U.S. GENERATION IV REACTOR INTEGRATED MATERIALS TECHNOLOGY PROGRAM

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An integrated R&D program is being conducted to study, qualify, and in some cases, develop materials with required properties for the reactor systems being developed as part the U.S. Department of Energy's Generation IV Reactor Program. The goal of the program is to ensure that the materials research and development (R&D) needed to support Gen IV applications will comprise a comprehensive and integrated effort to identify and provide the materials data and its interpretation needed for the design and construction of the selected advanced reactor concepts. The major materials issues for the five primary systems that have been considered within the U.S. Gen IV Reactor Program—very high temperature gas-cooled, supercritical water-cooled, gas-cooled fast spectrum, lead-cooled fast spectrum, and sodium-cooled fast spectrum reactors—are described along with the R&D that has been identified to address them.

KEYWORDS : Materials, Gen IV Reactors, Radiation Effects, High-temperature Strength, Environmental Effects, Codes and Standards, Design Methodology

1. INTRODUCTION

An integrated R&D program is being conducted to study, qualify, and in some cases, develop materials with required properties for the reactor systems being developed as part the U.S. Department of Energy's Generation IV Reactor Program [1]. The goal of the (program is to ensure that the materials research and development (R&D) needed to support Gen IV applications will comprise a comprehensive and integrated effort to identify and provide the materials data and its interpretation needed for the design and construction of the selected advanced reactor concepts.

For the range of service conditions expected in Gen 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 the high-temperature materials design methodology needed for materials use, codification, and regulatory acceptance.

The major materials issues for the five primary systems that have been considered within the U.S. Gen IV Reactor Program—the Next Generation Nuclear Plant [NGNP (a very high temperature gas-cooled reactor)], the Supercritical Water Reactor (SCWR), the Gas-Cooled Fast Reactor (GFR), Lead-Cooled Fast Reactor (LFR), and the Sodium-Cooled Fast Reactor (SFR)—are described along with the R&D that has been identified to address them.

The majority of materials-related research within the Gen IV program is currently focused on NGNP, since it is anticipated to be the first system to be deployed. Major research activities include:

- selection and qualification of graphite for core and core support structures;
- selection and qualification of high-temperature metallic materials for use within the primary circuit, with emphasis on the reactor pressure vessel and the intermediate heat exchangers;
- selection and qualification of structural composites for selected reactor internals that must withstand temperatures in excess of current metallic material capabilities;

- examination of the effects of the environment and long-term thermal aging on candidate materials;
- development of the improved high-temperature design methodology for metallic materials and components needed to meet codification and regulatory requirements;
- support for development and modification of materials and design codes and standards bodies; and
- coordination with and utilization of related materials research activities being conducted by the international Generation IV reactor community.

The other Gen IV reactor systems will take advantage of the materials research being performed for the NGNP where it overlaps their needs, but will also conduct their own more limited and highly targeted research. This research has been primarily focused on high-priority materials questions related to the viability of their concepts. The SCWR materials research has addressed the question of materials capabilities to withstand the particularly challenging effects of coolant corrosion combined with moderately high radiation doses on reactor internals and core support structures. The GFR materials research has addressed the combined challenges of high radiation doses and high temperatures on reactor internals and core support structures that must be constructed from materials other than graphite to minimize excessive moderation of the hard spectrum the concept requires. LFR system materials research must address the particularly difficult materials challenges resulting from a very aggressive coolant corrosion chemistry combined with extremely high radiation doses on reactor internals and core structures. The SFR system has recently been selected as the primary choice for fast reactor development and detailed materials research activities are currently being defined. Primary materials issues for the SFR include choice and qualification of fuel cladding, particularly for transuranic fuels, and qualification of the structural materials already identified for sodium-cooled reactor systems with regard to residual needs for codification and licensability for high-temperature behavior.

Additionally, it is recognized that there are several areas of materials technology that are common to all the nuclear systems being developed. Hence, crosscutting materials tasks have been established, where appropriate. Principal areas of technology being currently examined for crosscutting applications include high-temperature materials, materials for radiation service, microstructural analysis and modeling, and high-temperature design methodology.

2. MATERIALS RESEARCH PLANS

2.1 Materials Crosscutting Tasks

Research tasks in four principal areas have been recognized as crosscutting all reactor systems and are described below. Additionally, the overall management of the U.S. integrated materials technology program is conducted as part of materials crosscutting R&D.

2.1.1 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 Gen IV reactors.

