Contents

A9.1 INTRODUCTION AND BACKGROUND A9-5

A9.1.1 Description of National Materials Crosscutting Program A9-5

A9.1.2 Timeline for National Materials Crosscut Program A9-5

A9.2 RESEARCH AND DEVELOPMENT A9-5

A9.2.1 Objectives of National Materials Crosscut Program A9-5

A9.2.2 Scope of National Materials Crosscut Program A9-5

A9.2.3 Technical Issues for National Materials Crosscut Program A9-5

A9.2.4 Research Interfaces of National Materials Crosscut Program A9-6

A9.2.4.1 Relationship of Crosscutting and Reactor-Specific Materials Research A9-6
A9.2.4.2 Relationship to Generation IV International Forum (GIF) R&D Projects A9-7
A9.2.4.3 University Projects A9-7
A9.2.4.4 Industry Interactions A9-8
A9.2.4.5 I-NERI A9-8
A9.2.4.6 Interactions with Other Materials R&D Programs A9-8

A9.3 R&D HIGHLIGHTS A9-9

A9.3.1 MATERIALS CROSSCUTTING TASKS A9-9

A9.3.1.1 Materials for Radiation Service A9-9
A9.3.1.2 Materials for High-Temperature Service A9-10
A9.3.1.3 Development of Microstructure-Properties Models A9-12
A9.3.1.4 Development of Improved High-Temperature Design Methodology A9-13

A9.3.2 Reactor-Specific Materials A9-15

A9.3.2.1 NGNP Reactor-Specific Materials A9-16
A9.3.2.2 GFR Reactor-Specific Materials A9-32
A9.3.2.3 SCWR Reactor-Specific Materials A9-38
A9.3.2.4 LFR Reactor-Specific Materials A9-41

A9.3.3 Materials for Energy-Conversion System A9-47

A9.3.3.1 Summary of High Priority Materials Research Areas for Nuclear Hydrogen Production A9-48

A9.3.4 National Materials Program Integration A9-49

A9.4 INTEGRATED MATERIALS PROGRAM COST AND SCHEDULE.................................... A9-49

A9.4.1 Materials Program Budget.. A9-49

A9.4.2 Schedule for the Integrated Generation IV Materials Program A9-50

A9.4.3 High Level Program Milestones for the Integrated Generation IV Materials Program.. A9-51

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A9.1 INTRODUCTION AND BACKGROUND

A9.1.1 Description of National Materials Crosscutting Program

An integrated research and development (R&D) program will be conducted to study, qualify, and in some cases, develop materials with properties required for the Generation IV advanced reactor systems.

A9.1.2 Timeline for National Materials Crosscut Program

The integrated Generation IV Materials R&D program is planned to provide materials data needed to design, license, and construct the Next Generation Nuclear Plant (NGNP) by 2017 and to provide adequate data to assess the viability of the other Generation IV reactor systems by 2010.

A9.2 RESEARCH AND DEVELOPMENT

A9.2.1 Objectives of National Materials Crosscut Program

The objective of the National Materials Crosscut Program (NMCP) is to ensure that the Generation IV materials R&D program will comprise a comprehensive and integrated effort to identify and provide the materials data and its interpretation needed for the design, codification, licensing, and construction of the selected advanced reactor concepts.

A9.2.2 Scope of National Materials Crosscut Program

The NMCP explicitly includes materials R&D generally considered crosscutting: (1) qualification of materials for service that must withstand radiation-induced challenges; (2) qualification of materials for service that must withstand high-temperature challenges; (3) the development of validated models for predicting long-term, physically based microstructure-property relationships for Generation IV reactors; and (4) the development of an adequate high-temperature-materials design. Additionally, it contains the overall management and coordination function for the Generation IV Integrated Materials Program that also addresses materials issues specific to individual reactor and energy-conversion systems. An extensive summary of the overall Generation IV Integrated Materials Program is contained in the draft report Updated Generation IV Reactors Integrated Materials Technology Program Plan, Revision 1, ORNL-TM-2003/244 (R1), August 31, 2004.

A9.2.3 Technical Issues for National Materials Crosscut Program

For the range of service conditions expected in Generation IV systems, including possible accident scenarios, sufficient data must be developed to demonstrate that the candidate materials meet the following design objectives:

- acceptable dimensional stability including void swelling, thermal creep, irradiation creep, stress relaxation, and growth;
- acceptable strength, ductility, and toughness;
- acceptable resistance to creep rupture, fatigue cracking, creep-fatigue interactions, and helium embrittlement; and

- acceptable chemical compatibility and corrosion resistance (including stress corrosion cracking and irradiation-assisted stress corrosion cracking) in the presence of coolants and process fluids.

Additionally, it will be necessary to develop validated models of microstructure-property relationships to enable predictions of long-term materials behavior to be made with confidence and to develop high-temperature materials design methodology for materials, use, codification, and regulatory acceptance.

A9.2.4 Research Interfaces of National Materials Crosscut Program

A9.2.4.1 Relationship of Crosscutting and Reactor-Specific Materials Research

Since many of the challenges and potential solutions will be shared by more than one reactor concept, it will be necessary to work with the System Integration Managers (SIMs) for each individual reactor concept to examine the range of requirements for its major components, to ascertain what the materials challenges and solutions to those will be. It will then be necessary to establish an appropriate disposition of responsibilities for the widely varying materials needs within the Generation IV Initiative. It is expected that there will be two primary categories for materials research needs:

- Materials needs that crosscut two or more specific reactor systems and

- Materials needs specific to one particular reactor concept or energy conversion technology.

Where there are commonly identified materials needs for more than one system, it will be appropriate to establish a crosscutting technology development activity to address those issues. Where a specific reactor concept has unique materials challenges, it will be appropriate to address those activities in conjunction with that particular reactor system's R&D. The National Materials Program within the Generation IV Initiative will have responsibility for establishing and executing an integrated plan that addresses crosscutting, reactor-specific, and energy-conversion materials research needs in a coordinated and prioritized manner.

Reactor-specific materials research that has been identified for the individual reactor and energy-conversion concepts includes materials compatibility with a particular coolant or heat-transfer medium, as well as materials expected to be used only within a single reactor or energy conversion system, such as graphite, selectively permeable membranes, catalysts, etc. A special category of reactor-specific materials research will also include research that must be performed at pace that would significantly precede normal crosscutting research in the same area (e.g., NGNP reactor system materials R&D).

While the current plan addresses materials issues for all the reactors currently being examined within the Generation IV program, there is recognition that the plans to build a Very High Temperature Reactor (VHTR) as the NGNP by 2017 will strongly drive much of the materials research during the next ten years of the program. Accordingly, though the four crosscutting activities described in Section A.9.3.1, a major focus of materials research during the next ten years will be on the qualification of commercial and near-commercial materials and the related high-temperature design methodology (HTDM) needed to specify and order those components. Parallel studies on materials for other reactor concepts will both take advantage of the accelerated work for the NGNP and examine additional materials under other conditions where the NGNP materials studies are inadequate or inappropriate for their conditions.

A9.2.4.2 Relationship to Generation IV International Forum (GIF) R&D Projects

The Integrated Generation IV Materials Program directly supports several of the Generation IV International Forum (GIF) goals, including:

> Sustainability-1 Generation IV nuclear energy systems will provide sustainable energy generation that meets clean air objectives and promotes long-term availability of systems and effective fuel utilization for worldwide energy production.
>
> Sustainability-2 Generation IV nuclear energy systems will minimize and manage their nuclear waste and notably reduce the long-term stewardship burden, thereby improving protection for the public health and the environment.
>
> Economics-1 Generation IV nuclear energy systems will have a clear life-cycle cost advantage over other energy sources.
>
> Economics-2 Generation IV nuclear energy systems will have a level of financial risk comparable to other energy projects.
>
> Safety and Reliability-1 Generation IV nuclear energy systems operations will excel in safety and reliability.
>
> Safety and Reliability-2 Generation IV nuclear energy systems will have a very low likelihood and degree of reactor core damage.

Whereas the materials program supports the goals on sustainability and economics in general, it is the goals on safety and reliability to which the Generation IV Materials Program is a key contributor.

The U.S. Generation IV Materials Program directly addresses issues facing four of the six GIF reactor systems, including the VHTR, Gas-cooled Fast Reactor (GFR), Supercritical Water Reactor (SCWR), and Lead-cooled Fast Reactor (LFR). The R&D planned in the Generation IV Materials Program is being coordinated with those similar activities being performed by our GIF partners within the Materials and Components Project Management Boards empanelled as part of the System Steering Committees. Activities within the Generation IV Materials Program are intended to provide the materials information needed to enable U.S. Department of Energy (DOE) to assess the viability of the GFR, SCWR, and LFR by 2010, with respect to U.S. interests, as well as to provide materials data needed to enable the design, ordering of long-lead components, and licensing of the NGNP as the lead VHTR reactor by 2017.

A9.2.4.3 University Projects

University projects to be performed in conjunction with the Generation IV materials program will include (1) research directed to a specific university, based on unique capabilities of the performing institution or work considered to be of a critical, programmatically required nature, and (2) research competed and awarded as part of the Nuclear Energy Research Initiative (NERI) sponsored by the DOE Office of Nuclear Energy, Science and Technology (NE).

Examples of directed research currently include the assessment of corrosion and stress corrosion of candidate materials for the Generation IV SCWR system by universities with existing capability and expertise in this area. An area being considered for directed research is the establishment of a new low-flux irradiation facility for the exposure of reactor pressure vessel steels to replace a similar facility that was shut down in conjunction with the recent closing of the University of Michigan's Ford Nuclear Reactor. A decision on where the irradiation facility will be established is anticipated in FY-05.

Materials work to be performed as part of the DOE NERI program is selected with consideration of both technical excellence and programmatic relevance. Review of these attributes of any proposed materials work is performed as a part of the decision-making process leading to contract awards. In addition to providing technical contributions to the overall Generation IV materials program, work performed within the NERI program has the additional benefit of helping to recruit and develop professional personnel for many disciplines needed for the U.S. nuclear infrastructure, including advanced materials expertise. Examples of technical areas envisioned as fruitful for materials NERI programs include the development and validation of models relating materials microstructures, processing, and properties that can be used to predict behavior in Generation IV systems and contributions to HTDM development needed to assess, codify, and license materials time-dependent behavior.

A9.2.4.4 Industry Interactions

In the early portion of the materials R&D program, interactions with industry will fall into two broad categories: (1) interactions with designers and component suppliers to assist in formulating the requirements needed for the various reactor components and systems and (2) interactions with materials suppliers to obtain information on or develop materials specifications and architectures, fabrication and joining processes, and commercially developed and maintained databases. As the maturity of the designs for the Generation IV reactor systems and their components increases, prototypical materials and model components or subcomponents will be produced by commercial and industrial suppliers for evaluation of their performance and properties. Examples of areas envisioned as having required industrial interactions include: development of restricted or improved compositions and processing of high-temperature alloys for enhanced performance; assessment of manufacturing methods and materials for the very large reactor vessels needed for some of the Generation IV concepts; fabrication and testing of advanced heat exchanger designs; and the assessment of materials properties, fabrication, and performance of large, component-specific, carbon-carbon or silicon carbide matrices (SiC-SiC) composite architectures.

A9.2.4.5 I-NERI

Until the full, multilateral implementing agreements for the GIF are put in place, International NERIs (I-NERIs) will provide the primary framework for collaborative materials research with GIF partners. All I-NERI projects are currently envisioned to address specific reactor concept issues. I-NERI research projects on materials have been implemented with France, Canada, the Republic of Korea, and Japan. These collaborative research programs will provide generation and bilateral sharing of materials information by each GIF partner. These agreements currently include activities focused on an evaluation of materials for SCWRs with Canada, the development and testing of cladding materials for the LFR with the Republic of Korea, development of SiC-SiC for control rod structures for NGNP applications and the evaluation of materials for GFRs with France, and the development of materials for SCWR systems with Japan. Additional I-NERI materials R&D programs will be developed and implemented in keeping with joint research programs and activities within the U.S. and our GIF partners until replacement multilateral GIF agreements are established.

A9.2.4.6 Interactions with Other Materials R&D Programs

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

A final category of materials R&D that is recognized within the Generation IV Program is that which overlaps the materials needs for the development of fuels and reprocessing technology within the Advanced Fuel Cycle Initiative (AFCI) and for chemical processing equipment for the Nuclear Hydrogen Initiative (NHI). While both AFCI and NHI are independent programs with their own research objectives and funding, it has already been recognized their applications will contain many of the same conditions that exist for reactor systems and their components in the Generation IV Program and, hence, may utilize a common set of structural materials. A special collaboration among all three programs has been developed and is being maintained to help ensure that the materials R&D being conducted within them is coordinated to minimize duplication and costs and maximize mutually beneficial materials technology development and qualification.

A9.3 R&D HIGHLIGHTS

A9.3.1 MATERIALS CROSSCUTTING TASKS

A9.3.1.1 Materials for Radiation Service

The performance of structural materials is limited, in general, by the degradation of physical and mechanical properties by exposure to energetic neutrons or by exposure to the chemical environment provided by the primary coolant medium. Although there are very significant differences in operating environments between the various concepts under consideration, it is possible to identify a number of common environmental features. Of these common features, operating temperatures and neutron exposures will have the greatest impact on materials performance and component lifetimes, leaving aside for the moment the issues surrounding radiation-assisted corrosion phenomena. Therefore, combining the evaluation of materials as a function of neutron exposure offers an opportunity for addressing the development and qualification of materials for multiple concepts within a coordinated set of irradiation experiments. Evaluation of candidate materials that are applicable for multiple concepts offers both an improved overall database and the potential for significant cost savings compared to conducting separate irradiation programs for each reactor concept. A prime example would be the design and implementation of an irradiation program that would simultaneously serve the needs for an irradiation effects database for many of the Generation IV reactors.

A second important crosscutting feature to be considered is that data on radiation effects must be obtained for all Generation IV reactor concepts from a limited set of operating test reactors and irradiation facilities For example, these include the High Flux Isotope Reactor (HFIR) and Advanced Test Reactor (ATR) in the United States, the High-Flux Reactor in the European Union, the BOR60 in Russia, and the Japanese Material Test Reactor and Japanese Experimental Fast Reactor in Japan. Significant opportunities exist for the sharing of information on the technology of irradiation testing, specimen miniaturization, advanced methods of property measurement, and the development of a common materials property database system that would crosscut all potential reactor concepts. Although it is possible in a limited number of cases to provide an irradiation test environment that is prototypical for some of the components of a particular Generation IV concept, irradiation test conditions are generally non-prototypical, either because the required spectral conditions cannot be achieved or the required neutron lifetime exposures can only be achieved by testing at accelerated dose rates. Additionally, individual components experience spatial variations in flux, spectrum, and mechanical loading. Of necessity, materials selection will have to be based upon incomplete experimental databases, and consequently, there is a strong and crosscutting need for the development of physically-based models of critical radiation effects phenomena in both face-centered-cubic and body-centered-cubic alloy systems based upon advanced microstructural analysis. Such validated models are needed to provide a sound basis for making extrapolations and interpolations from the available experimental radiation effects database. While the development of such models will be conducted within a separate task within the NMCP focused on that area, the development of the experimental databases upon which those models will be based will be responsibility of work within this task.

A final important thread linking the structural materials of various Generation IV system in-vessel components is that several classes of structural alloys could find application in more than one system. Examples include creep-resistant, low-swelling austenitic stainless steels and ferritic/martensitic steels for in-vessel components for the SCWR, GFR, and LFR and nickel-based alloys for the NGNP and MSR internals. For very high temperature applications, refractory metal alloys and structural composites such as SiC-SiC could have potential applications for more than one concept. Within the rapidly evolving field of mechanically alloyed materials, oxide-dispersion-strengthened (ODS) alloys based on austenitic, ferritic, or ferritic/martensitic matrices have the potential to significantly advance the performance of components for all the Generation IV concepts under consideration. Programs to develop ODS materials for nuclear applications are being strongly pursued in Europe and in Japan. Efforts to understand the processing-microstructure-property relationships for mechanically alloyed materials could eventually lead to the development of alloys with exceptional high-temperature creep strength, microstructural stability, outstanding resistance to void swelling, and the ability to retain properties following off-normal temperature excursions.

