Rev. 1

ORNL/TM-2004/99

The Gas Fast Reactor (GFR) Survey of Materials Experience and R&D Needs to Assess Viability

GENERATION IV
nuclear energy systems

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LIST OF ACRONYMS

AGCRNR	Advanced Gas Cooled Nuclear Reactor
ANL	Argonne National Laboratory
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
BCC	body-centered cubic
BOP	balance of plant
B&PV	Boiler and Pressure Vessel Code
CV	cross vessel
DBTT	ductile-to-brittle transition temperature
DOE-NE	Department of Energy-Office of Nuclear Energy, Science, and Technology
dpa	displacements per atom
EU	European Union
GFR	gas fast reactor
GT-MHR	gas turbine-modular high temperature reactor
HHT	high temperature helium turbine systems
HTDM	high temperature design methodology
HTGR	high temperature gas cooled reactor
HTTR	High Temperature Test Reactor
IHX	intermediate heat exchanger
INEEL	Idaho National Engineering and Environmental Laboratory
JAERI	Japan Atomic Energy Research Institute
LMFBR	liquid metal fast breeder reactor
LWR	light-water reactor
MIT	Massachusetts Institute of Technology
NGNP	next generation nuclear plant
NPH	nuclear process heat
ORNL	Oak Ridge National Laboratory
ODS	oxide-dispersion strengthened
PBMR	pebble bed modular reactor
PCV	power conversion vessel
PNP	prototype nuclear process heat
R&D	research and development
RPV	reactor pressure vessel
SCWR	supercritical water reactor
VHTR	very high temperature reactor

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EXECUTIVE SUMMARY

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

Two alternate system options are currently being considered. The first alternate design is a helium-cooled system that utilizes an indirect Brayton cycle for power conversion. Its secondary system utilizes supercritical CO_2 (S-CO_2) at 550°C and 20 MPa. This allows for more modest outlet temperatures in the primary circuit (~ 600-650°C), 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 current materials R&D plan being developed for the GFR is to examine those materials issues that are expected to potentially limit the viability of the overall system, such as neutronically acceptable core and reactor vessel internals materials. Since detailed component designs, particularly for the reactor core and internals, are unavailable at this early stage in the GFR system design, much of the materials research identified in this plan will focus on identification 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 is 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 by 2010. Potential showstoppers will be identified and resolved. The information generated during this stage of the R&D is sufficient for the conceptual design of a prototype. It is not sufficient for the final design of the plant. The extended research required to provide the extensive data bases needed to qualify the candidate materials identified during the GFR materials scoping studies, detailed in this document, will be addressed at the conclusion of these studies and after the decision to proceed to the design phase has been made.

The needed materials development tasks, schedules, and costs to assess the feasibility of the GFR are presented in Section 9 of the report. The total cost estimate for viability R&D of the materials needed for the GFR is about $96 million dollars. The costs for the needed work for the GFR are summarized below:

Task	FY05	FY06	FY07	FY08	FY09	FY10	TOTAL
Ceramic Internals	2,000	4,800	6,450	7,400	7,500	6,500	34,650
Metallic Internals	1,100	3,600	5,700	6,800	5,900	4,900	28,000
RPV	900	900	900	900	500	500	4,600
High-Temperature Metallic Components	0	460	700	600	550	350	2,660
Power Conversion System	200	450	750	750	750	300	3,200
Materials Compatibility	0	1,200	3,400	6,200	5,000	2,900	18,700
High-Temperature Design Methodology	50	200	600	1,250	1,350	1,150	4,600
TOTAL	4,250	11,610	18,500	23,900	21,550	16,600	96,410

The funding specifically required for the GFR materials studies can be significantly reduced if (1) existing university facilities are used, (2) the costs are shared with our international GIF partners, and/or (3) the costs are shared with other Generation IV reactor development programs and crosscutting activities. Note that these costs are for “viability” research and development as defined in the Generation IV Roadmap (GIF 2002). Viability research and development examines the feasibility of key technologies and is that R&D necessary for proof of the basic concepts, technologies, and relevant conditions.

While there are significant materials development and qualification needs for the GFR, existing materials have been identified that have the potential to meet the requirements of all the GFR components and subsystems.

ABSTRACT

The GFR system features a fast-spectrum, gas-cooled reactor and closed fuel cycle. The GFR reference design is a helium-cooled system operating at 7 MPa with an outlet temperature of 850°C that utilizes a direct Brayton cycle turbine for electricity production and provides process heat for thermochemical production of hydrogen. Through the combination of a fast-neutron spectrum and full recycle of actinides, GFRs will be able to minimize the production of long-lived radioactive waste isotopes and contribute to closing the overall nuclear fuel cycle. Two alternate system options are currently being considered that utilize a supercritical CO_2 (S-CO_2) Brayton cycle for power conversion to maintain good efficiencies at reduced outlet temperatures. One of the alternate system options incorporates a helium-cooled primary circuit, the other an S-CO_2-cooled primary circuit.

The largest materials challenge for the GFR 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. Much of the GFR balance of plant will be able to utilize materials being evaluated or qualified for the Next Generation Nuclear Plant, though a number of items specific to the operation of the GFR will need to be evaluated.

The materials R&D plan for the GFR will examine those materials viability issues expected to potentially limit the GFR in time to allow a decision on the overall viability of the GFR system concept to be made by 2010. Potential showstoppers will be identified and resolved. The information generated during this stage of the R&D is sufficient for the conceptual design of a prototype. The extended research required to provide the extensive databases needed to qualify primary GFR candidate materials for final design and licensing will be addressed subsequently, during a materials qualification program phase. While there are significant materials development and qualification needs for the GFR, existing materials have been identified that have the potential to meet the requirements of all the GFR components and subsystems.

The total cost estimate for viability R&D of the materials needed for the GFR is about $96 million dollars. These direct costs may be reduced through collaborative research with related domestic and foreign research programs.

1. INTRODUCTION—GFR REACTOR DESCRIPTION

The GFR system features a fast-spectrum gas-cooled reactor (see Figure 1.1) 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 can also provide process heat for thermochemical production of hydrogen. This was chosen as the reference design due to its close relationship with Very High Temperature Reactor (VHTR), currently envisioned as the Next Generation Nuclear Plant (NGNP), and thus its ability to utilize as much NGNP material and balance-of-plant technology as possible. 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.

Since a point design for the GFR does not yet exist, two other options are currently being considered. The first alternate design is also a helium-cooled system, but utilizes an indirect Brayton cycle for power conversion. The secondary system of the first alternate design 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%).

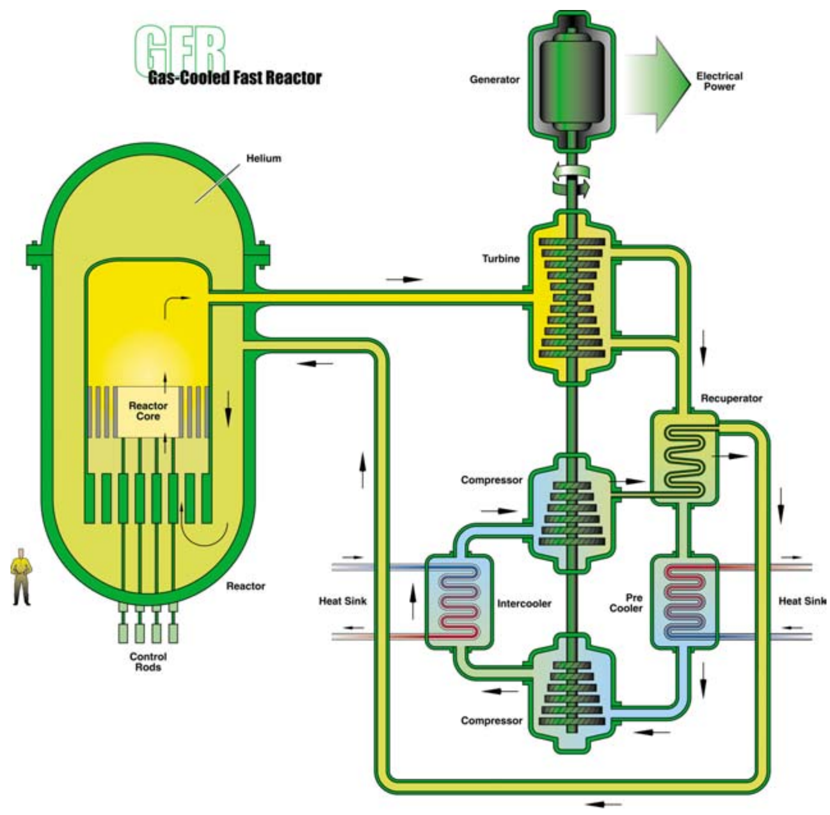

Fig. 1.1. The gas-cooled fast reactor concept

The second alternate design is a S-CO_2 cooled (550°C outlet and 20 MPa), direct Brayton cycle system. The main advantage of the second alternate design is a greater reduction in the outlet temperature in the primary circuit, while maintaining high thermal efficiency (~ 45%). Again, the modest outlet temperature (comparable to sodium-cooled reactors) reduces some of the materials requirements on fuel, fuel matrix/cladding, and materials related to high-temperature operation, but adds complications in the area of materials compatibility. This design has the potential of reducing the fuel matrix/cladding development costs as compared to the reference design, and also reducing the overall capital costs due to the small size of the turbo machinery and other system components. The power conversion cycle is equivalent to that shown in Figure 1.2, where the IHX would be replaced by the reactor and reactor pressure vessel.