A second important crosscutting feature to be considered is that data on radiation effects must be obtained for all Gen IV reactor concepts from a limited set of operating test reactors and irradiation facilities; for example, HFIR and ATR in the US, HFR in the EU, BOR60 in Russia, and JMTR and JOYO 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 Gen 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 FCC and BCC alloy systems based upon advanced microstructural analysis. Such validated models are needed to provide a sound basis for making extrapolations and interpolations from the experimental radiation effects database. While the development of such models will be conducted within a separate crosscutting task 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 that links the structural materials various Gen IV in-vessel components is that several classes of structural alloys 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 in the long term 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 primary Gen 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.

2.1.2 Materials for High-Temperature Service

In the Gen 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 can advantageously be addressed in a 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, as the first candidate system to be deployed. It follows that in qualification of materials for high-temperature service, early crosscutting efforts will be focused mainly on establishing the activities that will complement those being pursued for NGNP to establish a sound foundation for the multiple Gen IV reactor systems. This will pave the way for the crosscutting activities to gradually increase in scope as portions of the NGNP efforts approach completion.

The crosscutting materials evaluation and qualification activities will be initiated at the early stage of the program and gradually phase in to follow the development of the leading task on NGNP materials qualification. Analysis indicates that 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 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, 2 1/4Cr-1Mo steel (class 1) and modified 9Cr-1Mo-V steel (Gr91). To take full advantage of the potential of the reactor concepts in the Gen IV Initiative, it will be necessary to utilize the advances made in the 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 unirradiated 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 most strongly limiting and, hence, 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.

For the high-temperature materials to be evaluated for out-of-core applications for the Gen IV initiative, the destination of this crosscutting materials research thrust will be 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 oxide dispersion-strengthened (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 Gen 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 those 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, and (3) develop and maintain a *Gen IV Materials Handbook* that will serve as the definitive repository and source for materials data needed to evaluate, design, and license Gen 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, HAZ, and basemetal). Since the crosscutting activity involves Gen 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.

2.1.3 Development of Microstructure-Properties Models

For each design objective described in Section 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 Gen 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 data bases, 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 Nuclear Energy, Science, and Technology 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 Gen 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 Gen 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 promote 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 which 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 for the mechanisms that contribute to high-temperature, time-dependent plasticity (e.g. creep-fatigue, racheting, etc.) and the models describing them for application and insight into the Improved High-Temperature Design Methodology to be developed under a separate crosscutting task; and
- Performance of detailed microstructural analysis, down to the atomic scale, on Gen 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.

2.1.4 Development of Improved High-Temperature Design Methodology

The objective of the High-Temperature Design Methodology Task is to establish the improved and expanded structural design technology necessary to support the codification and utilization of structural materials in high -temperature Gen IV reactor system components. The temperatures and materials requirements of most Gen 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 Gen IV systems. The task will also provide the material behavior (constitutive) models for the detailed inelastic design analysis methods required by Subsection NH for accessing 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 five high-temperature alloys: 304 and 316 stainless steel, 2-1/4 Cr-1Mo steel, Alloy 800H and modified 9Cr-1Mo steel (Grade 91). 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, at most, 34 years whereas Gen IV components are to have a design life of 60 years. Thus, most Gen 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 Gen 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 the crosscutting development effort 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 alloys 617 and 230 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 developments for modified 9Cr-1Mo steel (Grade 92) and Alloy 617 will provide an initial focus for Code work, 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 Gen 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 Gen IV components must be formulated to reflect these behavioral features.

The High-Temperature Design Methodology Task must address several issues. 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 the sense that they will not distinguish between rate-dependent plasticity and time-dependent creep.

The second area, 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 multiaxial 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 in the Clinch River Breeder Reactor Plant licensing process).

Perhaps the most challenging area 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 developed, and quickly, since they are required in the early stages of design.

The final two areas 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.

2.2 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 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 a pace that would significantly precede cross-cutting 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 above are summarized below.

2.2.1 NGNP Reactor-Specific Materials

The U.S Department of Energy (DOE) has selected the Very High Temperature Reactor (VHTR) design for the Next Generation Nuclear Plant (NGNP) Project. The NGNP will demonstrate the use of nuclear power for electricity and hydrogen production without greenhouse gas emissions. The 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 to be the nearest term reactor design that has the capability to efficiently produce hydrogen. The plant size, reactor thermal power, and core configuration will ensure passive decay heat removal without fuel damage or radioactive material releases during accidents. The NGNP Project is envisioned to demonstrate the following:

- 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 about 2020

A summary of material R&D plans for NGNP materials [2] is provided in the sections that follow.