The activities and funding within this task and its associated high-level milestones included in Section A9.4 are expected to address general needs of materials for radiation service for the GFR, LFR, and SCWR systems. Specialized, schedule-driven, reactor-specific needs for NGNP system materials for radiation service are addressed in Section A9.3.2.1 on NGNP materials. Specialized, reactor-specific needs for SCWR system materials for radiation service that must additionally address stress-corrosion cracking and irradiation-assisted stress-corrosion cracking are addressed in Section A9.3.2.3 on SCWR materials. Specialized reactor-specific needs for the GFR system materials for radiation service for in-core and core support applications that must address neutronically acceptable materials are addressed in Section A9.3.2.2 on GFR materials. Funding for the reactor-specific NGNP, GFR, and SCWR irradiated materials research is included within the materials funding requirements in Appendices 1.0, 2.0, and 3.0, respectively.

A9.3.1.2 Materials for High-Temperature Service

In the Generation IV Initiative, although the operating conditions vary significantly from one reactor system to the next, significant commonality exists with regard to the selection of materials for their high-temperature structural components. These common issues will be advantageously addressed in this crosscutting task. However, in setting out the scope and schedule of this crosscutting task, it is recognized that the highest priority for development and qualification of materials for high-temperature service is given to NGNP because it is the first candidate system to be deployed. Therefore, for qualification of materials for high-temperature service, early crosscutting efforts will be focused mainly on establishing activities that will complement those being pursued for NGNP and on establishing a sound foundation for the high-temperature materials programs for the other Generation IV reactor systems. This will pave the way for the crosscutting activities for the other reactor systems to gradually increase in scope as portions of the NGNP efforts approach completion.

As noted above, despite the various operating conditions in the proposed reactor systems, significant commonality exists with regard to the selection of materials for their high-temperature structural components. As a result, the materials for Class I nuclear components for service above the temperature limits of American Society of Mechanical Engineers (ASME) Section III will be limited to those materials incorporated into Section III, Subsection NH. Currently, this subsection permits construction with a very few alloys, namely type 304H and type 316H stainless steels, alloy 800H, and 2 1/4Cr-1Mo steel (class 1). The incorporation of Gr91 (modified 9Cr-1Mo-V) steel is in progress. To take full advantage of the potential of the reactor concepts in the Generation IV Initiative, it will be necessary to utilize the advances made in structural materials technology, select the most promising candidate materials for higher temperature service, and move forward toward acceptance of these materials into the appropriate construction codes.

Even though many of the materials that will be required for construction of high-temperature, out-of-core components will be the same as those used for some in-core applications, the focus of this crosscutting technology development task will be on their un-irradiated high-temperature qualification. While short-term tensile and fatigue properties will need to be evaluated for these materials, it is time-dependent creep and creep-fatigue, which are the primary limitations for materials use, that will be principally addressed. The crosscutting technology development associated with high-temperature use of these materials in the presence of neutron irradiation will be addressed in the task on Qualification of Materials for Radiation Service described in Section A9.3.3.1. The related high-temperature corrosion and compatibility concerns for these materials will be addressed as part of reactor-specific R&D tasks and summarized in Sections A9.3.1 to A9.3.4.

For the high-temperature materials to be evaluated for out-of-core applications for the Generation IV initiative, the goal of this crosscutting materials research is their eventual incorporation into ASME Section III, Subsection NH. The materials for such high-temperature service may be separated into several categories by approximate upper-use temperatures. While there is some overlap, and more advanced materials within a class will somewhat extend the temperature limits of current materials, these classes roughly correspond to: (a) ferritic steels including bainitic and martensitic steels up to 12% chromium for use up to about 650°C; (b) austenitic stainless steels for use up to about 800°C; (c) high alloys, in which iron content is greater than any other element, and nickel-base alloys for use up to about 900-950°C; and (d) special materials such as ODS alloys for possible use up to about 1000-1050°C.

The three primary thrusts within this crosscutting activity in the next ten years of the Generation IV Initiative will be to:

1. Evaluate the current commercial or near-commercial materials for adequacy of data and properties to incorporate into Subsection NH of the ASME Section III for high-temperature service and begin the codification of the appropriate materials, including generation of incremental required data bases

2. Perform evaluation and screening of promising advanced materials for higher temperature service, resulting in the selection of candidate materials for further development and eventual inclusion into the Section III Subsection NH

3. Develop and maintain a Generation IV Materials Handbook that will serve as the definitive repository and source for materials data needed to evaluate, design, and license Generation IV reactor concepts.

These evaluation and development activities will include all appropriate product forms and section thicknesses needed for required reactor components, including weldments and their constituents (weldmetal, heat affected zone, and basemetal). Since the crosscutting activity involves Generation IV reactor systems with later anticipated deployment dates than that of the NGNP, more efforts for evaluation of advanced materials for high-temperature service can be included.

The activities and funding within this task and its associated milestones included in Section A9.4 are expected to address general needs of materials for high-temperature service for the GFR, LFR, and SCWR systems. Specialized, schedule-driven, reactor-specific needs for NGNP system materials for high-temperature service are addressed in Section A9.3.2.1 on NGNP materials. Specialized, reactor-specific needs identified for the GFR system for non-metallic materials for high-temperature service are addressed in Section A9.3.2.2 on GFR materials. Funding for the reactor-specific NGNP and GFR high-temperature materials research will be included within the materials funding requirements in Appendices 1.0 and 2.0, respectively.

A9.3.1.3 Development of Microstructure-Properties Models

For each design objective described in Section A9.1, the development and evolution of the fundamental microstructural features that establish materials performance need to be understood to further improve material performance and/or ensure the very long operational life envisioned for Generation IV reactor systems. This understanding will require a combination of theory and modeling activities tied to detailed microstructural characterization and mechanical property measurements. The models must be developed using the best current materials science practice in order to provide a sound basis for interpolating and extrapolating materials performance beyond experimental databases, as well as providing the fundamental understanding needed to make designed changes in material compositions and processing to achieve improved properties.

At the Higher Temperature Reactor Materials Workshop, sponsored by the DOE Offices of NE and Basic Energy Sciences in March of 2002, the issues associated with microstructural development and modeling were extensively discussed. Significant conclusions from the meeting, including needs for the Generation IV Reactor Initiative, are:

- Displacement damage during irradiation creates a non-equilibrium, structure-chemistry evolution at the nanoscale that alters plasticity, corrosion-oxidation, and fracture processes. The crucial elements of the microstructure that evolve with irradiation are voids and bubbles, dislocation loops and stacking fault tetrahedra, carbides and other precipitates, and network dislocations. Radiation-induced solute segregation (RIS) can lead to the formation of unexpected phases in the matrix and composition changes at free surfaces and interior interfaces. RIS influences both mechanical properties and corrosion behavior. In addition, the diffusion and segregation of helium and hydrogen to vacancy clusters and voids is a major contributor to swelling. Fundamental understanding of these complex, interdependent, radiation-induced material changes is essential to underpin the development of Generation IV reactor systems.

- The key structural performance issues for most irradiated metallic alloys are hardening-induced embrittlement at low temperatures and time-dependent deformation (creep and fatigue) and cracking at high temperatures. The evolution of non-equilibrium structures and chemistries promotes a hardened matrix and lower grain-boundary cohesive strengths thereby reducing the tensile stress required for cleavage or intergranular fracture. At high temperatures, the radiation-induced changes in the matrix and particularly at grain boundaries can promote creep embrittlement. The atomistics of fracture need to be combined with micromechanical models to better elucidate behavior in complex, radiation-induced, multi-component nanostructures.

A series of integrated, physically based, empirically validated models will need to be developed to address the issues raised above, guide overall materials development, and ensure long-term materials stability during operation. Six general topics will need to be addressed.

- Development of improved fundamental understanding and modeling of the nucleation-phase of the various defect types that are produced during irradiation (e.g., vacancy and interstitial aggregates, second phases, etc.);

- Development of atomistic and continuum models that describe the mechanisms responsible for radiation-enhanced, -induced, and -modified microstructural changes and the physical phenomena that account for the persistence of those microstructures that remain stable at high temperatures;

- Development of the kinetic and thermodynamic models required to provide an understanding of the formation and stability, particularly under irradiation, of both undesirable and desirable second phase precipitates. A critical example of a desirable second phase is the very fine oxide clusters that provide the high-temperature strength of ODS alloys;

- Development of improved micromechanical models to investigate the detailed interactions between dislocations and other microstructural features that control material strength and deformation behavior. Detailed atomistic modeling is required to provide parameters and insight for higher level deformation models;

- Development of improved understanding of the mechanisms that contribute to high-temperature, time-dependent plasticity (e.g., creep-fatigue, ratcheting, etc.) and the models describing them for application and insight into the Improved HTDM to be developed under a separate crosscutting task; and

- Performance of detailed microstructural analysis, down to the atomic scale, on Generation IV candidate materials using state-of-the-art characterization techniques (e.g., atom probe, X-ray and small-angle neutron scattering, positron annihilation, high-resolution transmission electron microscopy, etc.) to provide microstructural input for model development.

Although the detailed microstructural analysis required for model development may be carried out as part of this task, it is anticipated that the samples for examination will be obtained from materials irradiated in experiments carried out under other tasks, particularly those on Qualification of Materials for Radiation Service and Reactor-Specific Materials. In some cases, special-purpose experiments may be proposed and conducted as part of this effort.

The activities and funding within this crosscutting task and its associated milestones included in Section A9.4 are expected to address the anticipated microstructural analysis and model development needs for all Generation IV reactor systems.

A9.3.1.4 Development of Improved High-Temperature Design Methodology

The objective of the HTDM Task is to establish the improved and expanded structural design technology necessary to support the codification and utilization of structural materials in high-temperature Generation IV reactor system components. The temperatures and materials requirements of most Generation IV components exceed the time/temperature coverage currently provided by Subsection NH of Section III of the ASME Boiler and Pressure Vessel Code, which governs the design and construction of elevated-temperature, Class 1 nuclear components. This task will provide the data and models required by ASME Code groups to formulate time-dependent failure criteria and assessment rules and procedures that will ensure adequate life for components fabricated from the metallic alloys chosen for Generation IV systems. The task will also provide the material behavior (constitutive) models for the detailed inelastic design analysis methods required by Subsection NH for assessing critical structural regions, and it will provide the simplified inelastic design analysis methods that are allowed for less critical regions and are used for preliminary design.

Subsection NH of the ASME Code currently covers just four high-temperature alloys: 304 and 316 stainless steel, 2-1/4 Cr-1Mo steel, and Alloy 800H. Modified 9Cr-1Mo steel (Grade 91) has been approved but is not yet included. The maximum temperature coverage for these materials is inadequate for NGNP, GFR, and LFR (long-term version). In addition, the maximum design life allowed is, 34 years, whereas Generation IV components are to have a design life of 60 years. Thus, most Generation IV systems will require the inclusion in Subsection NH of new materials with higher permitted temperatures and longer operating times. Even for systems and components operating within the range of coverage of Subsection NH, new stronger materials may be desirable, and in any event, the time coverage must be significantly increased.

Candidate structural materials for Generation IV systems fall primarily into two classes: medium high-temperature alloys, characterized by the Cr-Mo steels and AISI 304 and 316 stainless steel, and very high-temperature alloys, characterized by nickel-base alloys. The strategy for this task is to focus initial efforts on a single representative and promising material from each class—modified 9Cr-1Mo steel (Grade 92) at medium high temperatures and nickel-base Alloy 617 at very high temperatures. As other key structural alloys are identified for the various reactor concepts, they will be factored into the effort, especially where an identified material is common to more than one reactor concept. The initial focus on modified 9Cr-1Mo steel (Grade 92) and Alloy 617 and the resulting criteria, design analysis, and assessment methods will provide the framework and springboard for introducing additional materials as they are identified. They will also provide the near-term tools needed by NGNP designers to develop conceptual and preliminary designs.

A unique requirement for most Generation IV materials is that they will operate at the upper end of their useful temperature range. At the lower end of a material's useful elevated-temperature operating range, the inelastic response to cyclic thermal and mechanical loadings, especially at discontinuities, can usually be separated into time-dependent plasticity and time-dependent creep. Current Subsection NH rules and criteria, as well as the associated inelastic design analysis methods and simplified methods, depend heavily on this assumed separation. At higher temperatures, the distinction between rate-independent plasticity and time-dependent creep blurs for many materials (e.g., modified 9Cr-1Mo steel, Grade 91, and Alloy 617), and the separation between behaviors is no longer valid. The response becomes very rate dependent, and both strain-and cyclic-softening occur. The criteria and analysis methods for Generation IV components must be formulated to reflect these behavioral features.

The HTDM Task has several subtasks. The first is the development of experimentally based constitutive equations required for inelastic design analyses. These equations, which will be developed for each key material starting with modified 9Cr-1Mo steel (Grade 92) and Alloy 617, will be unified in that they will not distinguish between rate-dependent plasticity and time-dependent creep.

The second subtask, which will be carried out in close coordination with the ASME Code Subgroup on Elevated Temperature Design, is the development of failure models for design criteria. These models, which again will be experimentally based, normally consist of two parts: (1) a damage accumulation model describing failure resulting from the accumulation of damage under time-varying thermal and mechanical loadings and (2) a strength criterion describing failures under multi-axial stresses. Major challenges of this subtask are developing an adequate treatment for creep-fatigue failures, especially at very high temperatures, and an improved means of addressing notches and weldments (both major unresolved NRC concerns from the Clinch River Breeder Reactor Plant licensing process).

Perhaps the most challenging subtask will be the development of simplified methods. While the underlying premise of Subsection NH is that the variation of stresses and strains with time in a high-temperature component should be predicted by detailed inelastic design analyses, the wide use of such analyses for preliminary design and for every region and loading condition of a component would prove impracticable. Thus, limited simplified rules for satisfying strain limits (ratcheting) and creep-fatigue criteria are included in Subsection NH. However, at the upper end of a material's operating range, the material response previously described violates basic assumptions used in developing the existing simplified methods. Thus, new methods must be quickly developed since they are required in the early stages of design.

The final two subtasks are (1) confirmatory structural tests and (2) procedures for safety and reliability assessments. The role of structural tests, which will involve the determination of deformation and failure behavior in generic features as opposed to actual components), is to either validate the high-temperature structural design methodology or if that does not occur, to guide required improvements. The safety/reliability subtask will focus on the safety assessment methodology that will be required for licensing. Included will be a high-temperature flaw growth and assessment procedure and a criterion for ultimate structural failure.

The activities and funding within this task and its associated milestones included in Section A9.4 are expected to address the HTDM needs for materials for the GFR, LFR, and SCWR systems. Specialized, schedule-driven, reactor-specific needs for development of HTDM for NGNP system materials are addressed in Section A9.3.2.1 on NGNP materials. Funding for the reactor-specific NGNP materials research is included within the materials funding requirements in Appendix 1.0.

A9.3.2 Reactor-Specific Materials

Reactor-specific materials research includes materials compatibility with a particular coolant or heat-transfer medium used in a single reactor system, as well as structural materials expected to be used only within a single reactor or energy conversion system, such as graphite, selectively permeable membranes, catalysts, etc. Additionally, where research must be performed at pace that would significantly precede crosscutting research in the same area (e.g., NGNP reactor system materials R&D), it has also been classified as being reactor-specific.

Reactor-specific research identified to date is described for each reactor system in the sections that follow. Materials needs for the NGNP, GFR, and SCWR systems have been fairly well identified at this point, and those needs that are not addressed in the crosscutting tasks described in Section A9.3.1 are summarized below. A high-level assessment of materials needs for the LFR systems has been performed but is not quite as detailed as those performed for the other systems, given the relatively recent definition of system conditions for the LFR. Future revisions of this appendix are expected to update the materials needs for all systems, in general, and expand upon the materials needs for the LFR, in particular.

While limited funding has been provided for a small crosscutting task established to provide coordination of reactor-specific materials research, the funding for the actual research, development, and qualification of reactor specific materials is included within the materials funding requirements within Appendices 1.0 to 4.0.