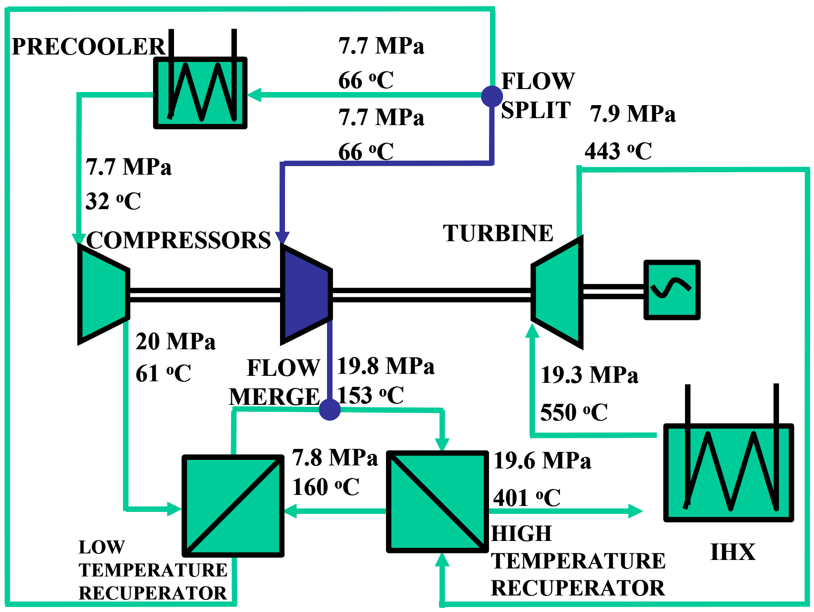

Fig. 1.2. Schematic of the supercritical CO_2 (S-CO_2) cycle.

Much of the GFR balance of plant will be able to utilize materials being evaluated or qualified for the NGNP [1], 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 the use of the normal structural material of choice for thermal gas-cooled-reactor core and internals, graphite, will be severely restricted due to its heavy moderation of the neutron spectrum in the core. Alternate, neutronically acceptable materials that can operate satisfactorily at the high-temperatures and very high neutron exposures anticipated for the reactor core and that are compatible with the coolants envisioned have not been demonstrated.

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. These components choices are highlighted in Figure 1.3.

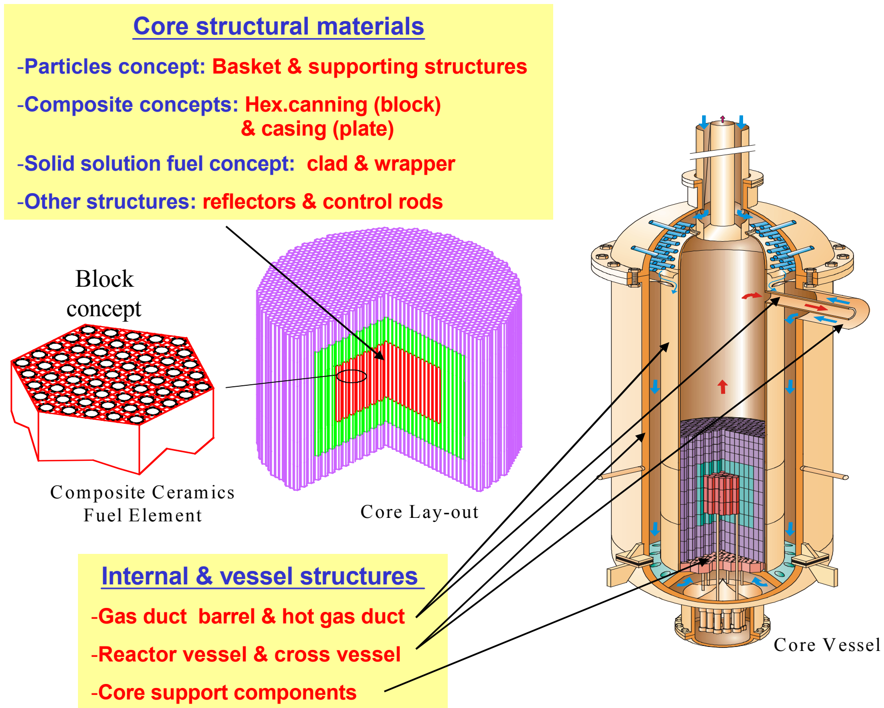

Fig. 1.3. Main components of the gas-cooled fast reactor concept

More details on the GFR reactor designs and associated materials requirements are provide in the a recent report by Kevan Weaver, et al. [2] A summary of target design parameters for the reference GFR system is given in Table 1.1.

Table 1.1. Target Design Parameters for the GFR system.

Reactor Parameters	**Reference Value**
Reactor power	600 MWth
Net plant efficiency (direct cycle helium)	42%
Coolant inlet/outlet temperature and pressure/Helium flow rate	490°C/850°C at 7 MPa, 312.4 kg/s
Core structures temperatures (normal operations)	500-1200°C
Transient temperature in accidental conditions	1600-1800°C
Out-of-core structures	440-850°C, low irradiation exposure, mechanical loading < 50-60 MPa and high useful life (400000 h)
Average power density	50-100 MWth/m^3
Reference fuel compound	UPuC/SiC (50/50%) with about 20% Pu content
Volume fraction, Fuel/Gas/SiC	50/40/10%
Conversion ratio	Self-sufficient (BR~0)
Burnup, Damage (initial values)	5% FIMA; 80 dpa

Alternate designs include the He-cooled, indirect S-CO_2 cycle and the indirect S-CO_2-cooled, direct cycle systems that were mentioned previously.

The goal of the current materials R&D plan being developed for the GFR will be to examine those materials issues that are expected to potentially limit the viability of the overall system. For example, it is not yet known if the materials needed to enable the operation of the GFR core as envisioned in current designs are available. Evaluations of neutronically acceptable materials that must operate under the combination of high-temperature, very high neutron fluence, and environmental interactions with the coolants will need to be made. Materials candidates for such service must be identified along with adequate information on their properties to allow refined preliminary designs to be developed and plans for the subsequent down-select and qualification of the candidate materials to be made. 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 is 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 by 2010. 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 the decision to proceed to the design phase.

References

[1] Next Generation Nuclear Plant Materials Selection and Qualification Program Plan, INEEL/EXT-03-01128, November 2003

[2] The Gas-Cooled Fast Reactor (GFR) Draft Material Requirements for the Material Selection and Qualification Program, INEEL/EXT-04-01606 (Rev 0), February 2004

2. GFR STRUCTURAL CERAMIC CORE AND INTERNALS MATERIALS

2.1 Operating Conditions and General Materials Considerations

Ceramics are being considered for in-core application in the GFR primarily due to their retention of high-temperature properties. Components for which ceramics are the likely option include the reflector, control rod guides, and the upper and lower support plates. Estimates of the temperatures for the various components for each of the design types are provided Table 2.1, and range from as low as 300°C to as high as 1000°C. The temperatures listed could change based on the materials used, the effectiveness of the decay heat removal system, and the core design. For all cases, the expected neutron dose is quite high, exceeding 100 dpa. The wide range in service temperatures will require likely require the use of several different materials as the radiation resistance of ceramic and ceramic composite materials is strongly affected by temperature of service.

Table 2.1. Normal and off-normal conditions for GFR vessel, core, and internals.

Component	Design Option	Normal Conditions		Off-Normal Conditions	Notes
		Temperature	Peak Dose	Temperature	
Fuel Matrix-Cladding	He direct	1200 ¡C	15-20 dpa/yr, total 60 dpa	Up to 1800 ¡C	
	He/S-CO_2 indirect	1000 ¡C		Up to 1600 ¡C	
	S-CO_2 direct	900 ¡C		1100 - 1500 ¡C	It may be possible to use metals in the core, depending on configuration.
Spacers/Wire Wrap	He direct	490-1000 ¡C	15-20 dpa/yr, total 60 dpa	Up to 1600 ¡C	
	He/S-CO_2 indirect	300-800 ¡C		Up to 1400 ¡C	
	S-CO_2 direct	400-700 ¡C		900 - 1300 ¡C	
Fuel Subassembly	He direct	490-1000 ¡C	15-20 dpa/yr, total 60 dpa	Up to 1600 ¡C	
	He/S-CO_2 indirect	300-800 ¡C		Up to 1400 ¡C	
	S-CO_2 direct	400-700 ¡C		900 - 1300 ¡C	
Fuel Subassembly Duct	He direct	490-1000 ¡C	15-20 dpa/yr, total 60 dpa	Up to 1600 ¡C	
	He/S-CO_2 indirect	300-800 ¡C		Up to 1400 ¡C	
	S-CO_2 direct	400-700 ¡C		900 - 1300 ¡C	
Reflector	He direct	490-850 ¡C	Up to 150 dpa	Up to 1100 ¡C	Normal operating temperatures are conservative; the high end may be less.
	He/S-CO_2 indirect	300-650 ¡C		Up to 900 ¡C	
	S-CO_2 direct	400-550 ¡C		Up to 800 ¡C	
Control Rod Guide	He direct	490-1000 ¡C	Up to 200 dpa	Up to 1600 ¡C	
	He/S-CO_2 indirect	300-800 ¡C		Up to 1400 ¡C	
	S-CO_2 direct	400-700 ¡C		900 - 1300 ¡C	
Upper Support Plate	He direct	850 ¡C	Up to 100 dpa	Up to 1200 ¡C	Normal operating temperatures assume the gas is well mixed at the core exit.
	He/S-CO_2 indirect	650 ¡C		Up to 1000 ¡C	
	S-CO_2 direct	550 ¡C		Up to 900 ¡C	
Lower Support Plate	He direct	490 ¡C	Up to 100 dpa	Up to 750 ¡C	
	He/S-CO_2 indirect	300 ¡C		Up to 550 ¡C	
	S-CO_2 direct	400 ¡C		Up to 600 ¡C	
Core Barrel	He direct	490-850 ¡C	80-100 dpa	Up to 1100 ¡C	
	He/S-CO_2 indirect	300-650 ¡C		Up to 900 ¡C	
	S-CO_2 direct	400-550 ¡C		Up to 800 ¡C	
Pressure Vessel	He direct	490-850 ¡C	< 1 dpa to 40 dpa	Up to 1100 ¡C	Dose is dependent on shielding used, and off-normal temperatures can be significantly reduced if insulation is used.
	He/S-CO_2 indirect	300-650 ¡C		Up to 900 ¡C	
	S-CO_2 direct	400-550 ¡C		Up to 800 ¡C	

For the purpose of this discussion, it is convenient to categorize the ceramics considered for GFR application as described in Table 2.2. 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. It is anticipated that a palette of different high temperature materials, each having unique performance requirements, will be needed.