Graphite Materials

Graphite will be used as a structural material and neutron moderator for the NGNP core, and 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 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, including fine grained isotropic, molded or isostatically pressed, high-strength graphite suitable for core support structures, fuel elements and replaceable reactor components, as well as near isotropic, extruded, nuclear graphite suitable for the above-mentioned structures and for the large permanent reflector components. These candidates would meet the requirements of the draft ASTM materials specification for the Nuclear Grade Graphite.

<u>Graphite Selection Strategy</u>

Several candidate graphites have been identified for components within the NGNP (Table 1). The scope of the NGNP graphite program will work collaboratively with activities within the GIF for graphite selection and database. 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.

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

Table 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

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 that cannot be abstracted 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 graphites 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 crystalinity 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/multiaxial strength, Stain to Failure, Elastic Modulus, Poisson's Ratio, Fatigue Strength, Fracture Toughness.
- Thermal Properties – Thermal Conductivity, Thermal Diffusivity, 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 coolant helium (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 sets of 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 High-Flux Isotope Reactor (HFIR) at ORNL. Post irradiation examination (PIE) 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). 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.

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 micro-structural 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 coefficient of thermal expansion (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 B&PV Code (Design requirements for Graphite Core Supports) was issued for review and comment in1992 and no action has been taken on this code since that date. There is activity underway currently (funded by the 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. Moreover, the NRC has indicated that the code should be revised to increase the neutron dose limits to levels appropriate to the PBMR.

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 NDE 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.

High Temperature Design Methodology

Within the NGNP Materials Program both high temperature materials testing and methodology development are both included in the HTDM activities. This includes developing baseline high temperature materials property data and 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 improved high-temperature design methodology are provided in the corresponding crosscutting activity in Section 2.1.4.

Potential candidate high-temperature alloys for service above about 600°C are listed in Table 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. 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 below.

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

Table 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-3.5Nb	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-Nb	S34709	347H SS	870	
18Cr-10Ni-Nb		347HFG	760	
18Cr-9Ni-3Cu-Nb-N		Super 304	1000	
15Cr-15Ni-6Mn-Nb-Mo-V	S21500	Esshete 1250	900	
20Cr-25Ni-Nb		NF 709	1000	
23Cr-11.5Ni-N-B-Ce		NAR-AH-4	1000	
Ni-20Cr-Al-Ti-Y_2O_3	NO7754	Inconel MA 754	1093	
Ni-30Cr-Al-Ti-Y_2O_3		Inconel MA 754	1093	
Fe-20Cr-4.5Al-Y_2O_3	S67956	Incoloy MA956	1100	

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 these alloys 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 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 the DOE programs, the Ultra-Supercritical Steam Generator program at ORNL and the Materials for Energy Research program, at 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.

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. While 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, an initial focal area 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 also for the carbon-carbon (C-C) composites structures of the core. A project team under Section III of ASME is currently undertaking these activities, being 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 are 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 are 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.

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.

<u>Aging Tests</u>

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 for long times in helium. It is expected that aging exposures will be performed to at least 25,000 hours. Mechanical and microstructural properties of bulk and weld structures will be evaluated and the determined experimental properties will also 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 so as to provide for validation of these models. The mechanical and microstructural data will also provide input into code rules for accounting for aging effects.

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.

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 in helium until at least 25,000 h and 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 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 (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.

<u>Evaluation of Helium Environments</u>

The overall stability of the 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.

<u>Helium Loops</u>

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 or 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 so as to determine the compositional range that should be used for the NGNP Materials R&D Program, as well as a review of the existing data/information on the environmental effects of impure helium on Alloy 617 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.

Qualification of Materials for Irradiation Service

Several possible primary coolant pressure boundary systems are envisioned for the NGNP. These could comprise a large RPV containing the core and internals, a second vessel containing an IHX and circulator or a PCU, and a pressure-containing cross vessel 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 either use 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 is in the range of 375-500°C advanced materials will be required. The advanced materials tentatively selected for further investigation for the gas-cooled primary coolant pressure boundary system service are ferritic/martensitic steels, alloyed primarily with chromium and molybdenum. The two most promising classes of commercially available steels are: 9Cr-1MoVNb steels for higher temperature operation and 2.25Cr-1Mo for lower temperature operation.

In order to evaluate the irradiation effects of candidate RPV alloys under the relatively low-flux test reactor 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 facility with the NRC. 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 10^{12} 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.

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 1x10^{19} 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, but 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 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.

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 3. The reason that composites are being considered for these applications is long-term exposure to temperatures greater than 1000°C and the potential for higher temperature short-term exposure. At these temperatures, most metallic alloys are ineffective.

A carbon/carbon (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.