A9.3.2.1 NGNP Reactor-Specific Materials

The DOE has selected the VHTR design for the NGNP Project. The NGNP will demonstrate the use of nuclear power for electricity and hydrogen production without greenhouse gas emissions. The reference reactor design is a graphite-moderated, helium-cooled, prismatic or pebble bed thermal neutron spectrum reactor that will produce electricity and hydrogen in a state-of-the-art, thermodynamically efficient manner. The NGNP will use very high burn up, low-enriched uranium, TRISO-coated fuel and have a projected plant design service life of 60 years.

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

- The capability to obtain an NRC operating license and provide a basis for future performance-based, risk-informed licensing
- Support the development, testing, and prototyping of hydrogen infrastructures
- The ability to produce electricity with high efficiency using a high temperature Brayton Cycle at full scale
- Demonstrate nuclear-assisted production of hydrogen using about 10% of the heat
- Demonstrate by test the exceptional safety capabilities of the VHTR
- Validation of the acceptability of the materials of construction as a bridge to commercialization
- A full-scale prototype VHTR prior to 2017

The NGNP Materials Program is responsible for performing research and development on NGNP materials in support of NGNP licensing and design activities. The NGNP Materials Program includes the following elements:

- Development of a specific approach, program plan, and other project management tools for managing the research and development program elements performed
- Development of a specific work package and project execution plan for the research and development activities to be performed for each government fiscal year
- Reporting status and progress of the work based on committed deliverables and milestones to DOE
- Developing collaborations in areas of materials R&D of benefit to the NGNP with countries that are a part of GIF
- Ensuring that work performed in support of the materials program is in conformance with established QA and procurement requirements

- Establishing an interface with the Project Integrator (following DOE selection) to continue to facilitate materials research and development in support of NGNP licensing and design activities.

A summary of material R&D plans for NGNP materials is provided in the sections that follow. More background and details on these plans are contained in the Next Generation Nuclear Plant Materials Research and Development Program Plan, INEEL/EXT-04-02347, Revision 1, September 2004. Estimates of funding required for the NGNP materials R&D are included in the NGNP Appendix 1.0 of this ten-year plan.

A9.3.2.1.1 Graphite Materials. Graphite will be used as a structural material and neutron moderator for the NGNP core as the permanent side reflectors, and for the core support structure. A significant challenge related to graphite for the NGNP is that the previous U.S. standard graphite grade qualified for nuclear service, H-451, is no longer commercially available. The precursors from which H-451 graphite was made also no longer exist. Hence, it will be necessary to qualify new grades of graphite for use in the NGNP. Fortunately, likely potential candidates currently exist. These include fine grained isotropic, molded or isostatically pressed, high-strength graphite suitable for core support structures, fuel elements, and replaceable reactor components and near isotropic, extruded, nuclear graphite suitable for the above-mentioned structures and for the large permanent reflector components. These candidates would meet the requirements of the draft ASTM materials specification for the Nuclear Grade Graphite.

Graphite Selection Strategy

Several candidate graphites have been identified for components within the NGNP (Table A9.1). The scope of the NGNP graphite program will work collaboratively with activities within the GIF for graphite selection and database development. A strategy for the selection and acquisition process for the NGNP graphite is being developed with the collaboration of several GIF members and potential vendors. A series of meetings in Europe in January 2005 will finalize the selection and acquisition process.

Table A9.1. Candidate graphites for the core components of the NGNP.

NGNP Concept	Component Description	Candidate Grades
Prismatic Block	Fuel Element & Replaceable Reflector	Graftek PCEA SGL Carbon NBG-10,17,18 Toyo Tanso IG-110
Prismatic Block	Large Permanent Reflector	Graftek PGX SGL Carbon HLM
Prismatic Block	Core Support Pedestals & Blocks	Graftek PCEA SGL Carbon NBG-10,17,18 Carbone USA 2020 Toyo Tanso IG-110
Prismatic Block	Floor Blocks & Insulation Blocks	Graftek PCEA SGL Carbon NBG-10,17,18
Pebble Bed	Reflector Structure	Graftek PCEA SGL Carbon NBG-10,17,18 Toyo Tanso IG-110
Pebble Bed	Insulation Blocks	Graftek PCEA SGL Carbon NBG-10,17,18

Variations between billets within a single lot and between different lots must be fully characterized. Sufficient data must be taken such that the data are statistically significant to quantify the extent of in-billet and between-billet property variations. Moreover, graphite purchased should meet the requirements established for the ASTM Nuclear Graphite Materials Specification. Graphite thus purchased will additionally be used for ASTM test method development.

A materials properties design database must be developed for the selected NGNP graphites, including data for the effects of reactor environment on properties (including neutron irradiation and irradiation creep).

Graphite Baseline Materials Test Program

The baseline graphite test program will fill in the database with data that cannot be obtained from European and Japanese programs. The baseline materials test program must be sufficient to fully characterize and quantify property variations within candidate graphite billets arising from the raw materials forming process (e.g., parallel and perpendicular to the forming axis) as well as spatial variations (i.e., billet edge and center). Microstructural characterization of candidate graphite will be conducted in order to establish filler particle and pore size distribution (required for fracture modeling). X-ray diffraction (XRD) will be applied to establish crystal parameters and appropriate crystallinity factors for neutron irradiation behavior modeling and prediction.

Prior work and data for nuclear graphite behavior will be reviewed and assessed in an effort to minimize the extent of the testing program.

Physical and mechanical properties to be determined include:

- Mechanical Properties – strength (tensile, compressive, flexural), biaxial/multi-axial strength, stain to failure, elastic modulus, Poisson's ratio, fatigue strength, fracture toughness.

- Thermal Properties – thermal conductivity, thermal diffusivity, coefficient of thermal expansion (CTE), emissivity, specific heat.

- Tribology – Significant work has been previously performed on graphite-graphite friction couples. This work needs to be reviewed and documented in the graphite materials database.

- The chemical purity and Boron equivalent content of the candidate graphite will be determined.

Property data is needed as a function of temperature and environment (helium). Moreover, the long-term effects of impurities in the helium coolant (air, water, oxygen) on the graphite properties must be established (Graphite oxidation). All of the properties determined under the baseline graphite materials test program will need to be re-assessed for the effects of oxidation from helium coolant impurities (air, CO_2, water). Graphite air oxidation kinetic data must be obtained for the candidate graphites for air-ingress accident simulation and modeling. Design specification data will be required on the helium coolant purity limits, as this will control the severity of the property degradations.

Graphite Irradiated Materials Test Program

Significant structural changes occur upon neutron irradiation. The single crystal effects and gross structural effects combine to modify practically all of the properties. Thus, for preliminary selection of candidate graphites, those properties listed in the baseline program above must be examined for the effects of irradiation at a temperature representative of service conditions.

The effects of neutron irradiation over the temperature and dose range appropriate to the NGNP must be established as part of the qualification process. A significant body of data on the effects of irradiation exists and more is being developed by other GIF partners. An initial set of scoping irradiations and two much more extensive set irradiation experiments focusing on very high temperature exposures and irradiation creep are planned to supplement data that will otherwise be available.

Graphite Scoping Irradiations

A series of 36 NGB-10 nuclear graphite bend-bar samples have been irradiated in rabbit capsules in the HFIR at Oak Ridge National Laboratory (ORNL). Post irradiation examination of the samples will include determination of irradiation effects data to assist in selection of NGNP graphites from among the candidates and to guide the more extensive irradiation experiments.

High-Temperature Graphite Irradiation Experiments

There are few data for the irradiation behavior of graphite at temperatures >1000 °C. Hence, a high-temperature graphite irradiation capsule will be designed which will be capable of irradiating graphite samples at temperatures up to 1200 °C. An evaluation will be made to determine the most appropriate HFIR vehicle for these irradiations based upon capsule size limitations, ease of attaining the desired temperatures, and availability of space in the HFIR (e.g., rabbit capsule, target capsule, or reflector capsule). The first capsule will be designed, along with an experimental plan and required QA documentation, in FY-05. Irradiation data to be determined on the candidate graphite(s) will include dimensional changes, elastic constants, strength, and coefficients of thermal expansion.

Graphite Irradiation Creep Experiments

Graphite samples will be loaded under compressive stress and irradiated at representative temperatures in an ATR creep capsule being designed at INL. In addition to creep rate data, post irradiation examination of the control samples will yield valuable irradiation effects data for the NGNP. The graphite samples will be selected from multiple vendors and grades of graphite. The capsule will be designed, all necessary Quality Assurance documentation prepared, and an experimental plan prepared in FY-05. Capsule construction and bench testing will commence in FY-06. It is anticipated that creep irradiations would be completed in FY-07 or FY-08.

Graphite Model Development for Predicting Irradiation Effects

Mathematical models must be developed that describe and predict the behavior of nuclear graphite under neutron irradiation. Such models should be based upon physically sound principles and reflect known structural and microstructural changes occurring in graphites during fast neutron irradiation, such as changes in crystallinity, pore shape, coefficient of thermal expansion (bulk and single crystal), etc. Models for the graphite irradiation dimensional changes and irradiation creep behavior are a priority. Existing irradiation data may be used for model development, but validation of the models must be conducted using irradiation data obtained on the newer nuclear graphites being considered for the NGNP. Input data for such models must be obtained from the NGNP candidate graphites. Several modeling approaches will be explored. For example, models based on microstructural changes as described by bulk and crystal CTE changes or fundamental atom-displacement models linked to finite element codes will be considered.

Codes and Standards

Significant activity is required to bring the existing graphite codes and standards to an acceptable condition. The proposed Section III Division 2, Subsection CE, of the ASME Boiler and Pressure Vessel (B&PV) Code (Design requirements for Graphite Core Supports) was issued for review and comment in 1992 and no action has been taken on this code since that date. There is activity underway currently (funded by the Nuclear Regulatory Commission [NRC]) to reinitiate the "CE" code committee and begin the process of code case approval. However, significant revision of the code is required as well as expansion of the code to the higher temperatures envisioned for the NGNP. The NRC has also indicated that the code should be revised to increase the neutron dose limits to levels appropriate to the Pebble-Bed Modular Reactor.

Graphite test standards have been developed for nuclear grade graphites (ASTM C-781: Standard Practice for Testing Graphite and Boronated Graphite Components for High Temperature Gas-Cooled Nuclear Reactors). This ASTM standard must be further expanded to cover required test methods including, Fracture Toughness, XRD, Graphite Air Oxidation, Boron Equivalency. Moreover, the standard must address specimen size issues as they relate to the preparation of graphite irradiation specimens. ASTM is currently preparing a nuclear grade graphite material specification under the jurisdiction of Committee DO2-F.

Fabrication Infrastructure Development Requirements and Program

Appropriate Non-Destructive Examination methods must be developed for large graphite billets and components. Such methods must be applied prior to accepting production billets for fuel element/component machining and will be useful for subsequent in-service inspection.

A9.3.2.1.2 High-Temperature Design Methodology (HTDM). Within the NGNP Materials Program, both high temperature materials testing and methodology development are included in the HTDM activities. This includes developing baseline high temperature materials property data and a design methodology applicable to several high temperature components including reactor internals, heat exchangers, electrical power conversion turbine and generator equipment, and primary coolant boundary components. The assessment of irradiation, environmental, and thermal aging effects on these materials will be addressed in separate tasks.

The HTDM project will provide the data and simplified models required by ASME B&PV Code subcommittees to formulate time-dependent failure criteria that will assure adequate life. Specifically, this project will also provide the experimentally based constitutive models that are the foundation of the inelastic design analyses specifically required by ASME B&PV Section III Division I Subsection NH.

The project will also provide data for use associated with regulatory acceptance. Safety assessments, required by NRC, will depend on time-dependent flaw growth and the resulting leak rates from postulated pressure-boundary breaks. This requires a flaw assessment procedure capable of reliably predicting crack induced failures as well as the size and growth of the resulting opening in the pressure boundary.

Additional background on issues associated with the development of an improved HTDM is provided in the corresponding crosscutting activity in Section A9.2.3.

Potential candidate high-temperature alloys for service above about 600°C are listed in Table A9.2. These materials include alloys for which significant databases exist and new state-of-the-art alloys which are being developed for other high-temperature applications.

Table A9.2. Potential Candidate Materials Selection for Intermediate and High-Temperature Metallic NGNP Components

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

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

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

Materials for somewhat lower temperature service in the reactor pressure vessel are identified and discussed in Section A9.3.2.1.5.

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

It is recognized that Alloy 617 is a very mature, high-temperature alloy and as such is the leading potential material candidate for very high-temperature usage in NGNP. It still has a number of issues that must be addressed to allow its longtime usage under the environmental and loading conditions envisioned. The development of joining and design methodologies for Alloy 617 will be important issues in component construction and long-term performance. Major shortcomings in the understanding of the interactions of creep, fatigue, and environment in all the alloys discussed and their weldments have been identified by the ASME and the NRC. Resolving these issues for Alloy 617 will both develop a technical approach to apply to other high temperature alloys and reinvigorate the ASME activities needed for their codification within ASME Section III, Subsection NH.

The NGNP HTDM program will begin to address these deficiencies by studying rate-dependent stress-strain behavior at relatively short times, creep, and creep-fatigue-environment interactions in Alloy 617, leveraging the results of existing programs on Alloy 617 base and weld metal and providing early data needed to complete development of high-temperature design methods required for its codification for nuclear service. Specific near-term activities are described in more detail in the tasks that follow. Other alloys will be added to the program based on need and funding provided.

- An evaluation of a controlled chemistry variant specification for Alloy 617 will be performed to investigate the potential for enhancing its high-temperature properties and minimizing their variation.

- Characterization of Alloy 617 fusion welds will be performed to assess basic microstructural properties and strength characteristics of the welds, thereby providing a better theoretical underpinning for component lifetime models and high-temperature structural design methodology.

- Creep and creep-fatigue testing of Alloy 617 base- and weld metal specimens in impure He and control environments at 800°C to 1000°C will be performed, leveraging testing ongoing in other DOE programs, the Ultra-Supercritical Steam Generator program at ORNL and the Materials for Energy Research program, at the Idaho National Laboratory (INL).

- Aging tests of Alloy 617 for 10,000 hour, 1000 °C, of inert atmosphere encapsulated base alloy and welded samples will be performed to provide a baseline of thermal aging effects in the absence of environmental effects related to impure helium exposure.

- As a companion activity to the high-temperature scoping tests and prior to the substantial efforts needed to generate the large database of mechanical property data needed for codification, a thorough assessment and compilation of existing data is required

- Additional approaches for simplified methods will be examined and developed. This will include investigation of new approaches in the type of creep-fatigue tests and the use of test data with design rules; the purpose is to avoid the deconstruction of cyclic creep damage into creep and fatigue damage.

- An in-depth survey of literature of component behavior at very high temperature will be conducted. This will include constitutive equations for stress-strain evolution under various loading conditions for Alloy 617 and Alloy X/XR, efforts at addressing multi-axial effects on damage, and extrapolation of relatively short creep data for use in designing a reactor for a 60-year life.

A9.3.2.1.3 ASME and ASTM Support of the Development of High Temperature Materials. Currently there are many areas relating to ASTM standards method development and ASME B&PV Code development that need to be pursued to meet NGNP goals. The NGNP Materials R&D Program must initiate a presence at the ASTM and ASME B&PV Code meetings at the relevant committee and subcommittee level to be able to incorporate new materials or extend the application of materials presently in the Code or existing test standards. Personnel will support appropriate committees and develop required standards and validation testing.

Much of this effort provides required technological support and recommendations to the Subgroup on Elevated Temperature Design (NH). While the codification or updating of code status of other alloys will be required for NGNP, it is recognized that Alloy 617 is a both a prime candidate for NGNP applications and a good choice for NH to use in establishing the codification activities for such materials. Hence, the initial focus will be addressing the existing Alloy 617 draft ASME Code case, which has a number of gaps and shortcomings that will have to be overcome before it can be written and satisfactorily and reliably applied. These following required tasks were identified as the code case was being developed:

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

- Weldment stress rupture factors must be added.

- Thermal expansion coefficients must be added.