Table 2.2 Maturity of Ceramics for GFR applications

Ceramics Class	Performance/Data Requirements	Maturity Level	Lead-Time for Preliminary Selection
Insulating Ceramics	Low/intermediate	Mature	3-5 years
Structural Ceramics	Intermediate	Adolescent	6-10 years
Structural Composites	High	Immature	10-15 years

Another metric for discussing these materials classes, and choosing among them for GFR applications, is the required fracture toughness for the material. Most engineering alloys such as steel have extraordinary ability to resist unstable crack propagation under load, with fracture toughness values in excess of 200 Mpa•$m^{1/2}$. Following neutron irradiation, the fracture toughness for steels, as with most engineering alloys can significantly drop, though this is not of great concern unless the fracture toughness drop to values below about 30-50 MPa•$m^{1/2}$. Contrast these numbers with the fracture toughness of monolithic insulating ceramics, which have fracture toughness value on the order of 3 MPa•$m^{1/2}$ and its clear that special considerations in design which is required. However, it is possible through incorporation of platelets, transformable phases (~ 7 MPa•$m^{1/2}$), chopped fibers (~ 10 MPa•$m^{1/2}$), or continuous fibers (~ 25-30 MPa•$m^{1/2}$) to increase the fracture toughness of ceramics. In these cases, the incorporation of continuous fibers are what is being referred to as a "structural composite," with the balance of the second-phase toughened materials falling into the "structural ceramic" category. In summary, when considering the ceramic thermophysical requirements for GFR ceramics, the response of the material and choice of material may be driven by the material toughness, which will drive the timescale and cost of materials R&D.

2.2 Status of Potential Candidate Ceramic Materials for GFR Core and Reactor Internals Applications

Insulating ceramics: This class of ceramics has a good knowledge base for application with low mechanical performance requirements (e.g., tensile stress below ~ 1 MPa) and would require the least time for qualification testing. These nonstructural ceramics might be used as spacers, electrical insulators, and/or thermal insulators in the reactor. Common commercial ceramics such as CaO and MgO are hygroscopic and therefore are not good candidates for applications that may be exposed to water vapor impurities during maintenance operations. Many of the alkali halide ceramics are highly susceptible to radiolysis from ionizing radiation with accompanying high swelling. Since residual gamma radiation would be present during cooling and heating operations, these radiolysis-sensitive ceramics would tend to crack and spall easily during service and/or maintenance operations. Radiolysis-sensitive ceramics therefore should be dismissed from consideration. 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.

As will be noted in the irradiation effects section, properties will need to be generated on specific trade-named materials, as there can be considerable difference in as-irradiated property changes for nominally the same materials.

The areas of insulating ceramics may cover a wide range of application from local duct insulation to block insulation at the periphery of the active core. Insulating ceramics can be broken down into separate functional classes fibrous and monolithic insulators. For example, there are many ways to achieve insulation in a reactor vessel such as a meter of graphite (K_{th}> 10 W/m-K) thickness plus 0.2 meter of carbon-carbon composite blocks is sufficient to insulate the lower metallic core support structure from the core outlet gas in a HTGR. However, where room is limited to a few inches of insulation thickness to do the same job, a more efficient form of insulation may be needed. Insulation design studies have determined that the best 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. As example, the pumpable Kaowool temperature limit for continuous operation is 1093°C. Maximum temperature rating is typically 1260°C for the highest performing Al_2O_3 and SiO_2 mixed ceramic fiber mat insulation. Typically, by reducing the fraction of silica in the wool, or through simultaneous reduction of silica and addition of ZrO_2, insulating mats can achieve continuous and maximum operating temperatures of 1300 and 1400°C respectively. High purity alumina mat can achieve operating temperatures above 1500°C. However, these higher temperature mats would not take advantage of previous data and experience gained with the Kaowool product, therefore a premium would be paid for their use.

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-$m^{1/2}$.

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-$m^{1/2}$.

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. However, one key to improving thermal efficiency of power reactors is increasing operating temperatures above the softening point of both standard alloys and superalloys. At these temperatures (>900°C,) the only materials that can be considered are refractory alloys and ceramic composites. [1]

A primary benefit to the use of composites is the inherent ability to design the properties of the systems and their more predictable failure mechanics. For structural applications, the architecture for both SiC/SiC and C/C will need to be three dimensional to avoid the very low inter-laminar shear stresses inherent in 2-D architecture. [2] However, the actual 3-D architecture can vary widely depending on the applications optimizing for strength, stiffness, or thermal conductivity in the most critical orientation. For example, control rod sleeves would likely use a spiral-weave as compared to a balanced or orthogonal weave in shroud or core-block application. It is important to note that due to the limited understanding of the mechanical performance, irradiation behavior, and design rules, each material and architectural variant will be treated on a proof basis. In other words, each material will undergo a complete series of irradiation and performance tests to prove itself, rather than relying on limited testing in support of standard modeling.

Up to the maximum off-normal temperature assumed for the GFR (~1500°C) neither SiC or graphite fiber composites exhibit significant degradation in mechanical properties (excluding oxidation effects.) Both materials have similar decreases in thermal conductivity with temperature, though graphite composites have significantly higher absolute thermal conductivity. The main differences between the systems is the relative maturity of manufacture of the C/C system, allowing more design flexibility and lower cost, and the relative insensitivity to irradiation of the SiC/SiC system at temperatures 300-1000°C. Because SiC composite manufacture is less mature than C/C, the determining factor in selecting the system is essentially economic, related to the up-front cost on deploying SiC/SiC balanced with the potential benefit of a longer-lived or lifetime component.

2.3 Effects of Neutron Irradiation

All ceramic materials, regardless of the classification given above, have very similar behavior under neutron irradiation. One notable exception is graphite, which has very anisotropic swelling behavior that limits its application lifetime. Anisotropic radiation growth in graphite occurs due to preferential nucleation of point defect clusters on basal planes that leads to significant expansion in the c-axis and contraction in the prismatic directions. The primary irradiation effects to consider during this initial phase of GFR follow.

Swelling

Swelling in ceramics can be caused by several mechanisms such as amorphization, lattice strain, void swelling, and other mechanisms. In the operating temperature range of interest for GFR (300-1000°C) sufficient defect mobility exists to be above the amorphization temperature regime and for some ceramics are below the void swelling regime. The swelling will be dominated by the mismatch between contraction due to vacancy formation and lattice dilation due to interstitial accommodation. This lattice strain dominated regime is very temperature dependent, as illustrated in Figure 2.1 for SiC. [3] The reason for the lower swelling as temperature is increased in the lattice strain regime is the reduced number of surviving defects due to temperature-enhanced diffusion. Essentially, the enhanced vacancy-interstitial recombination leads to less swelling. While the behavior of Figure 2.1 is typical of ceramics, it is very important to note that the magnitude of swelling will be dependent on the ceramic type and irradiation temperature. Also, certain ceramics, such as magnesium aluminate spinels ($MgAl_2O_4$) are resistant to irradiation induced swelling at GFR temperatures for damage levels in excess of 100 dpa. [4] It is important to note there can anisotropic swelling for some ceramics, for example graphite and others sharing a hexagonal crystal structure. Not only would anisotropic swelling challenge core designers, but generally lead to unacceptable loss in mechanical and thermal properties.

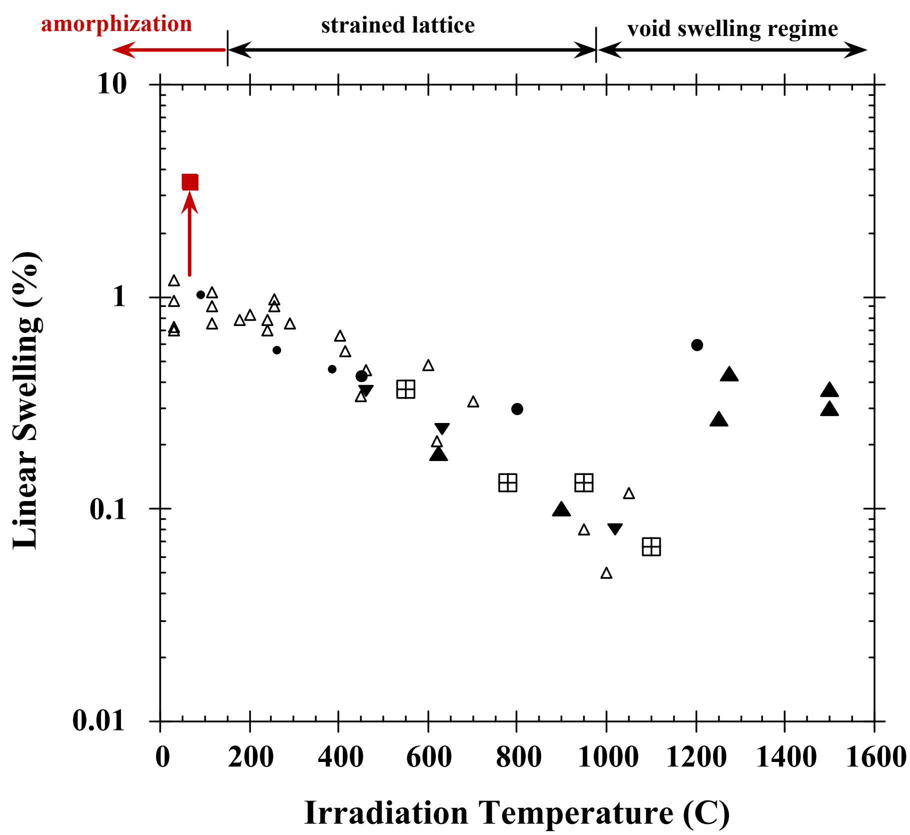

Fig. 2.1. Fission neutron induced swelling of pyrolitic SiC taken from various sources. [3]

For GFR applications, it is important that ceramics are selected such that the component application temperature is within the strained-lattice regime for a particular ceramic material to minimize swelling outside this temperature range. This is simply due to the very rapid onset of amorphization and the very large volumetric expansions associated with this transition for certain ceramics at low temperatures. As an example, amorphization for SiC has already occurred following exposure to only a few dpa and leads to greater than 10% volumetric expansion (Figure 2.1) In contrast at higher temperatures, the void swelling regime does not saturate at all and, while this is not well mapped by high-dose neutron experiments, may swell without bound. For the strained-lattice regime, the saturation in swelling will occur very rapidly (a few dpa) and the thermal conductivity degradation (see next section) will be less drastic for the higher temperature, lower swelling irradiation conditions.