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 are the best choice for a given component based upon the response of the composite to exposure conditions expected within the reactor. For simplicity, C-SiC composites were not included in the table, but were considered to be 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 silicon carbide matrices (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 be exposed to 10 and 20 dpa. Testing will include (but not be limited to) bend strength, dimensional stability, elastic modulus, and thermal conductivity. Based upon these preliminary results, the irradiation stability of SiC-SiC composites versus C-C composites at higher doses should be resolved. This work will answer the fundamental question "are SiC composites potential lifetime control rod materials" in contrast with C-C composites that 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

Table 3. Potential Structural Composite Applications

	Graphite	C-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

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 to be irradiated and to fit into ATR irradiation positions. 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, which would be used for the control rods.

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. 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.

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 *Gen 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 *Gen IV Materials Handbook*.

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

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 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.

Power Conversion Turbine and Generator Project

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 to metal interactions 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 in regard to 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 effects and environmental effects studies under simple and complex loading conditions.

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

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, codification status, prospects, and needs. The status of the joining technology will be reviewed. The weld metal and weldment database will be collected for the candidate alloys. The technology behind the weld strength factors under development by the ASME and other international codes will be reviewed in collaboration with activities on design methodology. The neutron fluences accumulated in the metallic core internal materials are expected to be low relative to the tolerances of the structural alloys. Nevertheless, these will be reviewed and details developed for confirmatory testing and evaluation. Based upon the results of the review, details of the program to evaluate the mechanical and fracture properties of the leading candidates, along with their environmental and irradiation response will be developed.

Intermediate Heat Exchanger and Piping Fabrication Test

The leading potential candidate alloys for these components are listed in Table 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 the compact IHX. The baseline materials data generation program for the IHX will focus on the characterization of the material of construction as it is influenced by the specific fabrication procedures needed to produce the compact IHX configuration. The material performance requirements will be developed and a list of leading candidates will be identified. It will be necessary to decide if the fabrication processes should be selected to produce a material of optimum metallurgical condition or if an off-optimum material condition is satisfactory. Above 850°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. Bench testing small models of the IHX will be performed to add confidence to life prediction methodologies. Metallurgical evaluations will be undertaken.

2.2.2 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 have been 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), reducing fuel, fuel matrix, and material requirements as compared to the direct cycle, while maintaining high thermal efficiency (~ 42%). The second alternate design is a 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 the 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 Next Generation Nuclear Plant (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. Use of alternate, neutronically acceptable materials must be demonstrated at the high GFR temperatures and very high neutron exposures that are also compatible with the coolants envisioned.

The goal of the materials R&D plan needed 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 and viability 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 being assumed that any relevant materials R&D performed for the NGNP will be available and hence will not be repeated within the GFR materials R&D plan. The resulting GFR materials scoping R&D plan contained herein was designed to provide the information needed on capabilities of current materials or those that can developed in time to allow a decision on the overall viability of the GFR system concept. If this concept is pursued, 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 [3] is provided in the sections that follow.

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 as well as 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 which 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 separate functional classes 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 (K_{th}<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, though performance data is incomplete. Moreover, 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 which would occur. These monolithic ceramics typically have fracture toughness values of 1 to 5 MPa-m$^{1/2}$.

The primary work needed in this area is the determination of the dimensional stability of select commercially available insulating ceramics under GFR appropriate 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 ^{14}N. 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 that performance requirement for a structural ceramic is more challenging than those of insulating ceramics, and given the limited data on irradiation performance of this class of materials, irradiation performance testing for GFR applications will be longer and more extensive. This is indicated by the 6- to 10-year lead-time in the above table, 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 Si_3N_4, 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 are 5 to 10 MPa-m$^{1/2}$.

In association with reactor design specialist, 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 as well as 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 will be carried out.

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-m$^{1/2}$.

To date, C-C's 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 the core to minimize excessive moderation, but even so, they will see significantly higher fluences. Hence, the only 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 data base developed primarily in a thermal reactor spectrum.

Metallic GFR Core and Reactor Internals Materials

Because the core operates at such high temperatures in normal conditions, 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 ~1ppm. In general, currently available refractory alloys are extremely susceptible to oxidation even at that level; it is understood that the technology is not currently available to maintain oxygen to such low levels in such a system as the GFR. Cermets or intermetallic structures have also been suggested. 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, oxide-dispersion strengthened (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, but 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 build heavily 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 need to 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 supercritical 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 temperatures and radiation doses of the GFR will be exposed in supercritical 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 supercritical CO_2 will be required. In addition, the stability of the proposed helium environment will need to be established.

RPV Materials Selection and Issues

Based on the currently estimated operating temperatures, 9Cr-1MoV or 2 1/4Cr-1Mo steel would be the most likely candidate pressure vessel materials 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 need to be performed. This review will build heavily 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.