- Additional isochronous stress-strain curves must be added.

- Create simplified methods for Alloy 617

ASME design code development is required for the graphite core support structures of the NGNP and for the carbon-carbon (C-C) composites structures of the core. A project team under Section III of ASME is currently undertaking these activities led by NGNP materials personal. Standard test methods are also required for the generation of data that may be used in the design code. Such methods are developed through the ASTM and then adopted by the ASME. The ASTM DO2-F committee of Manufactured Carbons and Graphites is currently engaged in the final stages of developing a Standard Materials Specification for Nuclear Grade Graphite, and it is also developing several standard test methods for graphites (crystallinity by x-ray diffraction, surface area, thermal expansion, fracture toughness, and graphite oxidation for example). NGNP participation in DO2-F committee work is vital to the timely completion and adoption of such standard test methods.

NGNP staff will also support the formation of an ASTM working group on SiC-SiC composite testing development and ensure that guidelines for testing of tubular SiC-SiC structures proceeds in the required time frame.

A9.3.2.1.4 Environmental and Thermal Aging Testing Program. The three primary factors that will most affect the properties of the structural materials from which the NGNP components will be fabricated are effects of irradiation, high-temperature exposure, and interactions with the gaseous environment to which they are exposed. An extensive testing and evaluation program will be required to assess the effects of these factors on the properties of the potential materials to qualify them for the service conditions required. The information given below provides an overall description of the work that needs to be performed with an early emphasis on aging and exposure to the reactor coolant.

Aging Tests

Procedures for the evaluation of aged and "service-exposed" specimens will be developed. Properties evaluation will be performed on a limited number of materials including Alloy 617, Alloy 800H, and Alloy X that have been aged at temperatures as high at 950 °C in helium to at least 25,000 hours. Mechanical and microstructural properties of bulk and weld structures will be evaluated, and the determined experimental properties will serve as input and checks of computational continuum damage modeling activity for high-temperature life prediction. Results of mechanical testing and microstructural evaluations of candidate alloys aged 1000, 3000, and 10,000 hours will serve as additional input to computational continuum damage models. The predictions of these models will be compared to results of testing of materials aged to at least 25,000 hours to validate these models. The mechanical and microstructural data will also provide input into code rules for accounting for aging effects.

A review will be performed of the extensive body of work on Alloy 617 and two other candidate materials to document the applicability of the available thermal aging effects data/information in the temperature range of interest to the NGNP. This review will also serve to highlight the areas where additional information is needed.

Reactor Pressure Vessel (RPV) alloy specimens will be prepared for thermal aging in air. Materials will be initially aged for up to 10,000 hours at 650 °C. These experiments will serve to provide a relatively early indication of each material's response to long-time, high-temperature exposure in air, a condition applicable to the uncoated outer surface of the RPV. Following aging at 10,000 h, a portion of each material will be further aged at about 650°C for 50-100 h. The aged materials will then be tested for tensile, creep, and toughness behavior and characterized microstructurally. Candidate materials and weldments will also be aged in the impure helium environment for the same times, mechanically tested, and microscopically examined. In addition, portions of the candidate materials and weldments will remain under thermal aging in both air and helium until at least 25,000 h, and it will then be tested to provide longer time data to allow for comparisons with predictive models. Finally, thermal aging of the prime candidate alloys at the RPV operating temperature will continue for several more years to accumulate data for very long-times.

Prototype C-C composite material components will be manufactured and tested under anticipated in-service conditions (i.e., service temperatures and environment). Properties data must be obtained for both C-C composite material and SiC-SiC composites. Following the initial down select to two vendors for both SiC-SiC and C-C, the candidate materials will be evaluated for the various in-service conditions. These service conditions may be unique to each component, so the architectures for each component may need to be individually tested at each condition (to be later specified by the designers). These activities will address both long-term corrosion due to helium impurities and short-term oxidation due to air ingress during accident conditions. Mechanical and thermal properties including tensile strength and modulus, dimensional changes, and thermal conductivity will be evaluated to verify and quantify effects of the time, temperature, and environment.

The results of the aging studies will be used to characterize the kinetics (reaction rate) such that activation energies can be calculated. From these activation energies, aging/life prediction models for the degradation of the materials can be developed. These models will be crucial because it will be impossible to determine the effects after a 60-year life without 60 years of testing. This accelerated life testing program will be used to reduce the time frame for gathering that data.

Evaluation of Helium Environments

The overall NGNP helium environment must be evaluated to ensure that testing proposed in various parts of the program are performed in environments that have consistent chemical potentials. In addition, the corrosion of metals and nonmetals will be evaluated to establish baseline data where it does not exist. These tests will be performed at temperatures to include at least 50 °C above the proposed operating temperature.

Helium Loops

Design and construction of a large, low-velocity helium loop with gas cleanup is underway. Special emphasis is being placed on the gas clean up system, which will serve as the prototype for a high-velocity loop. The system will be designed to operate using vacuum or inert gas as the reference atmosphere with capacity to mix ppm levels of impurities (e.g., H_2, CO_2, or water vapor) designed to simulate the NGNP environment. While the low-velocity loop is being readied, gas/gas studies will be performed in two small existing recirculating low-velocity helium loops to establish the dynamic stability of the helium environment.

An assessment of past helium test environments will be performed to determine the helium environment compositional range that should be used for the NGNP Materials R&D Program. In addition, a review of the existing data/information on the environmental effects of impure helium on Alloy 617 will be performed to document the applicability of existing data for the range of temperature and helium compositions of interest to the NGNP. These reviews will also delineate the ranges in which additional data is needed.

In future years, long-term creep testing capabilities will be designed and or augmented as needed. Existing creep facilities will be refurbished and additional creep-fatigue equipment procured as necessary to meet the need for high-velocity and long-term testing of materials in potentially contaminated helium environments. A new test loop will be designed and constructed for performing the required testing in helium with controlled impurity levels at temperatures up to 1100°C, 7.5 MPa pressure, and a flow rate up to 50 m/s.

A9.3.2.1.5 Qualification of Materials for Irradiation Service. Several possible primary coolant pressure boundary systems are envisioned for the NGNP. These comprise a large RPV containing the core and internals, a second vessel containing an intermediate heat exchanger (IHX) and circulator or a power conversion unit, and a pressure-containing cross vessel (CV) joining the two vessels. Because of the wide range of material thicknesses in the primary coolant pressure boundary system, it will be constructed in a segmented configuration. Although the specific design is not yet available, such a configuration will play a role in the materials selection as it relates to fabrication issues, effects of loading variables such as cycling, etc. The three vessels will be exposed to air on the outside and helium on the inside, with emissivity of the chosen material an important factor regarding radiation of heat from the component to the surrounding air to ensure adequate cooling during accident conditions.

The primary coolant boundary system will use either conventional materials as listed within ASME SA508/ SA533 specifications or it will be fabricated from materials never used previously for a nuclear reactor in the U.S. If the temperature can be maintained to less than 375°C by cooling or other means, conventional materials can be used. However, if the pressure boundary temperature is in the range of 375-500°C advanced materials will be required. The advanced materials tentatively selected for further investigation for pressure boundary service are ferritic/martensitic steels, alloyed primarily with chromium and molybdenum. The two most promising classes of commercially available steels are 9Cr-1MoVNb steels for higher temperature operation and 2.25Cr-1Mo for lower temperature operation.

More advanced developmental alloys may also be useful for RPV applications for subsequent generations of NGNP-type plants. These include the class of 7-9Cr2WV steels, currently being developed under the Fusion Materials Program to reduce activation under neutron irradiation with resultant advantages for decommissioning, and the class of 3Cr-3WV steels that offer the possibility of better high-strength properties and reduced section thicknesses.

In order to evaluate the irradiation effects of candidate RPV alloys under the relatively low-flux conditions applicable to vessel service, a new facility will be fabricated to replace the irradiation facility that was shut down recently at the Ford Test Reactor at the University of Michigan. The irradiation facility is anticipated to be a joint DOE and NRC facility. Preliminary design concepts envision two separate and independent operating capsules in the facility one for the NRC-funded Heavy-Section Steel Irradiation Program and the other for the Generation IV Reactor Materials Program. The capsules can be readily designed and fabricated to operate from 250 to 650 °C, with a preliminary fast neutron flux of about 1 to 2 x 1012 n/cm^2·s (>1 MeV). Approval to proceed with the design effort will first be obtained from the NRC and DOE, followed by site selection, and placement of a contract for facility construction. Any useful hardware from the Ford Test Facility will be retrieved and used in the new facility.

Although the operating temperature of the RPV and CV may change with evolution of the design, it is currently planned to irradiate mechanical test specimens at 400 and 600°C. The choice of these temperatures is based on the assumptions that (1) 600°C is the highest possible operating temperature that can be envisaged for the RPV and CV at this time, (2) 400°C is in the range of the lowest operating temperature that would allow for reasonable achievement of the objectives for the NGNP, and (3) the range between these temperatures would likely provide sufficient information for design and operation of the RPV at any intermediate temperature with respect to irradiation effects.

Irradiations of the preliminary candidate materials, both base metals and weldments, will begin in later years, with the choice of materials to be based on results of the literature review as well as the baseline and aging tests completed at the time. For purposes of this plan, specimens to be irradiated will include those for tensile, hardness, creep and stress rupture, Charpy impact, fracture toughness, and fatigue crack growth testing. Based on the currently estimated maximum exposure of about 1x1019 n/cm^2 (>0.1 MeV) and 0.075 dpa, the specimens will be irradiated to an exposure about 50% greater to accommodate uncertainties in the exposure estimates. A decision to conduct further test reactor irradiations beyond those noted above will be based on the results of the initial testing.

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

The fluences accumulated in the metallic core internal materials are expected to be low relative to the tolerances of the structural alloys. Nevertheless, for prudence, a review of the radiation effects on the metallic reactor internal components will be undertaken. The review will include a collection of data produced on austenitic alloys irradiated at high temperatures. This body of information will be characterized in terms of materials, exposure conditions, and testing conditions. Data judged pertinent to the NGNP will be evaluated in some detail and provided to the modeling activities. Consideration will be given to irradiation exposures. The selection of materials, exposure conditions, and the design of experiments will be undertaken. Exposures and evaluation of the irradiated materials will include an evaluation of the radiation-induced changes in microstructure, hardness, and ductility.

A9.3.2.1.6 Control Rod and Composite Structures. A number of structural composites were identified for potential use in control rods and other composite structural applications in the NGNP. The components and potential materials are shown in Table A9.3. The reason that composites are being considered for these applications is long-term exposure to temperatures greater than 1000°C. At these temperatures, most metallic alloys are ineffective.

A C-C or SiC-SiC composite material comprises a carbon or graphite matrix or a SiC matrix that has been reinforced with carbon or graphite fibers or SiC fibers.

Table A9.3. Potential Structural Composite Applications.

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

Composites of either C-C or SiC-SiC could be potentially used to fabricate several different components. Future qualification tests will be required to delineate which of the composites is the best choice for a given component based upon the response of the composite to exposures based on conditions expected within the reactor.

For simplicity, C-SiC composites were not included in the table, but were considered an intermediate between C-C and SiC-SiC composites. The C-SiC composites will be lower in cost than SiC-SiC composites but might exhibit cracking problems due to the use of dissimilar materials. The C-SiC composites were classified as a subcategory of SiC-SiC and would require the same qualification tests as SiC-SiC.

The use of C-C composites appears to be desirable for many applications within the reactor because of their strength retention at high temperatures. For example, C-C is a top candidate for the control rod sheath or guide tubes for a prismatic NGNP because metallic materials cannot withstand the level of neutron irradiation and high temperature of 1050°C or higher found in the core.

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

Initial irradiation studies

Currently, radiation resistant SiC-SiC composites have only been irradiated to fairly low (8 dpa) levels and exhibit little or no mechanical degradation. SiC-SiC composites may be stable out to at least 30 dpa without much degradation; however, this assumption needs to be validated.

High-purity SiC-SiC samples are being irradiated to higher irradiation levels in HFIR. It is expected that the specimens will reach about 10 dpa in FY-05 and 20 dpa in FY-06. Post irradiation examination will be carried out beginning in FY-05. Testing will include (but not be limited to) bend strength, dimensional stability, elastic modulus, and thermal conductivity. Based upon these preliminary results, the question of the irradiation stability of SiC-SiC composites versus C-C composites at higher doses should be resolved. Assuming that SiC-SiC composites are more stable, irradiation will continue to the 30-dpa levels in HFIR over the next few years. This work will answer the fundamental question "are SiC composites potential lifetime control rod materials" in contrast with C-C composites, which will not survive much above 10 dpa.

ASTM Standards Development– SiC-SiC composites

Assuming that basic SiC-SiC composite structures are shown to be stable at the doses required, it then becomes necessary to determine if they are suitable as control rod materials. This will require development of test and evaluation methods to carry out proof testing and defend component qualification. The initial step in this direction will be the generation of ASTM test methods for tubular SiC composites focusing on size effects on tubular properties. The primary motivation for the size effect study is to ensure that the small geometry required for irradiation studies are yielding adequate data. Representative samples from these tubes need fit into ATR irradiation positions; therefore, test samples much smaller than the actual control rod diameters will be required. In addition, in order to simplify irradiations in the ATR, “dog-bone” shaped flat tensile specimens have been proposed. This would provide a significant cost and time reduction in the SiC-SiC testing. However, before these smaller dog-bone flat tensile specimens can be used, it needs to be established that they are truly representative of large tubes used for the control rods.

Creep and Fracture Studies

Irradiation creep studies will include both out-of-pile and in-pile testing of composite creep samples (both SiC-SiC and C-C composites). Specific issues that must be addressed include:

- Upgrading or procuring creep test stands to accommodate inert atmosphere testing and accommodate very high temperatures (i.e., 1400°C) for off-normal events.
- Performing out-of-pile creep testing for baseline thermal creep results.
- Design, development, and coordination of SiC-SiC, C-C, and graphite creep capsules where applicable.

Environmental Effects

It is assumed that the fundamental irradiation response will be similar for all composite architectures and geometries. However, using different composite architectures (i.e., weave angles, fiber tow counts, weave structures, etc.) can lead to differences in the engineered materials due to infiltration efficiency, fiber bending stresses, or matrix/fiber interface characteristics. The environmental conditions these materials will be subjected to may change the overall creep response of the composite (i.e., creep crack growth for fiber-reinforced materials).

Existing creep crack growth models will be evaluated and augmented to predict the environmental factors on the overall creep of the SiC-SiC composite structures. The model will be expanded to include flat, thin specimens (i.e., simulate flat dog-bone shaped tensile specimens). It is anticipated that the model may be further expanded to include the 3-dimensional tubular geometry if applicable/desirable at a later time.

To improve the accuracy of the model predictions, a limiting environment for elevated temperature tests will be determined. Most likely, the limiting environmental species in the He loop will be the H_2/H_2O ratio. Assuming these species are the most damaging to the composites, a determination of the degradation potential for various H_2/H_2O ratios will be made using both modeling and experiments.

C-C Composites Studies and ASTM Standardization

The C-C composites have performance issues similar to the SiC-SiC composite structures for control rod applications. It is assumed that C-C will be used in all other composite applications where the dose is considerably lower (i.e., where irradiation stability is not as critical).

Since so many of the issues being addressed in the SiC-SiC composites are applicable to the C-C composites, extensive coordination will be required between the two programs. American Society of Testing and Materials (ASTM) subcommittee participation and the establishment of a material specification for a likely C-C architecture will be essential.

A survey of potential vendors will be conducted (domestic and foreign) to ascertain which vendors have the capability to fabricate complex architecture C-C composite components and what sizes can be processed. For the control rod assemblies, where neutron damage is a concern, consideration must be given to the ease of processing of the preferred fibers (mesophase pitch derived), which tend to have high modulus and are thus very difficult to weave. Heat treatment capabilities and furnace sizes/availability will be determined. This information will be required by NGNP designers in order to size the larger C-C components of the NGNP.

Candidate C-C composite materials for NGNP control rod applications will be procured and evaluated. The materials will be typical of those used in the NGNP components in terms of their fiber and matrix selections and processing conditions. It is anticipated that a review of New Production Reactor literature and R&D activities in this area will be conducted prior to the placement of a purchase order. Existing 3D C-C materials will be evaluated for the control rod applications.