Thermal Conductivity

The thermal conductivity of all ceramics degrades with neutron irradiation and tends to saturate at a very low dose (a few dpa) as compared with the expected component doses listed in Table 2.1. Figure 2.2 gives an example of the degradation in room temperature thermal conductivities for various ceramics of interest that have been neutron irradiated below 300°C. [5] It is important to note that there can be significant differences in the baseline, non-irradiated thermal conductivity within a given type of ceramic depending on many factors such as processing route, impurities, etc. As example, commercial grades of graphite, SiC, and Si_4N_4, range from values around 10 W/m-K to values over a few hundred W/m-K. However, as the defects produced due to neutron irradiation are more or less independent of the initial thermal conductivity, the saturation thermal conductivity for a given ceramic (i.e. “SiC”) will approach a unique value. [5] From Figure 2.2, it is clear that degradation for all these materials is substantial at doses of less than 1 dpa. [5] It is also seen that the thermal conductivity is tending toward saturation, and for SiC has already saturated (for this irradiation temperature.) Further degradation in thermal conductivity during irradiation in the “lattice strain” regime would not be expected unless the ceramic structure degrades (cracks) as would be the case for hexagonal close packed (HCP) ceramics such as graphite, BN, B_4C, BeO, etc.

Figure 2.2 demonstrates that neutron irradiation can have a dominant effect on the in-service thermal conductivity of ceramics. For example, the thermal conductivity of SiC and Al_2O_3 are similar following low-temperature irradiation to doses of 0.1 dpa and higher, despite the initial two orders of magnitude superiority in the unirradiated thermal conductivity of SiC. Similarly, the irradiated thermal conductivity of AlN is greater than that of SiC, whereas the unirradiated thermal conductivity of SiC is higher. It is obvious from these examples and other literature data that the unirradiated thermal conductivity is not a good metric for selecting materials with the highest irradiated thermal conductivity. Unfortunately, there are very few systematic studies of thermal conductivity of ceramics at GFR-relevant temperatures and damage levels. Further irradiation studies are needed to determine which ceramics have the highest thermal conductivity following high-dose, high-temperature neutron irradiation.

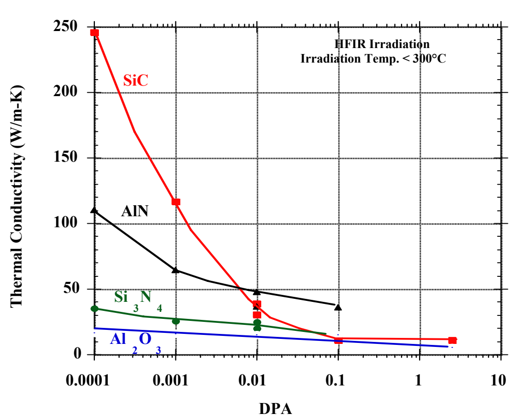

Fig. 2.2. Fission-neutron-induced degradation in room temperature thermal conductivity of various commercial ceramics. [5]

The cause of this degradation in thermal conductivity is phonon-scattering by irradiation-induced defects (primarily vacancies and small vacancy clusters) and is therefore unavoidable. [6] However, the magnitude of the degradation and corresponding saturation value of thermal conductivity is dependent on the type of ceramic as well as the irradiation and application temperature. The unirradiated thermal conductivity of ceramics generally decreases with increasing test temperature, whereas the effect of irradiation on thermal conductivity degradation is most pronounced for low irradiation temperatures. As previously noted, few data exist on high-temperature irradiation of commercial ceramics. However, example saturation room temperature thermal conductivities of a few common high-purity monolithic ceramics in GFR-relevant temperature ranges are given in Table 2.3 [7-10].

Table 2.3 Comparison of thermal conductivity for unirradiated ceramics and for ceramics irradiated to degradation saturation at GFR relevant temperatures.

Material	T_{irr} (°C)	K_{unirr} (W/m-K)	K_{irr} (W/m-K)*
High Purity Alumina	700	17.6	3.3
H-451 Nuclear Graphite	600	115	~20
CVD Silicon Carbide	800	380	23
Aluminum Nitride	470	266	<34

*The saturation of degradation in thermal conductivity typically occurs at relatively low doses relative to GFR conditions, e.g. less that a few dpa.

Mechanical Properties

With the exception of graphite, beryllium oxide, aluminum oxide, and silicon carbide, the effect of neutron irradiation on mechanical properties such as strength (compressive or tensile), elastic modulus and fracture toughness for ceramic materials has not received much study. For most ceramics of interest to GFR, elastic modulus, coefficient of thermal expansion, and specific heat will undergo negligible changes under irradiation. Of note is that there is very little information on the effect of neutron irradiation on the fracture toughness of ceramics (with the exception of graphite and very limited information on alumina, magnesium aluminate spinel, and silicon carbide.) However, strength change can be non-existent or considerable depending on factors such as ceramic processing route, impurities present, nuclear transmutations, and other factors that are unique to the particular trade-named ceramic. A clear example of this is seen when comparing pure forms of SiC with forms fabricated through a powder-processing route. Figure 2.3 shows the normalized strength of hot-pressed forms of SiC, which exhibit about a 50% decrease in strength after exposure to an exposure level of a few dpa. [11-18] In contrast, highly pure SiC formed through chemical vapor deposition undergoes no degradation, and may increase in strength at doses an order of magnitude higher than those of Figure 2.3. [18] In all cases, the degradation shown in Figure 2.3 is attributed to impurities associated with the fabrication process that can likely be avoided by appropriate selection of fabrication route.

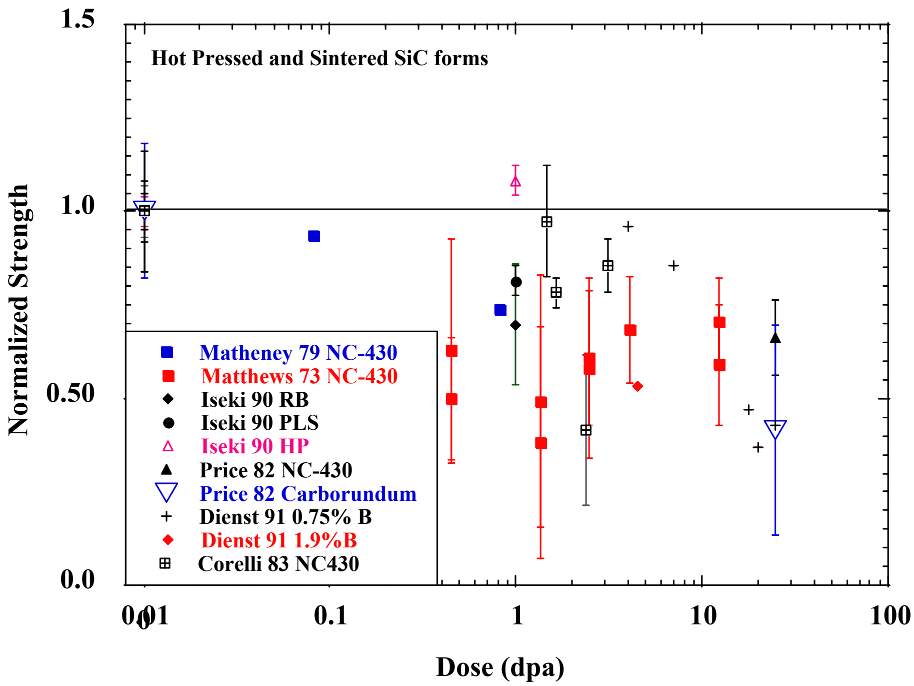

Fig. 2.3. Degradation in normalized bend strength for neutron irradiated hot pressed and sintered forms of SiC. [18]

2.4 Ceramic Composite Irradiation Effects

As discussed above, all ceramics show significant, and unavoidable, degradation in thermal conductivity under neutron irradiation. This degradation, which is caused by defect formation, is more pronounced for high-quality C/C owing to its higher initial thermal conductivity. Silicon carbide composites typically have thermal conductivity ranging from 20 W/m-k (Type-S fiber, CVI SiC matrix at 600°C) [19], which degrades somewhat under high-temperature irradiation. Graphite fiber composites can have extraordinary thermal conductivities (140 W/m-K @ 600°C) but degrade to < 50 W/m-K at saturation irradiation. [20] Over the temperature range of the GFR, the high-quality CFC's have a clear advantage in irradiated and non-irradiated thermal conductivity. It is important to note that the thermal conductivity of ceramic composites will not necessarily saturate as monolithic ceramics do. This is due to the fact that composites will have significant phonon scattering at fiber/matrix and other interfaces within the composite. It is expected that irradiation will disturb these interfaces leading to greater thermal conductivity degradation. [21] Other properties such as elastic modulus and thermal expansion have finite but minor changes under irradiation at the GFR application temperatures.

As mentioned earlier, the primary advantage of fibrous composites as compared to monolithic ceramics is their superior fracture toughness and more predictable failure characteristics. Both of these advantages are the result of engineering the material so that cracks propagating under load, which would cause catastrophic failure in monolithic ceramics, are tied-up at the fiber/matrix interface. Rather than a single crack causing failure a composite will endure a great number of crack fronts while maintaining load.

While these advantages create the possibility of using high temperature ceramics in a high-tensile-load structural application, the more complex structure raises new issues regarding radiation damage. As an example, continuous fiber composites are made up of at least three distinct components: fiber, matrix and fiber/matrix interphase. Because each of these components is critical to the performance of the composite, the performance is dictated by the "weak link." This problem is illustrated in Figure 2.4 by the significant strength degradation seen in earlier, less pure forms of SiC composite. In this case, the earlier forms, which have oxygen-containing fiber, degraded in strength by ~ 50% and lost considerable fracture toughness because of anisotropic volumetric changes between the fiber and matrix under irradiation. The upper curve of Figure 2.4 shows the most recent data on "pure" SiC fiber composites which appear to have no degradation following ~8 dpa, 800°C irradiation. However, it is important to note that at GFR-relevant doses there is no information on these "stable" forms of SiC/SiC composite.