High-Temperature Metallic Component Materials

The candidate materials for the high temperature components within the GFR are very similar to those for the NGNP listed in Table 2. Although the service temperatures are lower, the CO_2 service environment 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. Also, 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 should be placed on the elements that are different in the two systems. Specifically, it will be the environment that 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.

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 supercritical 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 in supercritical 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 effects of reactions with supercritical 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 supercritical 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.

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 feasibility of gas cleanup for the 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 range. These tests will require an appropriately sized, pumped loop with associated chemistry measurement and side stream gas cleanup equipment.

The helium side-stream cleanup studies are needed to establish feasibility of this approach to maintaining control of the helium environment and to determine whether the existing data can support validity of the GFR helium concept or the need for a more extensive test program. 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 HTGRs, will need to be evaluated for corrosion performance. These tests will be performed at temperatures up to 50°C higher than the expected exposure temperatures.

<u>Supercritical CO_2</u>

Because of the dearth of materials performance data in supercritical CO_2 at the pressures and temperatures of interest, an exploratory database must be developed to establish feasibility of the concept. The materials proposed for various components of the supercritical CO_2 cooled reactor will need to be evaluated over the expected temperature range. As a minimum, the corrosion performance and mechanical properties of proposed materials in supercritical 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 be evaluated for the supercritical CO_2 environment. It is envisioned that these tests will be performed in a supercritical 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 supercritical CO_2. This smaller subset of materials will then be evaluated in a in-reactor supercritical CO_2 loop. This will allow for exposure of the chosen materials to the radiolytic products of the supercritical CO_2 coolant. In addition, the chemistry of the supercritical CO_2 will be ascertained so as to allow for an understanding of the effects of radiolysis on the coolant and to correlate materials performance with environmental exposure.

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 efforts 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.

2.2.3 SCWR Reactor-Specific Materials

Supercritical water-cooled reactors (SCWRs) 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), and by reducing the containment volume and eliminating the need for recirculation and jet pumps, pressurizer, steam generators, 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 [4] 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 water environment.

Materials for SCWR Radiation Service

Factors that will determine the service life of materials for the SCWR are a combination of corrosion in supercritical water and radiation effects. The non-fuel materials of the reactor that are 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 reactor pressure vessel (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 inlet coolant, and for the pressure vessel outlet nozzles.

The reactor will operate at a pressure of 25 MPa, above the thermodynamic critical point of water. The above components will be exposed to supercritical water, ranging from the low temperature inlet at 280ºC up to the outlet slightly higher than 500ºC. The coolant changes from a compressed liquid to a fluid nearly an order of magnitude less dense than ordinary water in traversing the core from 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. At the same time the bottom of the core will be at a temperature of 280ºC. Under off-normal conditions the fuel cladding temperature could reach 840ºC.

Materials qualification will be carried out as a progressive program of selection from a range of candidates mainly in the Fe-Ni-Cr alloy system, then screening of materials by testing to select promising candidates, followed by alloy modification where necessary for specific conditions, and alloy development in the event that satisfactory alloys cannot be obtained in the earlier stages. The range of compositions within the Fe-Cr-Ni alloy system 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 silicon carbide 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 supercritical water of the materials described above to rank them in terms of irradiation-assisted stress corrosion cracking (IASCC). Within each category, there exist numerous compositions that 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 be about 50% greater in wall thickness 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 needed materials program would consist 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. Throughout the R&D program, it will be essential to adopt an integrated theoretical modeling and experimental approach in order 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 ensure the viability of the SCWR. It 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 data base 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 supercritical 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 irradiation facilities containing supercritical water. The approach will develop information on the broad response of the four alloy categories, as well as on the silicon carbide 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 supercritical water will be guided by existing data in three different areas. Firstly, 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. The 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.

SCWR Materials Compatibility

The mechanisms for environmentally sensitive cracking in water-cooled reactors that have been observed include intergranular stress corrosion cracking (IGSCC), irradiation-assisted stress corrosion cracking (IASCC), and corrosion fatigue. These mechanisms are affect by several variables including metallurgical structure, irradiation inducted grain boundary segregation, and oxidizers/reducers in the aqueous environment.

There are several aspects of the water chemistry of the SCWR that 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 supercritical 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, to a large extent, 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 supercritical 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.

2.2.4 LFR Reactor-Specific Materials

LFR systems are Pb or 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 has been 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. Pb having probable material corrosion improvements, but limiting core differential temperature, 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. [5,6]

General Considerations for LFR Materials Research

Three primary factors will most affect the properties and 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 for the NGNP. While the levels of neutron exposure for the LFR will be quite high (up to 200dpa) 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 which 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 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, 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). So 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, so there is an "engine" that 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 substantial amount of experimental evidence that the main features of the Russian lead-bismuth eutectic (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.