A9.3.2.1.7 Data Management and Handbook. The organizational structure to be used in the preparation, control, etc. of NGNP data needs will be finalized for incorporation into the Generation IV Materials Handbook being developed in the Crosscutting Task on Materials for High Temperature Service. Existing materials handbooks will be examined to determine what information might be extracted and incorporated into the Generation IV Materials Handbook.

A Generation IV Materials Handbook "Implementation Plan," to be prepared as part of this task, will consider NGNP needs and issues. It will provide details of purpose, preparation, publication, distribution, and control of the Handbook. It will also prescribe records required, QA, and review and approval responsibility and authority. Once fully implemented, the Generation IV Handbook will become the repository for the NGNP materials data and serve as a single source for researchers, designers, vendors, codes and standards bodies, and regulatory agencies. It is also planned to evaluate the potential for including similar data from GIF international partners. Near-term activities in this area will include assembling and inputting existing data on materials of interest to NGNP.

A9.3.2.1.8 RPV Transportation and Fabrication Project. RPV heavy-section fabrication is a major issue that needs to be evaluated for the very large sized vessels envisioned for the NGNP. It is very unlikely that the manufacturing of the RPV would take place in the United States without a significant investment. Preliminary considerations and discussions indicate that Japan Steel Works is the most likely source of forgings of the required size. The physical size of even the largest required forging appears to be within their range of capability; however, the specific material selection is critical in that very large forgings of most of the potential candidate alloys listed have not been manufactured, including the 9Cr-1Mo-V alloy.

The main issue is attaining the required through-thickness properties of the higher-alloy steels in the thick sections required. Additionally, weldability of the steels in thick sections is also an issue. However, because of the relatively short lead-time available for ordering of components for the primary coolant pressure boundary system, fabricability and availability will also be major considerations in the selection of materials. Besides the technical issues, transportation of the completed RPV or large ring forgings from the vendor facilities to the reactor site may be problematic. The diameter of the RPV is relatively well known from the design, but the thickness and, therefore, the weight is not as well known. It is possible that the RPV will require field fabrication, meaning welding of the ring forgings, heads, etc. onsite. In this case, the conduct of post weld heat treatment (PWHT) takes on more significance in that a PWHT in the field is more difficult to conduct and control than that performed in the shop environment.

An assessment of these issues and approaches to address current limitations in fabrication and transportation technology will be the primary thrust of this task.

A9.3.2.1.9 Power Conversion Turbine and Generator Project.

Turbine and Generator Baseline Materials Test

For the turbine inlet shroud collar, the turbine shroud insulation package container/boundary, and the turbine blade the property of greatest importance is very high-temperature creep strength. Further, it is extremely important that the creep behavior (strength and ductility) not be degraded by impure helium or thermal aging. Early work should be initiated on the turbine shroud material to assure that adequate long-term creep data is available in the temperature range 950°C to 1050°C. In addition to the creep and environmental work, it will be necessary to address questions relative to both low-cycle and high-cycle fatigue at very high temperatures and the effects of impure helium interacting with metal on fatigue behavior.

Testing efforts aimed at the materials for the recuperator should be minimal. All needed mechanical property data are available; confirmatory environmental exposures are desirable but no adverse effects are expected.

The helium circulator operates at 600°C. There are no pressure stresses, but some concern exists about high-cycle fatigue and creep-fatigue. Stainless and ferritic steels, such as 2 1/4Cr-1Mo and 9Cr-

1Mo-V, are potential candidates. The hot ducting and bellows operate at 600°C but could reach 700°C in event of an accident. Alloy 800H is the leading candidate. The material selections will be based to some extent on the fatigue or creep fatigue resistance of the candidate alloys. The testing will be largely confirmatory and will include aging and environmental effects studies under simple and complex loading conditions.

The turbine disk will be made from a wrought Ni-base alloy. Hastelloy X, Hastelloy XR, and Alloy 617 (also a candidate for the turbine inlet shroud collar) have been studied extensively in simulated gas-cooled reactor environments.

Turbine and Generator Surface Engineering/Coatings Test Program

Thermal barrier coatings (TBC) have been developed for turbine blades in recent years to provide some thermal insulation between the operating fluid and the metal substrate. In both aircraft and stationary power generation turbines, the TBC is a multi-layer system consisting of an insulating ceramic outer layer (typically Y_2O_3-stabilized ZrO_2) on top of a metallic bond coat that is applied to the substrate material. Should it be determined that a TBC is required for the NGNP, extensive testing and performance validation will be required. TBC systems have been developed for relatively short time service (thousands or tens of thousands of hours) in an oxidizing environment. Testing will be required to determine if the bond coat material will serve to protect the substrate under NGNP conditions where there may be insufficient oxygen partial pressure to maintain a protective scale.

A9.3.2.1.10 Reactor Pressure Vessel Emissivity. Emissivity data on the various potential candidate materials for the RPV are needed. This is necessary because passive cooling of the RPV by radiation from the outer surface to the air in the cavity between the RPV and surrounding concrete is required during any anticipated accident conditions throughout the life of the reactor. It is therefore necessary to have a stable, high emissivity surface on the external surface of the pressure vessel at elevated temperatures. Depending on the emissivity of the selected base material, it may be necessary to incorporate a high emissivity coating on the outer surface of the RPV.

Early testing to establish emissivity limitations of potential candidate materials and the performance and durability of proposed surface modifications to improve emissivity must be performed to provide design feedback and limitations. Preliminary emissivity testing of the potential candidate materials will be performed to determine the detailed experimental program needed for developing a stable surface with the minimum emissivity required for adequate cooling of the RPV. Concurrent with that testing, a surface treatment/coatings program will be conducted to investigate the efficacy of various potential concepts for either increasing the emissivity of the RPV materials or providing a coating that would have the required emissivity.

A9.3.2.1.11 Internals Project. The existing database for candidate alloys will be assembled, analyzed, and evaluated with respect to the design and operating requirements for reactor internals. Principal topics for review will include high-temperature strength, stability, and long-time performance under irradiation of the materials; effects of impure helium on the mechanical and physical properties of the materials; and codification status, prospects, and needs. For candidate alloys, the status of the joining technology will be reviewed, and the weld metal and weldment database will be collected. The technology behind the weld strength factors under development by the ASME and other international codes will be reviewed in collaboration with activities on design methodology. The neutron fluences accumulated in the metallic core internal materials are expected to be low relative to the tolerances of the structural alloys. Nevertheless, the impacts of these fluences will be reviewed and details developed for confirmatory testing on and evaluation of candidate alloys. Based upon the results of the review, details of the program to evaluate the mechanical and fracture properties of the leading candidates, along with their environmental and irradiation response, will be developed.

A9.3.2.1.12 Intermediate Heat Exchanger and Piping Fabrication Test. The leading potential candidate alloys for these components are listed in Table A9.2. New alloys such as CCA617, Alloy 740, and Alloy 230 will be considered as alternates. An assessment will be undertaken of the potential of C-C composites for use in the compact IHX. The baseline materials data generation program for the IHX will focus on characterizing the IHX construction material as it is influenced by the specific fabrication procedures needed to produce the compact IHX configuration. The material performance requirements will be developed, and a list of leading candidates will be identified. It will be necessary to decide if the fabrication processes should be selected to produce a material of optimum metallurgical condition, or if an off-optimum material condition is satisfactory. At 1000°C, most of the wrought nickel base alloys require relatively coarse grain size for good creep strength, but fatigue resistance is best for fine grain size.

Exploratory testing will be undertaken to establish the effect of fabrication variables on the subsequent creep and fatigue properties. Materials of comparable chemistry, grain size, and processing history will be used to produce data, which can then be used to model the performance of the IHX. It will be determined if the metallurgical state of materials included in the testing program for the core supports and internals are suitable for the IHX. If so, mechanical testing and aging work on materials for the IHX will not be needed. Bench testing small models of the IHX will be performed to add confidence to life prediction methodologies. Metallurgical evaluations will be undertaken.

A9.3.2.1.13 Hot Duct Liner and Insulations Test. Data on the performance of fibrous insulation are needed to ensure that the selected materials are capable of lasting for the life of the plant. The data include: physical properties (heat resistance, heat conductivity, and heat capacity); long-term thermal and compositional stability; mechanical strength at temperature; resistance to pressure drop, vibrations and acoustic loads; radiation resistance; corrosion resistance to moisture and air-helium mixtures; stability to dust and gas release; thermal creep; manufacturing tolerances; and mounting characteristics. The acquisition of these data requires testing of insulation specimens or small assemblies of thermal insulation panels and application of appropriate ASTM standards, which need to be developed. This standards development work will be supported within this program. Moreover, application of current non-destructive evaluation techniques, especially in support of the monolithic insulators, will be included within this test plan. Specific test facility requirements include helium flow, vibration and acoustic test equipment, an irradiation facility, and hot cell. The testing of prototype assemblies is not planned to include neutron irradiation.

A9.3.2.1.14 Valves, Bearings, and Seals Qualification Test. A few valves may be required in the primary or secondary piping systems for this plant, and a flapper valve is used in the SCS. Bearing surfaces exist between the RPV and the core barrel. Seals may be required in a variety of locations. However, insufficient information relating to the specific requirements and issues relating to valves, bearings, and seals is available at this time to initiate a selection activity. It is expected that a materials R&D program covering these areas will be added in later revisions to the plan.

A9.3.2.2 GFR Reactor-Specific Materials

The GFR system features a fast-spectrum, gas-cooled reactor and closed fuel cycle. The GFR reference design is a helium-cooled system operating at 7 MPa with an outlet temperature of 850°C that utilizes a direct Brayton cycle turbine for electricity production and provides process heat for thermochemical production of hydrogen. Through the combination of a fast-neutron spectrum and full recycle of actinides, GFRs will be able to minimize the production of long-lived radioactive waste isotopes and contribute to closing the overall nuclear fuel cycle.

Two alternate system options are currently being considered. The first alternate design is a helium-cooled system that utilizes an indirect Brayton cycle for power conversion. Its secondary system utilizes supercritical CO_2 (S-CO_2) at 550°C and 20 MPa. This allows for more modest outlet temperatures in the primary circuit (~ 600-650°C) and reduces fuel, fuel matrix, and material requirements as compared to the direct cycle while maintaining high thermal efficiency (~ 42%). The second alternate design is an S-CO_2 cooled (550°C outlet and 20 MPa), direct Brayton cycle system. This further reduces temperature in the primary circuit while maintaining high thermal efficiency (~ 45%), potentially reducing both fuel and materials development costs as compared to the reference design and reducing overall capital costs due to the small size of the turbomachinery and other system components.

Much of the GFR balance of plant will be able to utilize materials being evaluated or qualified for the NGNP, though a number of items specific to the operation of the GFR will need to be evaluated. The largest materials challenge for the GFR, however, will be to select and qualify materials for the core and reactor internals structures, since graphite use will be severely restricted due to its heavy moderation of the neutron spectrum. Therefore, alternate, neutronically acceptable materials must be identified that are able to withstand the high GFR temperatures, very high neutron exposures, and are compatible with the coolants envisioned.

The goal of the current materials R&D plan being developed for the GFR is to examine those materials issues that are expected to potentially limit the viability of the overall system, such as neutronically acceptable core and reactor vessel internals materials. Since detailed component designs, particularly for the reactor core and internals, are unavailable at this early stage in the GFR system design, much of the materials research identified in this plan will focus on identification of materials that meet the conditions that will likely envelop specific components. Where components designs are relatively more mature, such as for the reactor pressure vessel, more specific research tasks are identified.

Considering that many of the materials issues faced by the GFR, outside of the core region, are similar to those for the NGNP that is being developed on a significantly more rapid time scale than the GFR, it is assumed that any relevant materials R&D performed for the NGNP will be available and hence will not be repeated for the GFR. The resulting GFR materials R&D plan is designed to provide the information needed on capabilities of current materials or those that can be developed in time to allow a decision on the overall viability of the GFR system concept by 2010. Potential showstoppers will be identified and resolved. The information generated during this stage of the R&D is sufficient for the conceptual design of a prototype. It is not sufficient for the final design of the plant. The extended research required to provide the extensive data bases needed to qualify the candidate materials identified during the GFR materials scoping studies, detailed in this document, will be addressed at the conclusion of these studies and after the decision to proceed to the design phase has been made.

A summary of material R&D plans for GFR materials is provided in the sections that follow. More background and details on these plans are contained in *The Gas Fast Reactor (GFR) Survey of Materials Experience and R&D Needs to Assess Viability*, ORNL/TM-2004-99, April 2004. Estimates of funding required for the GFR materials R&D are included in the GFR Appendix 3.0 of this ten-year plan.

A9.3.2.2.1 Nonmetallic GFR Core and Reactor Internals Materials. Key in-core structures include: plate/block type composite fuels with casing/hexagonal canning and gas tubing, solid solution pellet fuel clad and wrapper, and particle basket designs. Materials must be qualified for the fuel and cladding, for supporting structures, and subassembly structures for control rods and reflectors. The key out-of-core structures include the core barrel and hot gas duct, core support components, the reactor vessel, and cross-vessel components.

For the purpose of this discussion, it is convenient to categorize the ceramics considered for GFR core applications as insulating ceramics, structural ceramics, and structural composites. These classifications are helpful when discussing materials requirement in the absence of solid design data needs such as stress levels and types of loading. The motivation for this classification is driven by the lack of robustness of the current GFR designs.

Insulating ceramics

This class of ceramics has a good knowledge base for application with low mechanical performance requirements (e.g., tensile stress below ~ 1 MPa) and would require the least time for qualification testing. These nonstructural ceramics might be used as spacers, electrical insulators, and/or thermal insulators in the reactor. Common commercial ceramics, such as CaO and MgO, are hygroscopic and, therefore, are not good candidates for applications that may be exposed to water vapor impurities during maintenance operations. Candidate monolithic ceramics with moderate radiation resistance include Al_2O_3, $MgAl_2O_3$, Si_3N_4, AlN, SiC, and ZrC. Required testing for GFR applications would focus on filling gaps in the existing database for thermal conductivity degradation and dimensional stability under irradiation of off-the-shelf materials.

Insulating ceramics can be broken down into the separate functional classes of fibrous and monolithic insulators. Insulation design studies have determined that the best fibrous insulation system for high-temperature, gas-cooled reactor application is the use of Al_2O_3 and SiO_2 mixed ceramic fiber mats (Kth<0.1 W/m-K) contained between metallic cover plates attached to the primary structure that requires insulation. Such insulating materials (particularly Kaowool) were used in the past; however, performance data is incomplete and the operating normal and off normal temperatures (1000 and 1200°C) are aggressive for application of the Kaowool.

Typically, monolithic thermal insulators can have very low (<10 MPa) tensile and (< 50 MPa) compressive strengths; thus, their mechanical performance is quite limited. However, in contrast to fibrous thermal insulation, they will be capable of withstanding much greater loading (e.g., gravity) without significant deformation. Following the example of the previous paragraph, it would not be possible to use fibrous matting to replace thermally insulating floor blocks due to the significant compression that would occur. These monolithic ceramics typically have fracture toughness values of 1 to 5 MPa-m1/2.

The primary work in this area will be the determination of the dimensional stability of select commercially available insulating ceramics under GFR fission neutron irradiation conditions. It is not expected that there will be a spectrum effect on the swelling of these materials except for nitride ceramics, which have enhanced gas production in mixed-spectrum reactors due to a high thermal neutron cross section for gas production by 14N. Therefore, any materials test reactor capable of high-temperature irradiation could be employed for initial scoping studies of non-nitride ceramics.