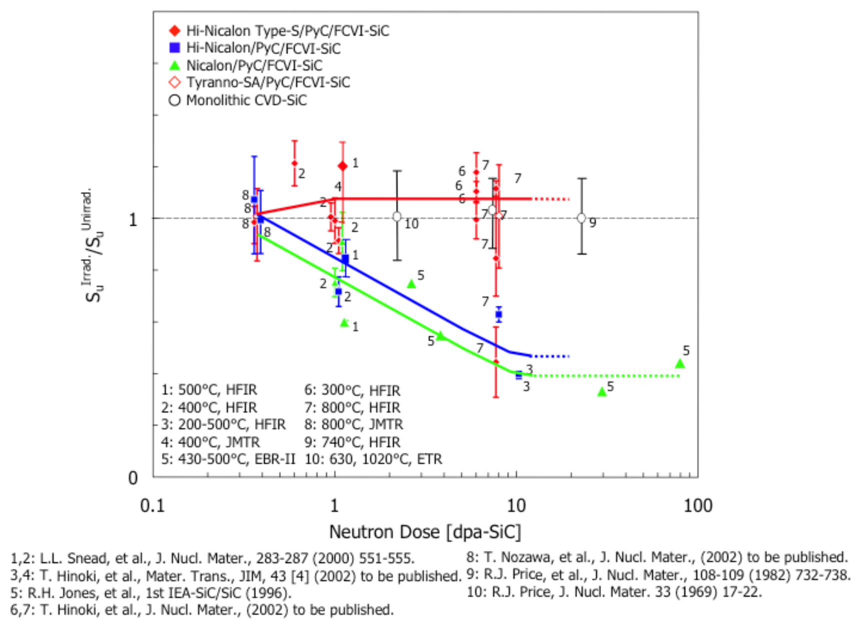

Fig. 2.4. Effect of neutron irradiation on the normalized strength of "pure and impure" fibrous SiC/SiC composite. Red curve is stoichiometric, blue curve contains ~1% oxygen-containing fibers, green curve contains ~ 10% oxygen-containing fibers.

Ignoring the potential issue of oxygen corrosion (which is most likely a bad assumption for GFR), the primary life-limiting irradiation consideration for high quality "pure" forms of SiC/SiC and C/C is swelling, which is fundamentally different for the two composite systems as explained elsewhere. [22,23] The essential point is that swelling in silicon carbide is isotropic, and saturates at < 1% (600°C-1000°C) while graphite swelling is highly anisotropic, exhibiting tremendous expansion perpendicular to its basal plane with contraction and inevitable cracking parallel to the basal plane. In the early stages of irradiation, graphite and C/C's undergo densification. At some neutron fluence, dependent on irradiation temperature, the material begins to swell and eventually disintegrate. The point at which the goes from densification, through the zero swelling point and into the rapid swelling regime typically defines life in C/C's. The effects of anisotropic swelling for high-quality 3-D graphite composite are illustrated in Figure 2.5. In this case, the radial swelling and longitudinal contraction of graphite fiber bundles are evident (note the gap between the fiber bundle and top bar edge in right micrograph.) The fact that the 800°C-irradiated sample exhibits more extreme anisotropic changes underscores the effect of temperature on swelling. Based on the behavior shown in Figure 2.5 the lifetime of C/C composite is estimated to ~1×10^{26} n/m^2 (E>0.1 MeV,) or about 10 dpa. While it is possible to engineer the composite weave to resist this anisotropic swelling and increase lifetime, in-core components will receive > 100 dpa (lifetime), well above the possibility range for C/C. However, applications of C/C composites in lower fluence areas within the GFR vessel are possible. It is important to hone, however that C/C usage for the S-CO2 cooled option of the GFR is impractical due to the severe oxidation of the composite expected in that environment.

For the case of SiC/SiC composite, swelling in the strained lattice regime (see Figure 2.1) saturates in a similar fashion to monolithic SiC. However, for temperature above 1000°C the swelling behavior is not well studied and swelling might transition to void-swelling, which does not saturate. Clearly, one of the main tasks in the scoping phase of GFR is to demonstrate stability of microstructure and mechanical properties of SiC/SiC at much higher doses than presently available.

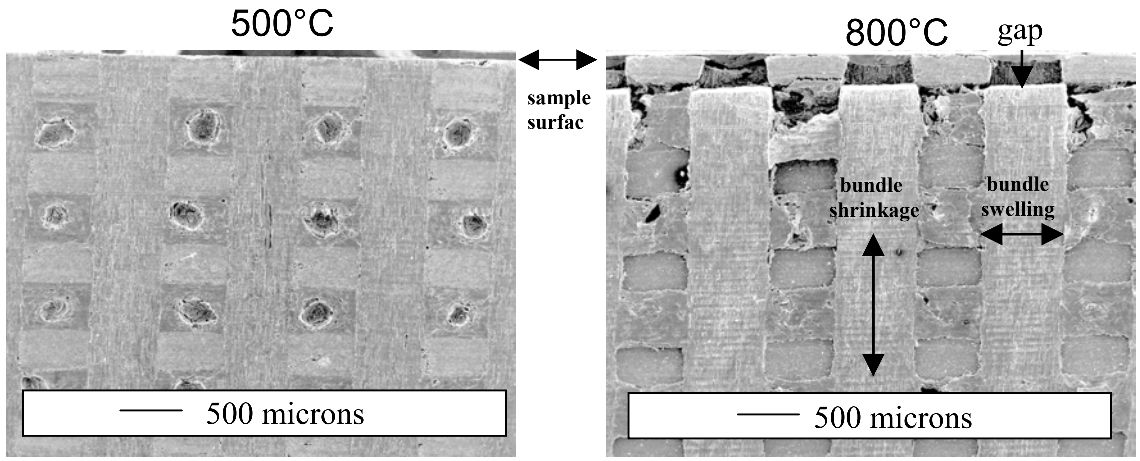

Fig. 2.5 SEM image demonstrating the dimensional instability of C/C under irradiation at 500 and 800°C, ~ 8 dpa.

Currently, a fundamental understanding of the irradiation performance of each of these materials exists, with the exception of the swelling behavior of SiC in the 1000-1500°C range. What is now required is a combined effort on the part of the materials and reactor designers to determine appropriate architectures for specific applications. Once determined, a directed program to develop the most resistant materials can begin. Fundamental aspects of this program will include development of relevant ASTM test standards for reproducible measurements of the required properties, development of incrementally more radiation-resistant materials, and a thermophysical property and proof testing campaign. Clearly, this represents a significant cost increase over conventional, code-qualified metallic alloys, but allows the only probable route to increased reactor temperature and thermal efficiency.

Examples of such a composite would be a polar-weave SiC fiber composite that is infiltrated with a SiC matrix for use in control rod application. Given the more complex nature of this class of materials, the fact that it is a custom made material, and the very limited database on irradiation performance that exists; the lead-time for establishing viability is quite long (10-15 years for high-fluence application.) Nonetheless, initial indications of viability should be obtainable with a focused effort by 2010.

2.5 Required Materials Testing and Evaluation

Insulating Ceramics

The primary work in this area will be 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

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

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. In parallel a committed ASTM standards development actively will be required to appropriately set standards for testing.

Carbon-Carbon Composites

Carbon-carbon composites will be heavily evaluated for use as structural materials for the NGNP. The primary difference between the C/C composites applications in the GFR and the NGNP is that the GFR C/C components will be limited to usage well outside to core to minimize excessive moderation, but even so, they will see significantly higher fluences. Hence, the 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.

Regulatory and Codification Requirements

An ASME code for composites used under GFR core conditions has not been developed. However, it is not clear that any codes will be required. General requirements for regulatory and codification requirements that may be needed for the GFR will be developed under the NGNP program. These may need to be extended to the more extreme conditions of the GFR, but not during the scoping phase of research.

Manufacturing Infrastructure Required

A mature manufacturing infrastructure for the advanced radiation-resistant SiC-SiC composites that will likely be used for the GFR does not exist at this time. Exploration of the path to developing this infrastructure will need to be examined during the scoping phase of GFR materials research.

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3. METALLIC MATERIALS FOR CORE COMPONENTS AND REACTOR INTERNALS

3.1 Operating Conditions

The main core components and their estimated operating conditions are summarized in Table 2.1. Three different designs will need to be considered as described earlier: the reference design (He direct), alternate design 1 (He/CO2 indirect), and alternate design 2 (S-CO2 direct). Because the outlet temperatures vary by 300 ºC, the structural materials in these three designs will experience substantially different temperatures. Therefore, the candidate materials for specific components in each design will differ in specific cases.

There are several distinct possibilities for the core design. These include the prismatic design where the core is constructed of blocks that incorporate the fuel. Other designs call for more or less conventional rods or plates that clad the fuel or for pebble bed arrangements contained within a core supporting basket-like structure. Control rods and associated sheaths or guides are additional in-core components that must be considered. The configurations of in-core structures will be quite different depending on the design chosen. However, all have in common the need to perform under approximately the same high fast neutron fluxes and high temperatures. Table 2.1 shows the values of several important parameters for the reference design.

The main in-vessel structures outside the core region are the gas duct barrel, hot gas duct, grid plate, upper and lower core support components and thermal insulation. Again, three different designs will need to be considered as described above. The estimated operating conditions for the metallic reactor internals for these designs are summarized in Table 2.1. Relative to the NGNP, some of the components in the GFR will experience higher temperatures, especially under off-normal conditions. The GFR core barrel, for example, is currently estimated to operate at temperatures up to 850°C, while that for the NGNP is 600°C. For off-normal conditions, the corresponding temperatures are 1200 and 1070°C, respectively. As shown in Table 2.1, the normal operating and off-normal temperatures decrease from the reference design to the He/S-CO2 indirect design and further decrease to the S-CO_2 direct design. The S-CO_2 design, however, presents a different set of compatibility issues with the use of supercritical CO_2 as the coolant. For the reference design and the He/S-CO2 design, the most significant demands placed on the reactor internals are the temperatures at which they will be required to operate and the radiation doses to which they will be exposed. For the S-CO_2 design, the radiation doses and exposure to the supercritical CO_2 are the most significant operational parameters. The potential candidate materials for the GFR are discussed in the next subsection, while the materials development and qualification research that will be required to resolve questions about the suitability of these materials are addressed in the remainder of this section.

