticipated for PBMR and will probably be required for NGNP.

Beyond baseline costs (Pebble Bed design additional)		
Procurement of graphite lots for sample fabrication Statistical char. Irradiated testing	$250k per lot 1-2 lots per graphite $500K per graphite	1. Costs are reduced since PBMR has currently an extensive non-irradiated materials property database for NBG-18 which is the graphite of choice for a pebble bed design. 2. May need to order more graphite for additional characterization data to meet USA regulator requirements.
Source qualification	$1M	1. Will establish the requirements for source qualification (qualify other coke sources for additional graphite production) in future cores. 2. Will be required for design certification over lifetime of reactor if new graphite is used. 3. Actual qualification of new coke/graphite sources will have costs similar to the current NGNP graphite development program.
Statistical thermo-mechanical characterization	$1.5-2M per graphite	1. Costs are reduced since PBMR has currently an extensive non-irradiated materials property database for NBG-18 which is the graphite of choice for a pebble bed design. 2. Non-irradiated material property database. Includes machining costs, all testing (including multi-axial), and data analysis.
High dose creep – design and approval	$5M	Some cost savings using previous experience with AGC and HTV 1&2 capsule designs.

High dose creep – irradiation	$8-12M	This experiment is 4X longer time in ATR than the longest AGC capsules. Potential savings could use HFIR since the flux in HFIR is ~ 3X higher than ATR. However, there is limited volume in HFIR and availability is at 50%.
High dose creep – PIE	$5M	The much higher dose may make these graphite samples more difficult to handle and subsequently test. Additional costs will be associated.
Beyond baseline additional costs	~ $21-25.5M	1. As noted above expenses for procurement and qualification of graphite are on a "per graphite basis". This cost estimate will increase for more graphite types being tested. 2. The high dose creep experiments are valid only for long term exposure of graphite (i.e. pebble bed reflectors). Costs may be reduced depending upon which design is selected.

As seen from Table 9, total costs are a function of the number of graphite types to be investigated, the selected reactor design, and the operating parameters (i.e. temperature, dose levels) of the selected reactor. A prismatic design is used for a cost baseline with the assumption that the R&D program will change if a pebble bed design is selected. A number of variables can adjust the costs for each reactor design.

6.2.1.1 Prismatic

PCEA graphite has been selected for the prismatic NGNP design but others may be considered (see graphite acquisition strategy) adding to the overall costs. A full thermo-mechanical characterization program for all graphite types will be required.

6.2.1.2 Pebble Bed

NBG-18 graphite has been selected for the pebble bed NGNP design but others may be considered (see graphite acquisition strategy). Graphite irradiations to 25 dpa will be required for front face of reflector blocks adding to irradiation experiments costs. A partial thermo-mechanical characterization program for NBG-18 graphite will reduce the overall costs since PBMR has already performed significant testing in this area.

6.2.2 Data Collection Schedules

A preliminary schedule for all graphite work has been recently developed. This master schedule incorporates not only the irradiation schedules for AGC as discussed previously but also graphite procurement, non-irradiated data collection, and required ASME & NRC licensing effort timelines. This master schedule is presented in modified form in Figure 19.

As shown the schedule does not incorporate those tasks needed for support of a pebble bed design selection (i.e. high dose experiments, minimal non-irradiated characterization, or adjustments to the existing irradiation program). The schedule is based upon the baseline assumption of a prismatic reactor design. Once key design decisions are made by the NGNP project, the schedule and cost baseline will be updated to reflect these decisions and a more detailed resource loaded schedule will be produced.

Activity ID	Early Start	Early Finish	Orig Dur	Predecessors	Successors
N3G1025	09JUN08	21OCT09	349	N3G1005*	N3G6500*
N3G2005	18JUN09	01NOV10	349	N3G1005, N3GA025*, N3GA035, N3TMS02	N3G2025*, N3G3005*
N3G2025	02NOV10	13DEC11	284	N3G2005*	N3G6550
N3G2H025	01OCT09*	09MAR10	107		N3G2H125*
N3G2H125	10MAR10	18MAR11	261	N3G2H025*	N3G6550
N3G3005	02NOV10	12JUL11	174	N3G2005*, N3GA025	N3G3025*, N3G4005*
N3G3025	13JUL11	23JUL12	261	N3G3005*	N3G6550
N3G4005	13JUL11	23OCT12	327	N3G3005*, N3GA025	N3G4025*, N3G5005*
N3G4025	24OCT12	04NOV13	261	N3G4005*	N3G6550, N3GA325*
N3G5005	24OCT12	05JUL13	175	N3G4005*, N3GA025	N3G5025*, N3G6005*
N3G5025	08JUL13	18JUL14	262	N3G5005*	N3G6550
N3G6005	08JUL13	14OCT14	324	N3G5005*, N3GA025	N3G6025*
N3G6025	15OCT14	08MAR16	350	N3G6005*	N21350
N3G6500	09JUN08	11DEC14	1,651	N3G1025*	N3G6550*
N3G6550	03SEP13	06APR16	654	N3G2025, N3G2H125, N3G3025, N3G4025, N3G5025, N3G6500*, N3GA035, N3GA045, N3GA135, N3GA145	N21350
N3GA005	09JUN08	07JUL08	20	N3G0005*	N3GA015*, N3GA250*
N3GA015	08JUL08	01SEP08	40	N3G0005, N3GA005*	N3GA025*
N3GA025	02SEP08	17JUN09	200	N3GA015*	N304995, N3G2005*, N3G3005, N3G4005, N3G5005, N3G6005, N3GA035*, N3GA045*
N3GA035	18JUN09	05APR10	200	N3GA025*	N3G2005, N3G6550, N3GA135*
N3GA045	18JUN09	05APR10	200	N3GA025*	N3G6550, N3GA145*
N3GA135	06APR10	30MAR11	250	N3GA035*	N3G6550
N3GA145	06APR10	30MAR11	250	N3GA045*	N3G6550
N3GA250	08JUL08	13JAN09	129	N3GA005*, N3TMS02	N3GA255
N3GA255		28JUL09	0	N120998*, N3GA250	N3GA275*
N3GA275	29JUL09	13MAY10	200	N3GA255*	N157250
N3GA310	08NOV12	04NOV13	250		N3GA325*
N3GA325	05NOV13	27OCT15	500	N3G4025*, N3GA310*	N3GA375*
N3GA375	28OCT15	12OCT18	750	N3GA325*	N304925*