Structural Ceramics

For many applications in gas-cooled reactor cores, the primary stress of concern is compressive in nature. In this case, structural ceramics, or toughened monolithic ceramics, would be appropriate. Given the performance requirement for a structural ceramic is more challenging than that of insulating ceramics, and given the limited data on irradiation performance of this class of materials, irradiation performance testing of structural ceramics for GFR applications will be longer and more extensive than that for insulating ceramics. This is indicated by a 6- to 10-year lead-time at the end of which the material would be ready to move into a qualification program. There may be off-the-shelf materials appropriate for these applications. Candidate monolithic, structural ceramics include Si3N4, AlN, SiC, and ZrC. Additional candidates include whisker-, platelet-, or transformation-toughened ceramics, such as whisker or platelet-toughened Al_2O_3, Si_3N_4, or AlN, and yttria-stabilized ZrO_2. Typical fracture toughness values for these materials are 5 to 10 MPa-m1/2.

A program to accurately determine the mechanical properties of select structural ceramics with particular emphasis on the statistical nature of failure should be carried out. In addition, an irradiation program will be required to determine the effect of high temperature neutron irradiation on standard thermophysical properties. In addition, non-standard tests such as creep and fracture toughness will be necessary. Depending on the coolant system selected, an environmental effects program will be required to study corrosion and grain boundary effects leading to mechanical property degradation.

Structural Composites

For application where compressive stresses are extreme (>100 MPa), or where tensile stresses are large (>50 MPa), the use of structural composites consisting of woven ceramic fibers and a ceramic matrix will be required. Currently, only SiC-SiC and C-C composites are of sufficient maturity to be considered for application in the GFR timeframe. An example GFR application would be a control rod sleeve or perhaps the core barrel. One essential difference between this class of materials and the structural ceramics is that structural composites would be uniquely engineered for their application and are, therefore, not off-the-shelf products. Structural ceramic composites typically have fracture toughness values of 15 to 25 MPa-m1/2.

To date, C-C composites have found only specialized use as structural materials, and SiC-SiC composites have never been used as a high-stress structural component. The limited application of these materials is due primarily to their relative immaturity, lack of design structural codes governing non-metallic materials, and a conservative approach to structural design.

A comprehensive program including processing of structural composites of appropriate architecture and composition for GFR application will be required. In parallel, a high-dose irradiation campaign must be carried out to determine not only the mechanical property changes under irradiation but also the swelling and thermal conductivity of structural composites under irradiation.

C-C and SiC-SiC composites will be evaluated for use as structural materials for the NGNP. The primary difference between the C-C composites applications in the GFR and the NGNP is that the GFR C-C components will be limited to usage well outside to core to minimize excessive moderation, but even so, they will see significantly higher fluences. Hence, the additional scoping research required for the GFR must address limits of neutron exposure applicable to C-Cs at the temperature of operation and limited studies to ensure the radiation in a fast spectrum is not significantly different that existing database developed primarily in a thermal reactor spectrum.

A9.3.2.2.2 Metallic GFR Core and Reactor Internals Materials. Because the core operates at such high temperatures in normal operation and greatly exceeds even those temperatures during thermal excursions in accidents, ceramics are the prime candidates for core internals. However, based on their high temperature capabilities, refractory alloys could also be considered as alternates, but only if the oxygen content in the system can be maintained well below ~1 ppm. In general, currently available refractory alloys are extremely susceptible to oxidation even at that level; it is understood, however, that the technology is not currently available to maintain oxygen at such low levels in a GFR. Cermets or intermetallic structures have also been suggested, and it may be possible to eventually develop very high temperature versions of more conventional alloys based on Fe-Cr-Ni systems with greatly improved microstructural stability under severe temperature excursions. For example, ODS ferritic-martensitic alloys have shown very good creep resistance at temperatures above 800°C and good structural stability up to 1300°C.

The normal operating temperatures for the three primary out-of-core internals components range from 490°C to 850°C for the reference design. For the lower temperatures, the low-swelling austenitic stainless steels and advanced versions of the 8-9Cr ferritic/martensitic steels are viable classes of candidate materials, and ODS versions of the ferritic and ferritic/martensitic steels produced by mechanical alloying, austenitic stainless steels, and nickel-base alloys are candidates at the higher temperature range.

Metallic materials for the reactor internals will be reviewed comprehensively. This review will be based on a similar review for the NGNP. The existing database for those alloys will be assembled, analyzed, and evaluated with respect to the design and operating requirements presented above. Of particular importance is the review of the irradiation performance data for each of the three main alloy classes. Based upon this review, a limited set of candidate advanced austenitic steels and ferritic/martensitic steels will be defined. Additional property measurement and testing will be carried out on these materials to cover specific aspects of the GFR environment for which the existing database may be inadequate. Examples of this are determination of: (1) the effects of long-term exposure to S-CO_2 on mechanical behavior, (2) long-term structural stability at GFR temperatures, and (3) the impact of off-normal temperature excursions on structure and properties. Irradiation experiments will be designed and carried out to complement and expand the existing database to cover the projected GFR conditions.

Materials deemed appropriate for use at the temperatures and radiation doses of the GFR will be exposed S-CO_2 in the temperature range 350 to 1250°C for time of up to 10,000 h. These tests will establish reaction kinetics, corrosion allowance, and effect on mechanical properties. It is anticipated that, even in the absence of graphite in the core, a helium environment can be established that is within the range of previous test environments. If this cannot be achieved, testing in the proposed helium similar to that stated for S-CO_2 will be required. In addition, the stability of the proposed helium environment will need to be established.

A9.3.2.2.3 RPV Materials Selection and Issues. Based on the currently estimated operating temperatures, 2 1/4Cr-1Mo steel would be the most likely candidate pressure vessel material for the GFR, if design and construction were to begin today and if the RPV was somehow shielded to reduce irradiation exposure significantly. However, given the lead-time available before material selection is anticipated for the GFR system, materials research and development efforts with other ferritic materials should be a definitive part of the GFR program. For example, advances in dispersion strengthened alloys and ongoing research with nitrogen-modified steels are indicating significant promise for extension of adequate creep strength to temperatures of about 800ºC. Alternate pressure vessel materials such as Fe-3Cr-3WV steel should also be considered.

A comprehensive and detailed review of the potential candidate materials for the RPV system will be performed. This review will be based on a similar review for the NGNP but will examine the materials with respect to the different operating temperatures and much higher radiation doses associated with the GFR RPV. A baseline materials test program will be conducted that augments the evaluation of all the basic mechanical and physical properties and microstructural characterization anticipated for the NGNP program.

The anticipated radiation exposure for the GFR RPV is significantly higher than that for the NGNP. Most of the ferritic-martensitic steels discussed earlier have good radiation resistance to embrittlement and swelling in the anticipated temperature regime and to the anticipated radiation dose. However, specific radiation experiments will be required for design conditions to validate that information for the designers and for the regulatory authority. Irradiations would be conducted in a high-flux facility to attain the necessary dose (~15 dpa) in a reasonable time.

A9.3.2.2.4 High Temperature Metallic Components Materials. The candidate materials for the high temperature components within the GFR are very similar to those for the NGNP listed in Table A9.2. Although the service temperatures are lower, the CO_2 service environment for one of the alternate GFR designs presents a major consideration in the selection of alloys. To avoid carburization or metal dusting, it is preferable to have alloys that are high in nickel and chromium. Nickel cladding of the structural materials could be an option. In addition, alloys that are alumina-formers could be considered if they could be heat-treated to form the needed protective coating prior to service.

The research and development plan for the high-temperature GFR materials assumes that the efforts on the NGNP will be directly applicable. The emphasis in the GFR R&D plan should be placed, therefore, on the elements that are different in the two systems. Specifically, the cooling environment will differ between the GFR and the NGNP. The GFR plan should include both helium and CO_2 effects on the mechanical properties. Here, it is assumed that corrosive characteristics of the helium and CO_2 environments will be established as another part of the GFR material research plan. The specific temperatures and times for the different materials should be linked to the components for which the materials are candidates. For example, testing of the nickel base alloys in helium should be extended to 850ºC.

A9.3.2.2.5 Power Conversion Components Materials Selection and Issues. The candidate materials for the various components of the 850°C GFR reference design power conversion system should be essentially identical to those proposed for the higher temperature NGNP. For example, the turbine inlet shroud, which sees the full normal operating temperature in the system, can certainly use the wrought Ni-base alloys (Alloy 617 and Hastelloy X) proposed for the NGNP. In fact, given the lower temperature in the GFR, Fe/Ni-base Alloy 800H might also well be acceptable for this application.

Only the issue of compatibility of materials with S-CO_2 is critical to establishing the viability of existing materials for candidate GFR power conversion systems. To this end, potential materials for the alternate concept power conversion system turbine and recuperator should be exposed to S-CO_2 at appropriate temperatures ranging from 350-650°C for times to ~10,000 h. These tests should be performed to establish reaction kinetics, set corrosion allowances, and to determine the effects of reactions with S-CO_2 on mechanical and physical properties. The results obtained will be important in the materials down-select process.

To this end, three turbine inlet shroud materials, two turbine blade materials, two turbine disk materials, and two recuperator materials should be selected from the preliminary candidate materials discussed earlier and exposed to S-CO_2. The materials tested for the turbine inlet shroud will likely overlap those for the indirect cycle IHX and for the direct cycle high-temperature metallic components. Recuperator materials may also overlap with those for latter alternate cycle.

A9.3.2.2.6 Materials Compatibility Feasibility Considerations for GFR. It is expected that the materials performance needs for the GFR in helium will be largely covered by the work needed for the NGNP and data generated in previous helium-cooled reactor work. The major exception is the demonstration of the feasibility of gas cleanup for a reactor with little or no graphite internals. Tests are needed to demonstrate that under the appropriate helium flow rate and atmospheric ingress, the composition of the helium can be maintained within the compositional range of previous testing. The helium cleanup studies are needed to establish that tested system for control of the helium composition can maintain it within the range of previous testing, and therefore, avoid the need for more extensive testing. These tests will require an appropriately sized, pumped loop with associated chemistry measurement and side stream gas cleanup equipment.

It is envisioned that a small number of the materials chosen for their ability to withstand the higher radiation exposure of the GFR, as compared to the previous High-Temperature Gas Reactors, will need to be evaluated for corrosion performance. These tests will be performed at temperatures up to 50°C that the expected exposure temperatures.

Supercritical CO_2

Because of the lack of materials performance data in S-CO_2 at the pressures and temperatures of interest, an exploratory database must be developed to establish feasibility of this alternate GFR concept. The materials proposed for various components of the S-CO_2 cooled reactor will be evaluated over the expected temperature range. As a minimum, the corrosion performance and mechanical properties of proposed materials in S-CO_2, and the lift-off and plating characteristics of the corrosion products must be determined.

A much more extensive array of specimens will need to be evaluated for the S-CO_2 environment. It is envisioned that these tests will be performed in an S-CO_2 loop for varying times up to 10,000 hours. These tests will provide for a down select of materials capable of surviving in the S-CO_2. This smaller subset of materials will then be evaluated in an in-reactor S-CO_2 loop. This will allow for exposure of the chosen materials to the radiolytic products of the S-CO_2 coolant. In addition, the chemistry of the S-CO_2 will be ascertained to allow for an understanding of the effects of radiolysis on the coolant and to correlate materials performance with environmental exposure.

A9.3.2.2.7 Required HTDM Experimental and Analytical Activities for GFR. The bulk of HTDM needs for GFR will be covered by activities already planned for the NGNP. Additional tasks to establish GFR viability will be needed to assess the viability of ODS, intermetallics, and the ferritic-martensitc alloys for core components and reactor internals where the operating conditions or materials selections are significantly different than the NGNP.

A9.3.2.3 SCWR Reactor-Specific Materials

Supercritical water-cooled reactors are among the most promising advanced nuclear systems because of their high thermal efficiency (i.e., about 45% vs. 33% of current light water reactors [LWRs]) and considerable plant simplification. SCWRs achieve this with superior thermodynamic conditions (i.e., high operating pressure and temperature), by reducing the containment volume, and eliminating the need for recirculation and jet pumps, pressurizer, steam generators, and steam separators and dryers. The reference SCWR design in the U.S. is a direct cycle, thermal spectrum, light-water cooled and moderated reactor with an operating pressure of 25 MPa and inlet/outlet coolant temperature of 280/500°C. The inlet flow splits, partly to a down-comer and partly to a plenum at the top of the reactor pressure vessel to flow downward through the core in special water rods to the inlet plenum. This strategy is employed to provide good moderation at the top of the core, where the coolant density is only about 15-20% that of liquid water. The SCWR uses a power conversion cycle similar to that used in supercritical fossil-fired plants: high-, intermediate-, and low-pressure turbines are employed with one moisture-separator re-heater and up to eight feedwater heaters. The reference power is 3575 MWt, the net electric power is 1600 MWe, and the thermal efficiency is 44.8%. The fuel is low-enriched uranium oxide fuel and the plant is designed primarily for base load operation.

A summary of the materials research and development needed to establish the SCWR viability with regard to possible materials of construction is provided below. The two most significant materials related factors in going from the current LWR designs to the SCWR are the increase in outlet coolant temperature from 300 to 500°C, and the possible compatibility issues associated with the supercritical (SC) water environment. More background and details on these plans are contained in *The Supercritical Water Reactor (SCWR) Survey of Materials Experience and R&D Needs to Assess Viability*, INEEL/EXT-0300693 (Rev. 1), September 2003. Estimates of funding required for the SCWR materials R&D are included in the SCWR Appendix 2.0 of this ten-year plan.

A9.3.2.3.1 Materials for SCWR Radiation Service. Factors that will determine the service life of materials for the SCWR are a combination of corrosion in SC water and radiation effects. The materials of the reactor expected to experience significant neutron displacement doses are: (1) core structural materials, (2) core support structures, and (3) pressure vessel. In the first category are the fuel cladding, fuel rod spacers (spacer grid or wire wrap), water rod boxes, fuel assembly ducts, and control rod guide thimbles. The second category includes control rod guide tubes, upper guide support plate (UGS), upper core support plate (UCS), lower core plate (LCP), calandria tubes, core former, core barrel, and threaded structural fasteners. The RPV includes two low temperature inlet nozzles and two high temperature outlet nozzles. Insulation materials will also be needed for the reactor internals that separate the hot outlet coolant from the colder inlet coolant and for the pressure vessel outlet nozzles.

The above components will be exposed to SC water, ranging from 280ºC at the core inlet up to about 500ºC at the core outlet. The coolant changes from a compressed liquid at a pressure of 25 MPa to a fluid nearly an order of magnitude less dense than ordinary water in traversing from core bottom to top. Doses vary over a wide range - from hundredths of a dpa for the RPV, UGS, UCS, LCP, and calandria tubes to 15-20 dpa for the replaceable fuel assemblies and core former. Under normal operation, the highest temperatures of up to 620ºC will be experienced in the upper part of the core by the fuel cladding, fuel rod spacers, and the core former. As noted above, the temperature at the bottom of the core is 280ºC. Under off-normal conditions, the fuel cladding temperature could reach 840ºC.

Materials qualification will be carried out as a progressive program. The program begins with selection from a range of candidates comprised mainly of Fe-Ni-Cr alloys, followed by materials screening (by testing) to select promising candidates and alloy modification, where necessary for specific conditions. The final step is alloy development in the event that satisfactory alloys cannot be obtained in the earlier stages. The range of compositions within the Fe-Cr-Ni alloys, within which alloys with acceptable mechanical behavior and dimensional stability currently exist or could be developed, may be divided into four broad categories namely, a) austenitic stainless steels, b) ferritic and ferritic-martensitic steels, c) high alloys (Fe < 50 wt.%) and d) Ni-based alloys.

Other materials are also included. For example, for control rod thimbles experiencing temperatures < 300ºC, zirconium alloys are candidates based on their proven performance in currently operating reactors. Consideration also will be given to the potential application of ceramic materials such as SiC composite materials. These materials have been developed primarily for applications requiring high strength at temperatures well above those of the SCWR. Although nothing is known regarding their behavior in SC water conditions, such materials could offer significant advantages over metallic in some cases. Where the application requires it, the outer composite layer could be fabricated with a higher porosity to act as an insulator.

There is insufficient knowledge at present regarding the behavior in SC water of the materials described above to rank them in terms of irradiation-assisted stress corrosion cracking (IASCC). Within each category, numerous compositions have the basic strength and ductility properties to meet the operating requirements of the SCWR. For the reactor vessel, with an operating temperature and irradiation exposure similar to that of current generation pressurized water reactors (PWR), the primary candidate materials for the RPV shell are those currently used in PWRs, namely variants of SA 508 steel. However, because of the high pressure of 25 MPa, a vessel of this material would have to walls about 50% thicker than current practice. Therefore, consideration will also be given to higher strength chromium steels containing solution strengtheners in order to reduce the section thickness.