3.2 Preliminary Candidate Materials

Because the upper end of 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. In addition, based on their high temperature capabilities, refractory alloys could 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 [1,2]; 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 [3,4].

The normal operating temperatures for the three primary out-of core components range from 490°C to 850°C for the reference design (Table 2.1). For the lower support plate, the low-swelling austenitic stainless steels and advanced versions of the 8-9Cr ferritic/martensitic steels are viable classes of candidate materials, depending upon design loading conditions. However the higher maximum temperatures for the upper support plate and core barrel (850°C) are beyond the operating capabilities of these materials, and for these applications it will be necessary to turn to the Ni-base alloys for the required high temperature strength and dimensional stability or to ODS versions of the ferritic and ferritic/martensitic steels produced by mechanical alloying. For alternate designs 1 and 2, where the outlet temperatures are 300K and 200-250K, respectively, below the outlet for the reference design, the normal operating temperatures for all three out-of-core components range from 300°C to 650°C. In all three cases, the advanced austenitic stainless steels, designed for swelling resistance, and the advanced version of the 8-9Cr ferritic/martensitic steels provide a group of viable candidate materials.

The preliminary estimates for off-normal transient conditions are of concern and the possible frequency and duration of various off-normal scenarios will require further evaluation since these parameters could strongly affect material selection. For the ferritic/martensitic steels, significant excursions above ~900°C could lead to serious embrittlement through an austenite to martensite transformation. Similar temperature excursions for the austenitic steels could lead to the destruction of the steady state swelling-resistant microstructure and subsequent rapid swelling. The Ni-based superalloys are potentially better able to withstand high temperature excursions and therefore should also be considered for each of these applications. However even these alloys may undergo incipient melting at temperatures as low as ~1200°C, and there are significant concerns about radiation-induced grain boundary embrittlement of Ni-base superalloys at temperatures above 500°C for damage levels above a few dpa. More detailed information on the austenitic stainless steels and ferritic/martensitic steels for reactor applications may be found in a recent report on the survey of materials requirements for the SCWR [5]. The application of Ni base alloys for out-of-core components has recently been discussed in detail in the materials selection and qualification report for the NGNP [6].

3.3 Status

There is a very large database of experience with the austenitic stainless for nuclear applications both from the fast breeder reactor and light water reactor (LWR) programs covering a wide range of neutron dose, temperature and loading conditions. There is also an enormous commercial-scale experience in the production, fabrication and joining of this class of materials. Based upon the initial experiences with the phenomena of void swelling and grain boundary embrittlement encountered in the 300 series stainless steels, a number of improved alloys were developed by the international community, for fast breeder reactor applications. Many of these advanced alloys have been produced in a variety of product forms on a commercial scale. Within the family of advanced austenitic steels candidate compositions include: a) composition-restricted 316 stainless steels with nitrogen modifications such as the French breeder program 316 and the Japanese 316FR; b) 316-type microalloyed with Ti, B and P such as the Japanese PNC 316, and the US D9 alloys and HT-UPS alloys; and c) more highly alloyed versions such as the French 15Cr-15Ni-Ti and the 12Cr-25Ni-Ti alloys. Many of these materials have demonstrated low swelling incubation regimes up to 130 dpa and many of them also exhibit superior creep strength relative to 316 stainless steel.

Ferritic/martensitic steels in the 9-12%Cr range are somewhat more swelling resistant than the austenitic stainless steels with very low swelling behavior at moderate temperatures to doses >150 dpa. The early commercial alloys examined by fast reactor programs included Sandvik alloy HT-9 containing 12%Cr and 1%Mo. Newer alloys have evolved with better properties based upon 9%Cr and 1%Mo such as T91 and a series of reduced activation alloys in which Mo and Nb are replaced with small additions of W, V and Ta; examples include the Japanese F82H and the US 9Cr-2WVTa alloy [7]. In addition, advanced steels such as NF616 and HCM 12A from Japan and E911 from the EU have been developed for operation up to 620C although radiation experience is limited.

The initial set of Ni-base alloys for the higher temperature applications will be based upon the materials selections for the support structures for the NGNP [6]. Candidate materials include variants of Inconel 617, Alloy 800H and Hastelloy X and XR. Alloy 617 has the advantage of a very large database developed in support of ASME Code deliberations, which led to the approval of a Code Case. Alloy 800H is already in Subsection NH and could be a candidate for applications up to ~750°C. There is a very extensive knowledge base on the Hastelloy materials in Japan. Within this body of information however there is very little information on the response of these alloys to the levels of displacement dose indicated in Table 2.1. Dose levels for the NGNP metallic support structures are limited to below ~1dpa and radiation-induced embrittlement is not expected to be a significant mode of performance degradation. More extensive irradiation experience under relevant fast reactor conditions exists for alloys such as In 718, In 706, and PE16. However, all commercial Ni-base alloys examined in the earlier fast breeder reactor programs were found to be susceptible to radiation embrittlement after exposures above a few dpa for temperatures ranging from 450°C to 750°C (the latter being the maximum investigated temperature for irradiated Ni-base alloys).

Refractory alloys based on body-centered cubic (BCC) (Nb, Ta, Mo, W) or HCP (Re) crystal structures offer very attractive high temperature strength properties, and may be the only alloys capable of long-term operation for structural applications above 900˚C. However, current commercially available refractory alloys have poor compatibility with oxygen and CO_2 at temperatures above ~500˚C that may preclude their use in the GFR. There is also decreasing industrial capability to produce large-scale refractory alloy structures.

3. 4 Irradiation and High Temperature Strength Issues

One of the unique features of the GFR environment is that the out-of-core components, such as the core barrel and upper and lower support plates, have to withstand high levels of displacement dose in temperature regimes where radiation effects such as radiation hardening, solute segregation and phase instabilities, void swelling and helium embrittlement are known to occur. These phenomena impact each of these 3 classes of materials to various degrees depending upon specific composition and microstructure. Within each alloy class, however, it is possible to devise metallurgical strategies to mitigate against the various types of property degradation that stem from these phenomena. For example, strategies have been demonstrated for a) minimizing the effects of radiation hardening on DBTT shifts in ferritic/martensitic steels, b) extending the incubation dose for void selling in austenitic steels to >100dpa, and c) minimizing the impact of helium on high-temperature grain boundary strength in austenitic stainless steels.

The Ni-base superalloys have the required strength/temperature capability for GFR reactor internals, coupled with excellent swelling resistance, in some instances. However, their susceptibility to grain boundary embrittlement stemming from radiation and thermally driven phase instabilities and helium accumulation are of concern in view of the long lifetime requirements and projected levels of displacement dose up to 100dpa. Further investigation of these phenomena and the development of compositional and microstructural modifications for improved radiation response present a major challenge requiring an aggressive R&D program. It is worth noting that although there is a substantial body of irradiation data on Ni-base alloys, the underlying mechanisms for embrittlement have never been unambiguously identified and consequently there has never been a significant attempt to design a composition/microstructure specifically for radiation performance. Within the three alloy classes discussed, there are a large number of commercially available wrought alloys, only a small fraction of which have been developed specifically for performance in neutron environments. In addition to these alloys, which are produced via melting and casting methods, consideration needs to be given to materials produced by mechanical alloying such as the ODS ferritic and ferritic/martensitic steels. There is a significant amount of information available on the radiation response of a limited number of such alloys. Recent R&D on advanced nano-dispersed ODS ferritic/martensitic steels has shown high promise for exceptional creep strengths at temperatures 200-250°C higher than the maximum operating temperatures for conventional steels. These materials are strengthened by an extremely fine dispersion of nano-sized atom clusters and particles that confer remarkable high temperature microstructural stability and creep resistance. This work is at a preliminary stage but clearly there is a potential for tailoring materials via mechanical alloying to meet some of the challenging requirements posed by the GFR concept. The processing techniques being developed to generate nano-scale clusters and particles in the 8-14Cr steels could conceivably also be applied to Ni-base alloy systems with the possibility of developing materials with greater microstructural stability in the face of severe temperature excursions. Additionally, it should be possible to select a combination of matrix solutes designed to confer superior compatibility in high temperature impure helium or possibly in supercritical CO_2.

Refractory alloys have superior strength compared to steels and Ni base alloys at high temperatures, and may be the only metallic alloy option from a strength perspective for structural applications at temperatures above 800 to 900˚C. As with all body-centered cubic (BCC) alloys, refractory alloys based on Nb, Ta, Mo and W are susceptible to low temperature radiation embrittlement. Although irradiation data are limited, the minimum allowable operating temperature for BCC refractory alloys during neutron irradiation ranges from ~500 to 800˚C [1,6]. Rhenium, which has an HCP structure, has good strength up to very high temperatures. However, it exhibits very low uniform elongation at high temperatures and irradiation has been found to decrease the temperature at which the onset of low ductility occurs [8].

3.5 Fabrication Issues

There do not appear to be any fabrication issues regarding the existing candidate austenitic stainless steels and ferritic/martensitic steels. Some concerns could exist regarding heavy-section welding of the nickel base alloys, if required, but research is currently underway in other projects to develop the required technology. Alloys based upon mechanical alloying, such as ODS materials, will require substantial efforts to develop satisfactory joining methods. Industry capability exists for fabrication of refractory alloys, although these capabilities generally have diminished over the past 30 years.

3.6 Infrastructure Issues

Although it is difficult to identify potential issues with infrastructure without knowledge of the specific designs for the various components, the preliminary opinion is that the manufacturing infrastructure is in place for all alloy classes that will be included in the testing program. There is a depth of historical experience in working with suppliers to obtain both small and large heats of these alloys to specifications. Similarly, large-scale fabrication of reactor internals components from selected materials for the GFR is well within the capabilities of the existing industry in the US, Japan and Europe. It is clear, however, that suppliers will need to have some in-house R&D capabilities in order to produce the required compositions, product forms and microstructures specified for radiation service.

With respect to infrastructure, it must be emphasized that the materials program described here requires extensive capabilities for irradiated specimen work. In turn this translates to a need for substantial and modern hot cell capabilities for irradiated specimen preparation, handling, testing and disposal. Due to lack of demand, the refractory alloy commercial fabrication and testing infrastructure in the US has significantly diminished over the past 30 years.