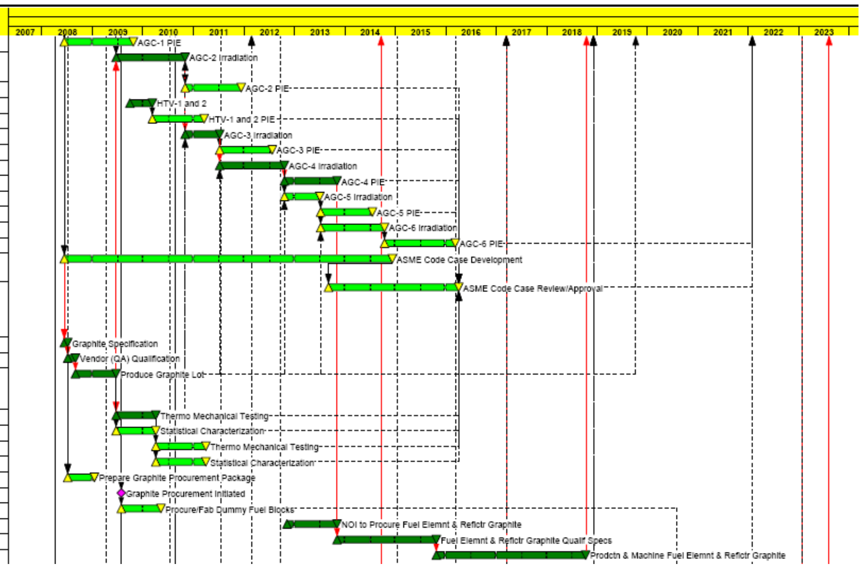


Figure 19. Schematic of master schedule for graphite R&D effort.

6.3 High Temperature Materials

Budget information for High Temperature Materials is included in Table 8. Further details for additional years on the High Temperature Materials portion of the program will be provided in an updated program plan.

6.4 Longer Term Considerations

What is presented above is a minimal "baseline" estimate for performing the required graphite R&D for the NGNP. Some discussion of each is warranted to determine the impact upon the R&D program.

6.4.1 Graphite Acquisition Plan

Currently, the world market share for nuclear graphite is extremely small. While graphite manufacturers are willing to produce nuclear grade graphite the petroleum industry which produces the raw starting material, specialty coke, is much less interested. The material specifications for specialty coke are much more exacting than what is needed for electrode production, the majority market share for graphite. Since this material's market share is so small the coke suppliers have very little financial interest in changing their production process to enable manufacture of these small batches of specialty coke necessary for nuclear graphite production.

As a consequence, there may not be enough specialty coke material needed for initial or sustained production for nuclear graphite for HTGR applications. Obviously this can significantly affect the graphite R&D schedule if multiple lots of graphite are required for testing and qualification. This potential shortage of coke sources has been addressed in much more detail within the NGNP Graphite Selection Strategy Report (ORNL/TM-2007/153).[29]

6.4.2 Graphite Disposition and Recycle Options

Currently, NSNFP and OCWRM will be disposing Ft. St. Vrain and Peach Bottom fuel blocks in Yucca Mountain. The C^{14} and Cl^{36} loading from these fuel blocks is insignificant compared to isotope inventories from commercial fuel's long lived fission products and transuranics. Graphite reflector blocks from Ft. St. Vrain were disposed in lower level radioactive landfill. Only Europe faces federal controls on C^{14} and Cl^{36} loading in graphite.

At this time there is no federal guidance on recycling irradiated graphite. Recycling irradiated graphite will depend on a number of factors including; the number of HTGRs (i.e. volume of graphite generated), the ability to decontaminate irradiated graphite, the performance of recycled graphite, and the total cost of recycling. It is expected that as the volume of irradiated graphite grows due to more HTGRs in operation the cost-to-benefit ratio of graphite recycle will become more favorable.

Euratom has begun development of decontaminating processes where the C14 is removed from along the grain boundaries of irradiated graphite using a heated oxygen gas. The contaminated gas is capture and the "clean" blocks are ready for Low Level Waste (LLW) disposal or possible recycling. Complete decontamination of graphite to below LLW thresholds (crushing plus chemical means) is possible but expensive. Once the graphite has been decontaminated two recycling options are currently envisioned; (a) reuse of blocks after heat treatment to anneal out radiation damage or (b) form new blocks using reconstituted graphite material by crushing and jet milling irradiated blocks to fine powder.

Once a successful technology is developed for decontaminating graphite the primary issue for recycling is the irradiation performance of the recycled graphite. A new qualification program will be necessary to validate the performance of this recycled graphite source, either for reuse of blocks or reconstituted material.