The materials program consists of two overlapping activities: a) research and development to define prime candidate alloys, and b) a materials engineering design data effort. The former entails a sequenced set of testing and performance evaluation stages in which an initial set of potential candidate materials is reduced to a limited number of prime candidates through testing in increasingly complex and aggressive environments. This R&D program will adopt an integrated theoretical modeling and experimental approach to build the scientific knowledge needed to understand the mechanisms controlling behavior and to provide a rational basis for developing improved alloys. R&D will yield alloy compositions and thermo-mechanical treatments with demonstrated capability to meet the intended service conditions. The second activity involves extensive evaluation and qualification of the prime candidates to develop a materials engineering design database that meets licensing requirements. The product of this phase will be specifications for producing materials in the required product forms; an approved database on properties; the structural assessment methods required to support design, construction, and licensing; and a reliable basis for the prediction of materials performance throughout the expected lifetime.

The behavior of alloys in SC water absent irradiation will be the dominant feature of the initial phases of the R&D program. In the following stages of the program, irradiations of selected materials will be carried out, culminating in irradiations of the best performing materials in SC water. The approach will develop information on the broad response to SC water of the four alloy categories and SiC composites and on the effects of specific compositional and microstructural variations within these classes.

Selection of alloy compositions and conditions for the initial evaluations in SC water will be guided by existing data in three different areas. First, materials will be included for which there is substantial information on behavior in current water reactors. These benchmark materials provide a basis for identifying acceleration of known phenomena or for detecting the development of new phenomena in supercritical conditions. A second source of information to be considered is the experience derived from the operation with a variety of materials in fossil-fired supercritical steam power plants. The third basis for alloy selection is the vast body of data on the effects of neutron displacement damage on materials, which has been developed over the past 30 years of LWR, fast breeder reactor, fusion power, and basic science programs worldwide. This database will provide a rationale for the exclusion of alloys based upon well-documented behavior in terms of radiation embrittlement and dimensional instability under the conditions of temperature, mechanical loading, and neutron dose projected for the core internals. Following alloy selection, the R&D work will be carried out in a coordinated program utilizing existing experimental facilities at various U.S. institutions in close collaboration with similar international efforts.

A9.3.2.3.2 SCWR Materials Compatibility. The mechanisms for environmentally sensitive cracking in water-cooled reactors that have been observed include intergranular stress corrosion cracking, IASCC, and corrosion fatigue. These mechanisms are affected by several variables including metallurgical structure, irradiation inducted grain boundary segregation, and oxidizers/reducers in the aqueous environment.

Several aspects of the water chemistry of the SCWR will impact the corrosion behavior of materials of construction. The concentrations of the transient and stable species due to radiolysis of the water at the higher operating temperature (as compared to LWRs) may well be significantly different.

The chemical potential of oxygen and hydrogen peroxide, which will be significantly different in the supercritical fluid, will affect the corrosion potential of the water. This in turn determines whether magnetite (Fe_3O_4) or hematite (Fe_2O_3) forms and the morphology of these films, which are important to corrosion control on low alloy steels. Note that the low alloy pressure vessel steel will generally not be exposed to an aqueous environment due to the stainless steel weld overlay cladding; however, possible contact of the pressure vessel steel with the SC water will need to be quantified in the safety assessment.

The chemical potential of the hydrogen should change as much as the chemical potential of the oxygen and hydrogen water chemistry may be just as effective in reducing the oxygen content. However, a decrease in the critical reaction rate of the OH radical with hydrogen above 300°C has been observed. Because the radiolysis in the core is kinetically controlled, it might require much more hydrogen to suppress the oxygen and peroxide generation. If too much is required, metal hydriding could occur. The trade-off between these effects will largely determine how much of the LWR and fossil plant water chemistry control experience is applicable to the SCWR. The control of pH, while theoretically possible, may be difficult in practice, especially in the 300 to 500°C temperature range. The pH of the water is important in setting the corrosion potential and rate and, to some extent, the mode of corrosion. A range of pH has been successfully employed in LWRs, and this approach will need to be explored.

The initial focus of the SCWR materials study will be the examination of the likely candidate materials for the reactor internals with respect to their general corrosion resistance and stress-corrosion cracking resistance in SC water. This work will be done initially on unirradiated materials, with previously irradiated materials being added to the sample set as funding and materials availability allows.

A9.3.2.4 LFR Reactor-Specific Materials

LFR systems are lead (Pb) or lead-bismuth (Pb-Bi) alloy-cooled reactors with a fast-neutron spectrum and closed fuel cycle. Options include a wide range of plant ratings, including a long-refueling-interval, transportable system ranging from 50–150 MWe, a modular system from 300–400 MWe, and a large monolithic plant at 1200 MWe. These options also provide a range of energy products. The focus of the U.S. program is on transportable concepts that are small, factory-built, turnkey plants operating on a closed fuel cycle with very long refueling interval (15 to 20 years or longer) cassette core or replaceable reactor module.

Near-term systems are limited by material performance to outlet temperatures of about 550°C. Both Pb and Pb-Bi are coolant options for this reactor with Pb having less material corrosion issues but limiting core ΔT, and Pb-Bi providing more temperature flexibility but raising issues of Po-210 and Bi corrosion. The favorable properties of Pb coolant and nitride fuel, combined with development of high temperature structural materials, may extend the reactor coolant outlet temperature into the 750–800ºC range in the long term, which is potentially suitable for hydrogen manufacture and other process heat applications. In this option, the Bi-alloying agent is eliminated. The required R&D is more extensive than that required for the 550ºC options because the higher reactor outlet temperature requires new structural materials, coolant technology, and nitride fuel development.

More background and details on these plans are contained in *The LFR Coolant & Materials Technology Plan* (Draft), by W. Halsey et al, Lawrence Livermore National Laboratory, June 25, 2004. Estimates of funding required for the LFR materials R&D are included in the LFR Appendix 4.0 of this ten-year plan.

A9.3.2.4.1 General Considerations for LFR Materials Research. Three primary factors will affect the properties and, therefore, the choice of the structural materials from which the LFR components will be fabricated. These are effects of irradiation, high-temperature exposure, and interactions with molten lead or lead-bismuth coolants to which materials in the primary circuit are exposed. An extensive testing and evaluation program will be required to assess the effects that these factors have on the properties of the potential materials for LFR construction to enable a preliminary selection of the most promising materials to be made and to then qualify those selected for the service conditions required. Structural materials needs for LFR systems can be divided into five general classes, those for: cladding, reactor vessel, internals, heat exchangers, and balance of plant.

Two of the three primary considerations for LFR service, irradiation and high-temperature exposure, will largely be addressed with the research planned for crosscutting materials those for the NGNP. While the levels of neutron exposure for the LFR will be quite high (up to 200 dpa) for the metallic components, most of the same mechanisms identified at lower fluences will still be of concern, though at a much greater level. Irradiation-induced swelling of structural alloys at the very high fluences anticipated for LFR internal components will be a much greater limitation for selection and operation of metallic materials. The third primary consideration, materials interactions with molten lead or lead-bismuth coolants is unique to the LFR and described below.

Materials compatibility concerns for structural metal alloys that are in contact with the coolants for the LFR will be very significant. General corrosion, thermal-gradient-induced mass transfer, and even stress corrosion cracking and liquid metal embrittlement are all potential failure mechanisms that must be addressed.

Most of the historical understanding of structural metal in a Pb or Pb-Bi environment is derived from Russian programs. In these programs, significant development was performed to understand and deploy materials and coolant chemistry control schemes for lead-alloy cooled systems. Outside of Russia, the technological readiness level of lead-alloy nuclear coolant technology is at a much earlier development stage, but the partial knowledge of the Russian experience available to the Western technical community has been factored into this materials plan.

Russian lead-bismuth eutectic (LBE) nuclear coolant technology relies on active control of the oxygen thermodynamic activity in LBE to control corrosion and coolant contamination. Within this framework, a series of structural materials were developed and tested in Russia for enhanced corrosion resistance and acceptable lifetime for operating temperatures below 550°C and with fuel cladding temperature below 650°C. Unfortunately, the most advanced Russian alloys, although similar to some Western alloys, have no direct counterpart.

The oxygen control technique, when properly applied, leads to the formation of “self-healing” protective oxide films on the surfaces of the materials in contact with lead-alloys. This is because the base element (typically Fe) and alloying elements (Cr, Ni) of many structural materials have higher chemical affinity to oxygen than to the coolant alloy constituents. Without such protective measures, Fe, Cr, and especially Ni all have non-negligible solubility in lead-alloys that causes severe dissolution attacks.

Oxygen sensors and control systems are thus important components of the reference coolant technology. Alloying materials with elements promoting tenacious and protective oxides (e.g., Si and Al), or treating/coating the surface with appropriate materials for enhanced corrosion resistance have been developed and tested with oxygen control.

For materials used for operating conditions at the high end of the reference technology (above 500°C), it is necessary in some cases to precondition them, i.e., pre-oxidize them so that the kinetics is favorable for growth of protective oxide film during operations. There has been little systematic evaluation and development in this area. For promising candidate materials, especially the ferritic and martensitic steels for fuel cladding and other high temperature applications, preconditioning (e.g., hot dipping in oxygen saturated LBE bath) tests and subsequent corrosion testing in lead-alloys needs to be performed.

Using steels as the main structural materials, the existing LBE technology requires a proper control of the oxygen level to mitigate the steel corrosion problem. Under this framework, if oxygen is depleted, liquid metal corrosion via dissolution attack, and possibly liquid metal embrittlement, can occur. However, at high temperatures in Pb, oxidation kinetics may be accelerated too much and become detrimental. Within this higher temperature range, the mechanical properties of some refractory metals and alloys improve but oxidation problems compound (e.g., internal oxidation of Nb). As such, oxygen-free coolant technology may be needed for high temperature reactors.

It will also be very important to assess weight loss by corrosion. Temperature gradient mass transfer will likely be an important phenomenon in these systems, and experiments should be designed specifically to investigate it. In a system with a temperature difference and with alloy constituents that are soluble in the coolant, it is possible to dissolve from the higher temperature regions and reprecipitate on cooler regions. Because there is a temperature gradient, equilibrium levels could never be established in the coolant. Further, a process unavoidably transfers mass from one part of the system to another. This would occur in addition to other forms of corrosion. In some liquid metal systems, temperature gradient mass transfer has turned out to be the primary issue, even leading to complete blockages in some cases. Test loops with higher temperature and lower temperature sections and appropriate specimens in each region would be needed to assess this issue.

Recent development of lead-alloy spallation target and coolant technology worldwide for accelerator driven systems (ADS) has advanced the state of the art in the West considerably. There is now a substantial amount of experimental evidence that the main features of the Russian l LBE nuclear coolant technology are valid for forced circulation in small to medium loop type systems. Corrosion tests by various international groups indicate that there are qualified structural materials (US, European and Japanese) for the temperature and flow conditions of the Russian reactors. However, to achieve the high potential aimed for in the advanced reactor system concepts, a significant amount of R&D is needed in the areas of materials and coolant chemistry control.

A9.3.2.4.2 Cladding and Core Internals Materials. Cladding material for LFR systems must be compatible with metal or nitride fuel; corrosion resistant in lead or lead-bismuth coolants; and have adequate strength, ductility, toughness, and dimensional stability over the operating temperature range and to doses up to 200 dpa.

Because of the desire to operate to high dose, ferritic-martensitic steels are the primary candidates for cladding in the lower temperature LFR. Because of the extensive work on HT9 for the earlier Liquid-Metal Reactor (LMR) program, for lower temperature LFR systems, HT9 is the initial reference cladding material. However, other steels, as discussed below, offer substantial strength and toughness advantages over HT9, and they will probably perform better.

The corrosion resistance of HT9 or any other ferritic-martensitic steel still needs to be proven before it is chosen as the cladding. Both Russian experience and preliminary U.S. corrosion studies indicate that elevated silicon levels may be required to provide adequate corrosion resistance when using oxygen control as the method for cladding corrosion protection. Additionally, earlier U.S. work has indicated that the formation of intermetallic or nitride surface layers based on Zr, Ti, and/or Al may provide satisfactory corrosion resistance. If alloys with higher silicon are required, the irradiation test base must be established for the new higher silicon alloys.

The martensitic steel HT9 was developed by Sandvik, Sandviken, Sweden, for the power-generation industry in the 1960s. It was introduced into the U.S. fast reactor and fusion materials programs in the 1970s. However, since that time, several improved ferritic/martensitic steels have been developed for the power-generation industry that are significant improvements over HT9. For these newer steels, no lead corrosion data exist, and limited irradiation data exist. However, it would not be expected that these steels would behave differently from steels for which more extensive data are available (HT9, EM12, FV448, 1.4914, etc.). Fairly extensive irradiation data were developed in the U.S. fusion materials program on modified 9Cr-1Mo (T91 in Table A9.2), a second-generation steel. T91 showed significantly improved irradiation resistance compared to that of HT9 primarily because of the lower carbon concentration in T91. In particular, under irradiation conditions where HT9 develops an increase in the ductile-brittle transition temperature of 120-150ºC, the modified T91 developed a shift of only 52-54ºC. For the very high neutron exposures anticipated for some LFR components, the reduced radiation sensitivity may be critical.

Other candidate materials that emerged from the U.S. fusion materials program include the reduced-activation 9Cr-2WVTa. Extensive irradiation testing of this steel showed even more improvement than T91 in irradiation resistance compared to HT9. These results are indications that, although HT9 can and should serve as a reference material for potential ferritic/martensitic steels, given the irradiation experience available, there is every indication that better steels than HT9 are available and should be exploited if their corrosion resistance is sufficient.

Based on the observations on the 9Cr steels T91 and 9Cr-2WVTa, the third-generation steel NF616 (a 9Cr-0.5Mo-1.8WVNB steel) may offer the same possibility of improved irradiation resistance plus better elevated-temperature strength than either of these two steels. One potential problem with the 9Cr steels is corrosion resistance, which may indicate the need for a higher chromium concentration. Therefore, another third generation steel HCM12A (a 12Cr-0.5Mo-1.0WVNbN steel) should be given consideration. To obtain further significant improvements in high-temperature creep strength from ferritic steels, ODS steels will likely need to be evaluated (see Section A9.3.2.3 on SCWR RPV Internals Materials Selection and Issues for more information on ODS steels).

Qualification of any of these materials requires establishing both corrosion resistance and acceptable mechanical performance and dimensional stability. Corrosion testing of all of the ferritic-martensitic steels is important in increasing the potential operating temperature of LFR systems. To increase corrosion resistance, the possibility exists to coat HT9 or another steel in a manner that provides corrosion protection but maintains the acceptable mechanical and dimensional stability performance. Coating and surface modification technology is an important component of the cladding (and core internals) development program and will need to be evaluated, particularly for the higher desired operating temperatures.

For significantly higher temperature (800°C) applications, steels are not likely to be successful as cladding materials. For the higher temperature applications, ceramics, refractory metals, or coated refractories may be necessary. For these high-temperature candidates, the existing materials database comes from the fusion and space programs.

Based on development work in the fusion program and early promising results in lead corrosion tests, SiC and SiC composites would be primary candidates for 800°C application although high dose radiation resistance, cost, and fabricability are still major, open issues. Tantalum alloys are also expected to be resistant to lead corrosion although they may not be adequate from a neutronics standpoint.

Core internals include ducts, grid plates, core barrel, and other piping. In lower temperature LFRs, these can be constructed of either ferritic-martensitic steels for higher dose components or austenitic stainless steels for lower dose components. Advances in structural steels will allow operating temperatures to rise above 550°C, but steels available at present will not support 800°C options. The only alternative steels presently on the horizon for possible 800ºC operation are the ODS steels (see above), but they are still in an early development stage. For the 800°C options, new classes of refractory metals or ceramics are likely to need to be developed. The requirements for internals are very similar to those of cladding with the exception that core internals do not interact with fuel and will operate at lower temperatures and doses than the cladding.