3.7 Regulatory and Codification Issues

It is prudent to assume that materials property test data approaching that required for ASME code qualification, for each of the materials specified for service, will be required in order to license the GFR demonstration plant. Metallic core support structures must conform to ASME Sect. III, Div. 1, Subsection NG. Other core internals may conform to different rules. The applicable section for delineation of allowable design stress intensity factors is ASME Section II, Part D, Tables 2A, 2B, and 4. These tables cover temperatures to 370°C for ferritic alloys and 425°C for austenitic alloys. Subsection NH of Section III permits construction to higher temperatures for a limited number of materials. These are 2 1/4 Cr-1 Mo steel (Class 1), 304H stainless steel, 316H stainless steel, and alloy 800H. Some 300 series stainless steels are now qualified for service but these are not low-swelling compositions. Similar comprehensive experimental data will be needed for the low-swelling variants, as well as the ferritic-martensitic steels and the high nickel alloys described in the previous section. Much of the needed information on unirradiated properties already exists for certain of the alloys as a result of work in other programs, especially the LMFBR cladding and duct program, the Fusion materials program and the Japanese fast reactor development program. At least initially, the approval for use of these alloys in reactor service for the GFR is expected to be as a code case rather than as full code qualification

3.8 Other Issues

Issues related to materials performance in GFR helium are very similar to that of the VHTR with exception of fluence and possibly, the helium gas composition as discussed in Section 8 on GFR materials environmental compatibility considerations. Feasibility evaluation of proposed alloys will be performed at the temperatures of interest for approximately 10,000 h to obtain missing data. Data does not exist to establish viability of materials performance with supercritical CO_2 at the needed temperatures. As needed, materials will be exposed in supercritical CO_2 at appropriate temperatures ranging from 350-1250°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.

3.9 R&D Plan to Establish Materials Viability and Downselect Candidates

Although the initial approach to the reference design for in-core structures will be based upon the application of ceramic materials, a review of the current status of selected refractory metal alloys will be carried out with emphasis on mechanical and oxidation behavior and radiation effects. It will then be possible to evaluate possible R&D approaches to developing refractory metal alloys for applications in the reference GFR environment.

The first step in the research program on metallic materials for the reactor internals will be a comprehensive and detailed review of the potential candidate alloys discussed. 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 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.

The Ni-base alloys present a different situation since every known set of irradiation data has indicated the potential for high-temperature grain boundary embrittlement. Additional mechanical property assessment for Ni-base alloys, beyond what is already planned for the NGNP, is unwarranted until feasible approaches to solving the grain-boundary embrittlement problem have been demonstrated. Following an in-depth review of the available data and the possible mechanisms involved, low-dose irradiation experiments will be conducted on a series of modified and exploratory alloys to investigate compositional/microstuctural strategies to mitigate high temperature embrittlement.

The development of nano-structured alloys fabricated by mechanical alloying presents a promising approach to expanding the high-temperature capability in terms of both creep and swelling resistance and oxidation behavior. It is proposed to evaluate existing ODS materials and, if warranted, initiate R&D on the design and fabrication of exploratory new materials, both Fe and Ni-based, specifically designed to meet the more challenging aspects of the GFR environment primarily through a collaborative program with on-going research efforts in this area.

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.

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4. REACTOR PRESSURE VESSEL SYSTEM MATERIALS

4.1 Operating Conditions

The reactor pressure vessel system envisioned for the GFR is similar in many respects to that of the NGNP and is illustrated in Section 1. It will comprise a large reactor pressure vessel (RPV) containing the core and internals, a second large vessel for power conversion (PCV) containing the main turbine, generator, and associated turbo machinery and heat exchangers, and a pressure-containing cross vessel (CV) joining the RPV and the PCV. A summary of the anticipated operating conditions for the pressure vessel system is provided in Table 2.1. Reference [1] provides the relevant material needs for the NGNP pressure vessel. The NGNP materials report describes candidate pressure vessel material for lower (850°C outlet) and higher (1000°C outlet) gas-cooled systems. The outlet temperature envisioned for the GFR is 850°C. It is noted that the preliminary RPV size for the GFR indicates a smaller diameter and smaller height than that for the NGNP, while the thicknesses are also less, except in the case of the S-CO_2 design for which the RPV will need to be appreciably thicker than the NGNP vessel. The vessels will be exposed to air on the outside and either helium or supercritical CO_2 on the inside. The materials tentatively selected for gas-cooled RPV service are low-alloy ferritic/martensitic steels, alloyed primarily with chromium and molybdenum. The most significant demands placed on the RPV system are the temperatures at which they will be required to operate. Although the currently envisaged operating and off-normal conditions are shown in the Table 2.1, there are uncertainties regarding the actual temperatures and times, loads, load-time history, time-temperature-load histories, and the temperature and neutron flux gradients through the RPV wall, especially for the S-CO_2 design. Moreover, there is no current estimate for fatigue cycles for the RPV system, although the estimate for the NGNP is for about 150 cycles plus hydrogen cycles for a total of about 600 small cycles. It is recognized that the normal operating temperatures for the RPV system are dependent on the capabilities of the materials of construction. Thus, an iterative approach will be required to eventually match the limiting material capabilities and the design operating conditions.

The potential candidate materials for the three pressure vessels and closure bolting are discussed in the next subsection, while the materials development and qualification research that will be required to resolve questions about the suitability of these materials are addressed in the remainder of this chapter.

4.2 Preliminary Candidate Materials

Based on the currently estimated operating temperatures, 2 1/4Cr-1Mo steel would be the most likely candidate pressure vessel material for the GFR, if design and construction were to begin today and if the RPV was somehow shielded to reduce irradiation exposure significantly. However, given the lead time available before material selection is anticipated for the GFR system, materials research and development efforts with other ferritic materials should be a definitive part of the GFR program. Even for the NGNP, for which the lead time is very short, it is anticipated that further developments with variations in the modified 9Cr-1Mo class of ferritic-martensitic steels will provide a material with superior high-temperature creep strength than currently available and with far superior radiation resistance than 2 1/4Cr-1Mo steel. In the case of the GFR system, the research and development program should incorporate more advanced materials in the overall class of ferritic-martensitic steels, some of which are currently in progress. 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 and some of these materials are discussed in more detail in a later section.

4.3 Status

For the ferritic steel option, there are four classes of advanced, higher alloy ferritic-martensitic steels that have been identified as potential candidate alloys, while the 2 1/4Cr-1Mo alloy is also listed because of its extensive database and experience history. These five alloy classes are shown below. There are specific alloys identified within each class, as well as statements regarding available data, experience, and Code status. These classes of alloys are listed in the order recommended as priority for consideration as the structural material for the RPV and CV components for the GFR because of the long lead time available. However, as mentioned above, there are ongoing developments in a number of cases that are still in the research stage with development to follow in the out years. Some of the research and development advances are applicable to all the cases listed, such as the case of the developing technique of dispersion strengthening without the use of conventional processing techniques.

1. Class of 9Cr-1MoVNb

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

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

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

2. Class of 7-9Cr2WV

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

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

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

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

3. Class of 3Cr-3WV

   a. This class of alloys offers good high strength properties, but is one of the newer alloys under development and, as a result, has a very limited database. In relatively modest section sizes evaluated to date, the yield strength of the specific 3Cr-3WV alloy under development in the Fusion Materials Program is about twice that of the SA508 grade 3 forging steel used for current LWR RPVs.

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

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

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

4. Class of 12Cr-1MoWV

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

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

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

5. Lower temperature operation: 2 1/4Cr-1Mo

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

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

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

   d. Low relative radiation resistance of this material would require a significant shielding feature to reduce the radiation level well below 0.1 dpa.

Potential candidate alloys for the PCV could include those for the RPV and CV, but there are lower cost options available because of the lower operating temperatures. Even under abnormal conditions, the PCV will be subjected to temperatures about the same as those currently used for commercial light-water reactor (LWR) vessels (~290ºC). Moreover, the size of the vessel is well within normal fabrication capability. Thus, the current LWR pressure vessel materials, SA508 grade 3 class 1 forgings or SA533 grade B class 1 plates are potential candidates, as is the 2 1/4Cr-1 Mo alloy, dependent on material compatibility issues. It is noted that the CV is welded to the PCV and the welded joint with dissimilar materials must be a consideration.

Potential candidate alloys for high-temperature closure bolting are alloy 718 and types 304 and 316 stainless steels. Although alloy 718 has superior strength, it is currently only approved up to 566ºC in ASME Section III, Subsection NH. The two types of stainless steels, however, have allowable stress intensities for bolting up to 704ºC. An evaluation of the database for the alloy 718 will be conducted to assess the data needed, if any, for increasing the allowable temperature to that required for the GFR. Also, the estimated irradiation exposure for closure bolting will be assessed to evaluate the need for inclusion of bolting in the irradiation program.

4.4 Irradiation and High-Temperature Strength Issues

Although not expected to be a limiting issue in the case of the NGNP, the radiation levels are significantly higher in the case of the GFR. As noted earlier, the anticipated radiation exposure for the RPV is too high for 2 1/4Cr-1Mo steel without some form of shielding to reduce the exposure. A tungsten liner has been suggested as one possible option to reduce the exposure to allow for the use of the 2 1/4Cr-1Mo alloy, though this would introduce significant design and cost issues into the GFR. Moreover, tungsten will not be usable in option utilizing supercritical CO2 as a coolant, due to severe corrosion interactions. Even in the helium-cooled designs, tungsten would be seriously challenged and only useable if the partial pressure of oxygen was keep extremely low (<<1ppm). Irradiation effects data on the RPV and CV material are needed for regulatory requirements and for assessment of structural integrity. For the 9Cr-1MoV (grade 91) alloy, there are some very promising data available on irradiation effects, but data at the specific temperatures for the GFR are sparse at best. There are some data for some of the other potential candidate alloys listed in Section 4.3 but, again, insufficient data for the GFR conditions. Moreover, because of the higher radiation field compared with the NGNP, and the fact that the RPV is shorter, it is likely that the closure bolting material will need to be included in the irradiation program. Similarly, long-time thermal aging data are needed as a complement to the irradiation effects data for potential embrittlement due to either hardening or softening of the RPV materials and the closure bolting materials.