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 LMR program, for lower temperature (550°C outlet) LFR systems, HT9 is the initial reference cladding material. However, other steels, as discussed below, offer substantial strength and toughness advantages over HT9, and 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, although it would not be expected that these steels will 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 24), a second-generation steel. The 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 9Cr-1Mo 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 fusion program include the reduced-activation 9Cr-2WVTa steel developed in the U.S. fusion materials program. Extensive irradiation testing of this steel showed still 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 mean 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, oxide dispersion-strengthened (ODS) steels will likely have to be produced and evaluated.

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. A final possibility is 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.

Based on the development work in the fusion programs 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 have any interaction 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.

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 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 chemical plant fluid is HBr plus steam at 750ºC and low pressure. For interfacing with turbomachinery, the working fluid options are supercritical 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 supercritical carbon dioxide, and aqueous HBr solutions, and molten salt. Decisions on establishing this aspect of the LFR materials program will require better definition of system requirements.

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 supercritical carbon dioxide 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.

A key unknown is corrosion resistance in supercritical carbon dioxide for a Brayton cycle.

Expected Research, Testing, and Qualification Needs for LFR Materials

<u>Survey and Selection of Candidate Cladding, Duct, and Structural Materials</u>

The objectives of this area include:

- Identification of materials of construction 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 continue to be selected based on literature surveys and investigation of materials usage in industrial applications. 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 lead/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.

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

The objectives of this area include:

- Acquire corrosion performance and properties data for candidate materials of construction 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 LANL. Therefore, the testing will coordinated in a long-term experimental program that includes development of Lead/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 of construction 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 workscope, screening studies may be performed using high-energy ion beams to induce irradiation-damage microstructures in samples that can 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 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.

2.2.5 SFR-Specific Materials

Recent changes in advanced reactor development plans have elevated the SFR to higher priority than had been heretofore the case for the Gen IV reactor family. The ability of the SFR to function as an Advanced Burner Reactor (ABR) for the reduction of actinides, lanthanides, and plutonium has been recognized as having a strong potential to assist in closing the overall fuel cycle from the fleet of light-water-reactors (LWRs) within the U.S. The extensive work done within the U.S. on the sodium-cooled Experimental Breeder Reactor-II (EBR-II), the Fast Flux Test Facility (FFTF), and the Clinch River Breeder Reactor (CRBRP), as well as overseas for the Phenix, Superphenix, Monju, and Joyo reactors, have established an excellent framework for the development of an SFR to reduce the overall high-level waste generated by LWRs. However, the use of the SFR in this role still has a number of materials-related technology challenges that will need to be addressed prior to its successful deployment.

At the current time, the details of the SFR systems to be developed cover a range of sizes—from initial test reactor to demonstration and eventually, commercial reactors—as well as design options. The type of system now being considered as most likely for development is similar to the Power Reactor Innovative Small Module Liquid-Metal Reactor (PRISM LMR) reactor design developed by General Electric, in which the majority of the primary loop piping and heat exchangers are located within the reactor vessel. Details of the reactor hardware, and especially the fuel and fuel-cladding, that would be most conducive to a deep-burn of LWR fuel have yet to be developed. Hence, details of the materials R&D required for such a system has not been addressed, however two general areas of such research have already been recognized: (1) issues associated with high-temperature materials qualification and design methodology required for codification and regulatory approval, and (2) irradiation- and materials-compatibility issues for cladding and reactor internals associated with the new fuels containing high-levels of actinides, lanthanides, and plutonium and their significantly higher burnups anticipated.

Details of the specific materials R&D program needed for the SFR and associated estimates of the funding required for such research have yet to be developed.

High-Temperature Materials Qualification and Design Methods

The design and operation of reactors at elevated temperatures, such as the CRBRP, include time-dependent effects of creep, creep-fatigue, and creep-ratcheting that are significant. These and numerous other failure modes must be considered in the design process. At least two critical reviews of elevated temperature structural design criteria have been conducted by/for the National Regulator Commission (NRC)—the NRC licensing review of the CRBRP for a construction permit in the mid-1980s and an NRC-sponsored evaluation in the early 1990s of materials and design bases issues in American Society of Mechanical Engineers (ASME) Code. Both reviews clearly summarize numerous issues that remain to be addressed.