Because transitions between ferritic-martensitic and austenitic materials may occur, properties of welds will also be important for some core internals applications. For ferritic-martensitic components, the candidates are the same as for cladding. For austenitic components where the neutron exposure is low enough to avoid the inevitable swelling that occurs at high doses, cold-worked 316 stainless steel is the primary candidate, with 304 also a nearer term possibility. 316 and 304 have an established mechanical properties and irradiation performance databases. Corrosion resistance of 316 and 304 in lead alloy coolants still needs to be proven. If the corrosion resistance is inadequate, then a complete corrosion, mechanical properties, and irradiation performance database will need to be developed for alternate candidates. For both ferritic-martensitic and austenitic materials, an option would be to coat a material in such a manner that corrosion protection is afforded without loss of mechanical properties or irradiation stability.

A9.3.2.4.3 Materials for LFR Heat Exchangers. Heat exchanger materials must have good corrosion resistance in lead alloy coolant, particularly given the thin sections typically employed for such applications. Corrosion test requirements are similar to those for other core components, but without the requirement for radiation resistance.

For process heat applications associated with high temperature LFRs, an intermediate heat transport (IHT) loop is probably needed to isolate the reactor from the energy converter for both safety assurance and product purity. Heat exchanger materials screening will be needed very early in the program for potential intermediate loop fluids, including molten salts, He, $CO_{2,}$ and steam. For interfacing with thermochemical water cracking, the IHT loop will interface with the chemical plant fluid HBr plus steam at 750ºC and low pressure. For interfacing with turbomachinery, the IHT loop interacts with working fluid options such as S-CO_2 or superheated or supercritical steam.

Corrosion resistance for candidate heat exchanger materials must be established. This may include corrosion resistance to lead alloys, high temperature S-CO_2, aqueous HBr solutions, and molten salt. Decisions on establishing this aspect of the LFR materials program will require better definition of system requirements.

A9.3.2.4.4 Materials for LFR Balance-of-Plant Materials. For lower temperature LFRs, the energy production side is likely to be either a Rankine cycle or a Brayton cycle using S-CO_2 as the working fluid. No development is needed for the Rankine cycle as this is commonly used in commercial energy production. Qualified materials for a supercritical Brayton cycles do not exist. If the proposed Ca-Br cycle is selected for hydrogen production, materials qualified for HBr acid use will be chosen.

A key unknown is corrosion resistance in S-CO_2 for a Brayton cycle. Another is fabricating joints between heat exchangers and bromic acid containing piping.

A9.3.2.4.5 Expected Research, Testing, and Qualification Needs for LFR Materials.

Survey and Selection of Candidate Cladding, Duct, and Structural Materials

The objectives of this area include:

- Identification of materials that make the LFR concept feasible
- Early indication of materials behavior or characteristics that limit in-service conditions for LFR components

Candidate materials have been and will be continue to be selected based on literature survey and investigation of materials usage in industrial application. Materials will be screened for adequate mechanical performance, corrosion resistance, and fabricability. Testing will take place over the range of temperatures, flows, and stresses expected in the LFR system. The materials of interest will be different for the lower temperature (550°C) and higher temperature (800°C) versions. For long-life cores, there is a strong need for accelerated materials testing coupled with benchmarked materials performance modeling to reliably predict lifetime performance. For cladding, compatibility with Pb/LBE on the coolant side and metal or nitride fuel on the fuel side is required. Weight loss under typical temperature, coolant chemistry, and coolant velocity conditions must be ascertained, as must general corrosion. Weight loss as a function of exposure time in lead alloy is required for all candidates. Stress corrosion cracking and liquid metal embrittlement resistance must be demonstrated.

Lead/LBE Corrosion Testing of Candidate Cladding, Duct, and Structural Materials

The objectives of this area include:

- Acquire corrosion performance and properties data for candidate materials for support of conceptual and preliminary design efforts
- Determine corrosion-based limiting conditions of operation for selected materials

Lead/LBE corrosion properties of candidate materials will be investigated under LFR-relevant coolant conditions of chemistry, flow, and temperature. These tests will be conducted using various techniques and facilities, but most notably by using the DELTA loop at Los Alamos National Laboratory. Therefore, the testing will be coordinated in a long-term experimental program that includes development of Pb/LBE technology using the loop facility.

Irradiation Testing of Candidate Cladding, Duct, and Structural Materials

The objectives of this area include:

- Acquire irradiation performance and properties data for candidate materials for support of conceptual and preliminary design efforts
- Determine irradiation properties-based limiting conditions of operation for selected materials

Candidate materials will be irradiated under fast spectrum conditions at LFR relevant temperatures and stresses. Following irradiation, materials will be evaluated to determine mechanical properties, microstructural evolution, and corrosion resistance. These efforts are will be performed as part of a larger materials development and assessment activity within the Generation IV program. As part of the LFR-specific work scope, screening studies may be performed using high-energy ion beams to induce irradiation-damage microstructures in samples that can then be characterized and tested for corrosion properties.

High-Temperature Design Methods

Design methods will be evaluated and extended to cover the temperature and stress regime of the LFR. Developing high temperature design methods is expected to be addressed within the crosscutting Materials R&D.

Materials Modeling

The objectives of this area include:

- Develop mechanistic models of phenomena that control materials behavior in LFR environments

- Use mechanistic materials behavior models to better understand the phenomena that control materials behavior in LFR environments for the purpose of informing design efforts

Advanced, mechanistically based models for irradiation performance and corrosion of materials in Lead/LBE will be developed. These developments will need to be coordinated with related activities to be addressed in Crosscutting Materials R&D.

A9.3.3 Materials for Energy-Conversion System

The various approaches for energy conversion currently being considered within the Generation IV reactors include both electrical generation and use of process heat for hydrogen production. While many of the materials issues for electrical generation are similar to those in the fossil fuel industry, the same cannot be said for hydrogen production. The multiple approaches for nuclear hydrogen production include the use of thermo-chemical separation and thermally assisted electrolysis (high temperature electrolysis [HTE]) as the two leading candidate processes. Both of these approaches will have significant materials challenges including high-temperature structural stability, stability and effectiveness of special functional materials for catalysis and separation technology, thermal barrier materials, and materials compatibility with a variety of heat-transfer media and process-related chemicals.

Of particular concern are the very high-temperature heat exchangers envisioned both on the reactor side and the hydrogen production side of the process-heat transfer loop as well as the lower temperature heat exchangers used within any chemical separation system. The combination of high-temperature operations and simultaneous exposure to multiple process and heat transfer fluids will present significant challenges to maintain the integrity of the thin sections inherent in heat exchangers.

While some of the requirements for the high-temperature materials will be addressed as part of the crosscutting task described in Section A9.3.2 or within the R&D identified within the individual reactor systems, the remaining specialized materials requirements for energy conversion systems will need to be addressed separately. Those tasks that address the generation of electricity will continue to be conducted within the Generation IV Program itself. Those tasks that will address the production of hydrogen will fall under the newly established NHI Program. The extensive systems description of the nuclear hydrogen production processes and more background and details on the materials plans for the NHI Program are contained in the Materials Requirements for Nuclear Hydrogen Generation Systems (Draft), by W. Corwin et al, Oak Ridge National Laboratory, September 21, 2004. A brief summary of the highest priority NHI materials research is included below. Schedules and funding estimates for the NHI research are currently being developed and will be included in later versions of this document.

A9.3.3.1 Summary of High Priority Materials Research Areas for Nuclear Hydrogen Production

A wide range of materials research areas have been identified that will need to be addressed before the three Generation IV systems proposed for nuclear hydrogen production can be deployed. This research covers issues of materials compatibility, high-temperature strength and stability of materials, and fabrication technologies. However, a number of key areas were identified as particularly high priority items that must be addressed early in the program. These areas are summarized below. Obtaining the candidate materials identified and evaluating them under the environmental conditions that will envelop their service will comprise the next steps for the NHI materials program.

A9.3.3.1.1 High-Priority Materials R&D for the SI System. Three areas of materials compatibility research were identified for the SI system as very high priority issues. These include the screening of materials for service in both the concentrator and vaporizer portions of the sulfuric acid concentration and decomposition section and those used in the reactive distillation column of the hydrogen iodide decomposition section. At present, candidate materials have been identified for these service conditions, but the environments are known to be extremely aggressive and performance of even the most promising materials is not adequately established to ensure system viability.

The one additional area of high priority research identified for the SI system was the assessment of high-temperature inorganic membranes for separation of decomposition products of sulfuric acid to potentially reduce peak required temperatures, and associated structural materials requirements, in the hydrogen generation plant and the nuclear reactor providing the process heat.

A9.3.3.1.2 High-Priority Materials R&D for the Ca-Br System. One area of materials research judged to be of high priority for the Ca-Br systems is that of corrosion screening of the materials for the internal heat exchanger within the reaction beds. The wide range of high operating temperatures and widely varying reactants in which these heat exchangers will operate as the beds change from modes of production, where HBr is replaced by steam, to regeneration, where bromine is replaced by pure oxygen, will create a significant challenge for the heat exchanger materials.

Investigation of the corrosion resistance of materials for the heat exchanger that will cool the process stream from the reaction beds prior to its introduction into the plasmatron, where the combination of the HBr, bromine, and water must be accommodated, is also a very high priority.

The final area that may or may not be a high priority for the Ca-Br system is that of the vessel materials that will enclose the reaction beds. If the decision is made to internally insulate this vessel, the reduction in temperature will allow the use of nickel-clad, low-alloy steel. However, if the vessel is not insulated from the operating temperatures required for the bed, the corrosion resistance of the limited materials identified as possible candidates for that application will need to be screened as a high priority task.

A9.3.3.1.3 High-Priority Materials R&D for the HTE System. Several areas of research were identified for the HTE system that will need to be addressed to determine if materials are available to enable higher efficiency operation and improved economic viability. These include materials for metallic interconnects in the electrolytic cell as well as materials to enable higher temperature operation of the steam-hydrogen separator and the recuperators for hydrogen and oxygen cooling. Since alternate approaches are available, such as ceramic interconnects or partial adiabatic cooling, these issues should not affect the operational viability of the system and hence, were not judged to be a high technical priority. If the economic consequences of using the more expensive or less efficient alternate approaches are later judged to affect the overall viability of the system, these issues should be given a high priority in the NHI materials program.

The only other issue that might also be raised to a high priority on economic grounds is the investigation of the use of organic membranes to enable nearly isothermal (and hence much more economical) separation of hydrogen from steam in the output stream from the cell.

A9.3.4 National Materials Program Integration

To help ensure that the materials R&D activities conducted within the overall Generation IV Reactor Initiative form an integrated, efficient program, an additional task is included as part of the crosscutting materials program activities to coordinate, prioritize, and manage materials cross-cutting research with that needed for each specific reactor concept and the energy-conversion system. Principal activities within this task will be to work with the product teams to:

- Develop a detailed understanding of the conditions that all major components and subsystems in each reactor concept and energy-conversion system must withstand (e.g., temperature, irradiation dose, corrosive media, etc., and their combinations);

- Collect and evaluate existing related data from domestic and foreign sources to determine deficiencies in materials data or capabilities;

- Provide cross-platform guidance to ensure appropriate materials R&D is performed in support of each reactor concept, with minimum overlap and no technical voids;

- Ensure that the cross-cutting materials research provides needed and useful information that can be applied to support all reactor concepts; and

- Help ensure that an integrated materials research program is developed, prioritized, and implemented to address the materials needs of the overall Generation IV Reactor Initiative.

The major products of this task will be to provide initial and regularly updated reports assessing potential materials for use in all Generation IV reactor concepts and providing recommendations for reactor-specific materials screening and evaluations to identify viable candidate materials.

A9.4 INTEGRATED MATERIALS PROGRAM COST AND SCHEDULE

A9.4.1 Materials Program Budget

Only the costs associated with the Materials Crosscutting Tasks are included in Table A9.4. Costs for materials activities associated with the specific reactor concepts and the Nuclear Hydrogen Initiative will be funded by those activities and are delineated elsewhere.

Table A9.4 Funding Requirements for the Generation IV Materials Crosscutting Task

Task	FY-05	FY-06	FY-07	FY-08	FY-09	FY-10	FY-11	FY-12	FY-13	FY-14	TOTAL
Materials for Radiation Service	391										
Materials for High-Temp Service	195										
Microstructural Modeling	80										
High-Temp Design Methodology (a)	278										
System-Specific Materials (b)	119										
National Materials Program Management	500										
TOTAL	**1,563**										

(a) Detailed required materials database development to be provided under Materials for High-Temperature Service task

(b) Primary funding included in specific system and NTD budgets, only coordination funding shown

A9.4.2 Schedule for the Integrated Generation IV Materials Program

Within the ten years addressed in this plan for the National Materials R&D program within the Generation IV Initiative, it is expected to:

- Complete an assessment of cross-cutting and reactor-specific materials for use in all Generation IV reactor concepts to identify viable candidate materials;
- Complete the initial development of a comprehensive irradiation-effects materials database for materials needed for radiation service in Generation IV reactors;
- Complete initial development of a comprehensive high-temperature materials properties database to support the design, use, and codification of materials needed for Generation IV reactors;
- Complete adequate qualification of the materials to be used in the NGNP reactor to enable the design and ordering of all major components and subsystems;
- Complete initial development of an improved HTDM that will support design, use, and codification of materials needed for Generation IV reactors;

- Complete development of an interim comprehensive model for predicting long-term properties of materials needed for Generation IV reactors as a function of thermal and irradiation exposure; and

- Interface with GIF and relevant domestic and foreign materials research programs to optimize the effectiveness of materials R&D plan

The anticipated deployment of the NGNP in 2017 will require a strong acceleration of the materials qualification needed to enable design and ordering of long-lead components by about 2010. As a result, a major focus of materials research during the next ten years will be on the qualification of commercial and near-commercial materials and the related HTDM needed to specify and order those components. Parallel studies on materials for other reactor concepts will both take advantage of the accelerated work for the NGNP and examine additional materials under other conditions where the NGNP materials studies are inadequate or inappropriate for their conditions. To help level required resources to the extent possible, the additional studies on materials for other reactor concepts will generally increase in scope as portions of the NGNP-related materials studies are completed.

A9.4.3 High Level Program Milestones for the Integrated Generation IV Materials Program

The high-level milestones of the ten-year plan are as follows:

- Complete establishment of an initial database within Generation IV Materials Handbook for candidate materials for high-temperature and radiation service for all Generation IV reactor systems 9/05

- Complete establishment of low-flux RPV irradiation facility 9/06

- Complete population of Generation IV Materials Handbook with the historical data available and initiate additions of advanced materials data and new data developed in the Generation IV Program 9/06

- Provide interim constitutive equations for 9 Cr, Grade 92 steel and Alloy 617 to aid in Code development and design studies 9/06

- Complete initial assessment of candidate graphites for irradiation service in the NGNP reactor 9/07

- Complete preliminary assessment of candidate materials for high-temperature and radiation service for all Generation IV reactor systems and issue recommendations for final qualification 9/08

- Recommend interim unified constitutive equations for selected Generation IV materials 9/08

- Prepare a report on the results of comprehensive modeling of radiation-induced microstructural evolution in the primary Generation IV candidate structural materials and identify areas for further model development 9/09

- Recommend final revised simplified methods for satisfying strain limits and creep-fatigue criteria in high-temperature structural design 9/09

- Complete development of materials design data needed to order major NGNP components — 9/10
- Finalize constitutive equations for all key Generation IV structural metals — 9/10
- Provide revised design basis for Generation IV materials in database — 9/11
- Validate final simplified design rules for ratcheting and creep-fatigue damage for Generation IV materials — 9/11
- Prepare final report on micromechanical models used to predict relationship between microstructure and mechanical properties in structural materials for use in the Generation IV reactor program — 9/12
- Resolve identified shortcomings, issues, and regulatory concerns in high-temperature structural design methodology — 9/13
- Recommend thermal-striping assessment guidelines — 9/13
- Provide final design basis for Generation IV commercial materials in database — 9/14