Based on a review of the material presented by the CRBRP Project [7], the NRC identified concerns in over twenty areas, which were reduced down to nine key issues in the materials and design methodology areas: 1) weldment safety evaluation, 2) elevated-temperature seismic effects, 3) design analysis methods, codes and standards, 4) elastic follow-up in elevated temperature piping, 5) notch weakening, 6) creep-fatigue evaluation, 7) plastic strain concentration factors, 8) intermediate heat transport system transition weld, and 9) steam generator. Some of these issues may have been resolved, but most of them remain today.

Another NRC review [8] was conducted by Oak Ridge National Laboratory (ORNL) in the early 1990s; this review addressed overall material and design bases issues within ASME Code Rules rather than design-specific issues. Twenty-two unresolved issues related to materials and design bases for elevated temperature reactor design and operation were identified. The ten major issues are: 1) lack of material property allowable design data/curves for 60-year design life, 2) lack of understanding/validation of notch weakening effects, 3) lack of validated weldment design methodology, 4-5) degradation of material properties at high temperatures due to long-term irradiation and corrosion phenomena, 6) lack of validated thermal striping materials and design methodology, 7) lack of reliable creep-fatigue design rules, 8) lack of flaw assessment procedures, 9) lack of inelastic design procedures for piping, and 10) lack of rules/guidelines to account for seismic effects at elevated temperature. These concerns are consistent with the review of the CRBRP Project as well

The Department of Energy (DOE) submitted General Electric's (GE) Preliminary Safety Information Document (PSID) for the Power Reactor Innovative Small Module Liquid-Metal Reactor (PRISM LMR) for NRC review in 1986. PRISM is a prime candidate for development into an ABR. The NRC released a Preliminary Safety Evaluation Report (PSEP) for the PRISM LMR in 1994 [NUREG-1368]. The review was based on a conceptual design, for which confirmatory R&D programs had not been completed. A brief summary of the report includes the following issues. The NRC will be required to identify the data, analyses, acceptance criteria, confirmatory research, and program plans in much greater detail in order that the Commission, designers, and the public are more fully aware of the technical regulatory requirements for a prototype demonstration and design certification. Some sources of uncertainty regarding the conceptual design are lack of final design information, unverified analytical tools, limited supporting technology and research, and limited construction and operational experience. Coolant chemistry, service degradation of properties, creep, fatigue, creep-fatigue, and stress rupture were cited by the NRC to be issues, similar to the CRBRP due to high design and operating temperatures and the use of sodium as a coolant. The NRC's evaluation of the general design criteria proposed by GE for the PRISM design identified concerns relevant to the CRBRP that were not included in the PRISM design. The NRC has not yet endorsed ASME Code Case N-47 (now ASME III-NH) and in general, did not accept the application of inelastic stress and strain deformation limits in the initial design evaluations. NRC expects that considerable discussion and correspondence pertaining to application of ASME code and code cases and the inelastic analysis to the PRISM design will be required. This will include addressing all types of time-dependent failure modes relevant to the ABR and identified or to be identified in ASME code and code cases, including those that might occur from degradation of material. Extrapolation of stress allowables to 60 years from current 34 years is an issue as well. The effects of irradiation exposure (damage and dpa limits), particularly for a reactor envisioned with a significantly deeper burn cycle, and the validity of not using in-situ testing/exposure of specimens is also a concern. Long-term metallurgical stability and mechanical properties of 2 1/4 Cr-1Mo tubes in sodium must also be understood for use in the steam generator, if this power conversion system is used. Comparable materials issues for alternative power conversion systems, such as a supercritical CO_2 (S-CO_2) Brayton cycle, will need to be examined, once a decision is made regarding design in this area.

While no significant safety issues were identified based on the PSEP, significant effort and R&D is required to address issues and concerns related to numerous materials issues, design and analysis methodologies, modification/development of ASME code, and subsequent endorsement of such codes or criteria by the NRC in support of the PRISM reactor. While the ABR will not use identical materials as the NGNP, the materials in the ASME Code that are currently codified are limited and will certainly include common material selections. The materials, design methods, and structural testing programs currently underway for the Gen IV and NGNP programs will provide valuable support for similar materials needs for the ABR. Similarly, the active and planned design methods and structural testing for NGNP that are needed for code verification and modification in support of NRC licensing will certainly be applicable to many of the design considerations identified by NRC for PRISM that will still be an issue for the ABR.

Recently, a DOE-ASME collaboration plan for support of codification and regulatory acceptance of Gen IV Reactors was established. The need to build confidence in the regulatory community that the resulting designs will have adequate safety margins is critical, for both Gen IV reactors and the PRISM reactor. One of the tasks in the DOE-ASME collaboration is a review of all the safety issues relevant to ASME Code Rules, and an assessment of how ASME code currently addresses these issues or not. The identification of possible new failure modes relevant to specific Gen IV reactor concepts will also be addressed. The review will serve as a foundation to initiate communications with the NRC on these issues, and facilitate future consultation with the NRC in improving, developing, and confirming design and fabrication procedures, stress and strain limits and material design curves. Numerous other tasks include review and assessment of ASME stress allowables for various code materials, revision of code cases relevant to Gen IV and the ABR, review of material allowables for Mod 9Cr-1Mo which might be considered for use in the ABR, creep-fatigue of existing code materials and desirable code materials, structural testing for validation of design criteria, considering environmental and fluence effects in ASME code, and flaw assessment. All of these tasks will be of considerable value to Gen IV and ABR designs.

Clearly, the need to address materials issues for the ABR similar to those identified by the NRC in review of the CRBRP Project, overall elevated temperature reactors, and specifically the NRC PSER for PRISM will require a combination of materials and design R&D. The challenges for the ABR may not be as difficult as those for the very high-temperature NGNP reactor, but nonetheless, they remain to be addressed. The currently established programs in support of Gen IV and NGNP should be leveraged and/or utilized to address these issues in support of an SFR for use as an ABR.

Core Support and Cladding Materials

The details for the type and burnup levels of LWR-waste-containing fuels envisioned for the ABR are only beginning to be developed and will be highly dependent upon the overall reactor design, reprocessing approach, and fuel form (metallic versus oxide) selected. Hence, it is premature to define detailed materials R&D required for the fuel cladding and the surrounding core support materials. However, considering that the challenges faced by the materials within these components will be outside the range of those already investigated, a initial assessment can be made regarding the type of additional research that will likely need to be made to address the resulting uncertainties.

Given the extensive research and development programs already conducted for the fairly wide array of sodium-cooled reactors, it can be said that ferritic and austentic steels are prime candidates for fuel clad and structural materials in the Advanced Burner Reactor. In particular, the extensive mechanical properties databases, irradiation data over 100 dpa, and established liquid metal compatibility have been developed for candidate ferritic/martensitic steels (e.g. T-91, HT-9, and HCM12A, etc.), austenitic stainless steels (e.g. 316LN), and advanced austenitic alloys (e.g. D9). However, there are still gaps in the databases that must be evaluated under relevant operating conditions for the detailed ABR designs to be develop. The following materials issues related to irradiation service of ABR materials will need to be examined.

- A confirmatory evaluation of interactions and compatibility between clad and fission products, particularly considering the significant departures in likely fuel chemistry compared to historical data, must be performed
- Radiation-induced segregation (RIS) has been examined extensively in stainless steels. However, there are only very limited measurements in ferritic/martensitic steels (due to magnetic nature of material) and D9. This data is needed as the redistribution of elements under RIS can result in precipitation of second-phases, which may impact mechanical properties, particularly considering that fuel rod internal pressures from deep-burn cycles may require improved mechanical strength of the cladding
- RIS can also influence stress-corrosion cracking, which must be evaluated under relevant environments and materials of interest, since the deep-burn cycles anticipated for the ABR may require the selection of a modified fuel clad material compatible with both the coolant and the fission products.
- Modeling of RIS and precipitation effects will be a key element for improved lifetime prediction. Again, this must be done for specific alloys of interest under relevant irradiation conditions.
- Joining of structural materials or clad materials must be evaluated; the impact of irradiation service is of great importance.
- Low-temperature embrittlement must be examined. There are several recent observations of embrittlement at low doses for ferritic/martensitic steels that must be further evaluated.
- In-situ testing of mechanical performance under irradiation is a key step in confirming that the high temperature design methodology developed for ABR designs and materials is sufficient to ensure that all post-irradiation data generated is representative of in-reactor conditions.

While most of the issues described above will require new irradiation experiments to fully assess, significant insight can be gained by a thorough review of the extensive existing information already developed in these areas. Additionally, examination of the materials still remaining from test programs and actual reactor hardware (fuel pins, ducts, spacers, etc.) from the ERB-II and FFTF may provide a very valuable and cost-effective means to address many aspects of the irradiation-exposure issues raised above, since the exposure conditions are, in many cases, extremely relevant. An aggressive approach to preserving and utilizing historically available materials from these two major U.S. reactor experiments should be immediately undertaken.

3. CONCLUSIONS

Based on extensive assessments of materials requirements for the major Gen IV reactor systems that have been considered within the U.S., a description of R&D needs for those systems has been developed. At the current time, the principal activities that are being conducted address the needs of the VHTR reactor system. Decisions regarding the implementation of materials R&D for the SFR are being considered and will likely be implemented in the near future.

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