4.5 Fabrication Issues

The issue of vessel fabricability is a major issue that will need to be comprehensively evaluated in the case of the S-CO_2 direct design because of the greater anticipated thickness of the RPV associated with the 20 MPa coolant pressure. Similar studies to be performed for the supercritical water reactor, which also operates at high pressures, will be of little value, since the higher temperatures of operation of the GFR will required other materials. It is very unlikely that the manufacturing of that RPV could 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 above have not been manufactured, including the 9Cr-1Mo-V alloy. The main issue in this regard is the attainment of the required through-thickness properties of the higher-alloy steels in such thick sections. Additionally, weldability of the steels in thick sections is also an issue. For the reference and the S-He/S-CO_2 indirect designs, the lines between domestic and foreign capabilities are not so sharp. A review of domestic capability for manufacture of such vessels is required.

4.6 Regulatory and Codification Issues

The RPV system is, of course, part of the primary pressure boundary and, as such, must conform to ASME Section III, Division 1, Subsection NB for Class 1 components. Section II, Part D, Properties, provides the maximum allowable stress values (design stress intensity values) with Tables 2A and 2B being applicable to ferrous materials and nonferrous materials, respectively, for Class 1 applications. Table 4 of Section II similarly provides maximum allowable stress values for Class 1 bolting. Detailed discussion of the bases for establishment of the design stress intensity values is beyond the scope of this report, but suffice it to note that they are nominally based on the lower of 1/3 tensile strength or 2/3 yield strength, with different criteria for wrought and cast alloys, and welded pipe or tube. Moreover, there are ratios applied to those basic criteria that are dependent on temperature dependent trends. Table 2A covers temperatures to 371ºC (700ºF) for ferritic alloys and 426ºC (800ºF) for austenitic alloys, while the maximum temperature in Table 2B is 426ºC (800ºF). Subsection NH to Section III permits construction to higher temperatures for a limited number of materials. These are 2 1/4Cr-1Mo steel (Class 1) to 649ºC (1200ºF), types304H and 316H stainless steels to 816ºC (1500ºF), and alloy 800H to 760ºC (1400ºF). Those maximum temperatures are, however, somewhat deceiving in that they do not necessarily represent the approved temperatures for long-time operation (300,000 h). For example, the upper temperature limit for that long-time operation for the 2 1/4 Cr-1Mo steel is

593ºC (1100ºF). At the maximum allowable temperatures for long-time operation, the maximum allowable stress intensity values are extremely low; for the type 304 stainless steel, for example, it is 4 MPa (0.6 ksi). For the 2 1/4Cr-1Mo steel at 593ºC, the maximum allowable stress intensity is 19 MPa (2.7 ksi).

At the current time, 9Cr-1Mo-V steel (grade 91) is approved to 649ºC (1200ºF) for Classes 2 and 3 applications in Section III of the Code with the maximum allowable stress intensity at that temperature of 30 MPa (4.3 ksi). However, at this time, grade 91 is in the final stages of acceptance into Subsection NH, with the proposal for its inclusion to 649ºC (1200F) and the same allowable stress.

Dependent on the actual design RPV operating and off-normal temperatures, extension of the required data bases and ASME Code acceptance of the materials for GFR RPV service may not be needed for the few current materials noted above. However, given the long lead-time available for material selection, the GFR materials research program should incorporate advanced materials that offer significant increases in elevated temperature creep strengths relative to those currently in the ASME Code. All activities in this regard will need to be closely coordinated with the high-temperature design methodology activities.

As mentioned earlier, a major issue for the RPV and CV materials is the conditions experienced under abnormal operating conditions. Currently, the estimated maximum temperature is 650ºC (1200ºF) with the time at that temperature currently unknown, although operation at that temperature for about 50 h and, more importantly, at full operating pressure or higher is estimated for the NGNP.

4.7 Infrastructure Issues

As mentioned earlier, fabricability will be a major consideration in the selection of materials, especially in the case of the S-CO_2 direct system. Besides the technical issues associated with fabrication of the vessel, transportation of the completed RPV or even the large ring forgings to the reactor site may be problematic. The diameter of the RPV is apparently relatively well defined, 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 takes on more significance in that a post weld heat treatment in the field is more difficult to conduct and control than that performed in the shop environment. Also, as mentioned earlier, a review of domestic capability is required.

4.8 Other Issues

Besides the requirements discussed above, other application specific information and data are required:

1. For example, the contamination levels of the various coolants are needed to determine and bound the effects of the coolant environment on the material properties and the materials must be tested under those conditions. Small amounts of impurities such as CO_2, CO, H_2O, H_2, CH_4, O_2, etc. can contaminate the coolant from a variety of sources throughout the reactor system and quite small amounts of these contaminants can degrade the materials by corrosion/oxidation processes, and by effects on mechanical properties. Carburization and decarburization are issues of particular interest. Because the degradation levels of the coolant on the potential candidate alloys are unknown, the concept of application of a weld overlay cladding on the inner surfaces of the RPV and CV will be considered, especially in the case of the S-CO_2 design. Because the cladding is not a structural member, the primary issue is corrosion; the first material to be

considered and evaluated would be type 308 stainless steel because it is a common clad used for current LWR RPVs and it has good elevated temperature properties.

2. The ASME Code requires determination of the RT_{NDT} for Section III, Class 1 components such as the RPV and CV. Fracture toughness data will be required, primarily for regulatory needs, but also for providing complete information to allow for a comprehensive assessment of structural integrity for the pressure boundary components.

3. As mentioned previously, damage accumulation data are needed due to long-time high temperature exposure. Particular attention is needed in the area of welding to ensure that the issues of hot cracking and premature creep failures in the heat-affected-zones of ferritic-martensitic steels, observed in the fossil industry, are adequately addressed.

4. Besides the traditional physical properties needed, damping coefficient data are also required by the designers for the RPV and CV materials.

4.9 R&D Plan to Establish Materials Viability and Downselect Candidates

The first step in the research program on materials for the RPV system will be a comprehensive and detailed review of the potential candidate alloys discussed. 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. The existing database for those alloys will be assembled, analyzed, and evaluated with respect to the design and operating requirements presented above. Fabrication and transportation for the RPV or the ring forgings are critical issues to be included within that first step. For example, the thickness of the various portions of the RPV is, of course, directly related to the maximum allowable stress at a particular temperature. For a given operating pressure, as the allowable stress decreases the thickness necessary to resist deformation increases. However, it is recognized that the various parameters will evolve with maturity of the design and associated analyses and close contact between the designers and materials researchers will be a continuing aspect as the program progresses. Research will be directed toward a better understanding of the high-temperature strength, stability, radiation resistance, and long-time performance of the materials. Because the design service life is 60 y for this reactor, long-time performance is an issue of particular importance since that amount of time exceeds the experience database for most of the potential candidate materials. For current LWRs, the temperature allows the RPV to operate well below the creep regime. However, because of the relatively high temperatures of the GFR, the RPV and CV materials will operate well within the creep regime. Thus, damage assessment and life prediction are of high importance in this case, as it is for the in-core metallic internals and the materials data produced in this area will need to be closely coordinated with the high-temperature design methodology needs and schedules.

As always, but especially in the case of structural components operating in the creep regime and, even more especially in the case of an RPV, the first line of defense for a nuclear reactor, evaluation of welding processes, welded joints, and component inspections must be emphasized. Of course, inclusion in the ASME Code for the materials of construction is required and the research, testing, and qualification needs will be directed towards development of the data and information necessary to meet those requirements. For an alloy such as 9Cr-1MoV (grade 91) that has already attained Code approval for operation to 649ºC, additional testing will be required but not nearly as much as that required for an alloy that is not currently approved for Code use. However, given the target of 2010 for downselect of candidate materials for the RPV system, other advanced materials, such as those discussed earlier, will be evaluated. The required data for most of materials discussed earlier are not available and a research program to obtain the data is required.

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 properties needed for all the various materials are essentially the same for all three GFR design concepts, with the exception of the S-CO_2 direct design, which requires additional considerations. As mentioned earlier, this design is of much higher pressure and will require a significantly thicker vessel with the concomitant issues of fabricability, both with respect to through-thickness properties and welding. Moreover, this design presents a more aggressive environmental situation with regard to corrosion/oxidation of materials and additional creep testing in the anticipated environment will be required. Thus, especially in this case, the environmental issues will require substantial evaluation.

Because of the 60 y design life, thermal aging is a significant issue for the GFR, as it is with the NGNP. Thus, as with the NGNP research plan, thermal aging experiments will be required to obtain data not currently available. Although there is some temperature overlap with the NGNP, it is not comprehensive and additional experiments will be required specifically for the GFR.

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 at the specific design conditions to validate that information for the designers and for the regulatory authority. As a first step, a detailed review will be conducted of irradiation effects on all the potential candidate alloys mentioned above. An experimental program will be designed based on the results of the review and irradiations of preliminary candidate materials will begin once an irradiation facility is identified. In addition to irradiation of the currently identified materials, selected advanced materials will be included. For purposes of this plan, specimens to be irradiated will include those for tensile, creep, and stress rupture, Charpy impact, and fracture toughness. In the case of an RPV without heavy shielding against radiation, irradiations would be conducted in a high-flux facility to attain the necessary dose (~15 dpa) in a reasonable time. For the use of standard 2 1/4 Cr-1 Mo steel, some shielding (e.g., tungsten) would be required to decrease the dose to below about 0.1 dpa, and a low flux irradiation facility would be more appropriate to obtain the necessary data.

A review of joining technology is required as well and the literature review will include a comprehensive review of joining issues for the potential candidate alloys. In the case of 2 1/4 Cr-1 Mo, it is not likely that joining issues will be identified based on the extensive industrial experience with welding of thick pressure vessels of that material. For the other materials discussed, however, that is not the case and the NGNP program is addressing those issues for some of the potential GFR candidate materials. The task on joining will require the fabrication of welded joints with the various potential candidate materials for inclusion of specimens in the baseline, aging, and irradiation effects tasks.

An evaluation of fabrication infrastructure will be conducted as part of this program to identify those areas where domestic capability exists and to identify alternate foreign sources. As with the NGNP, fabrication of the RPV is not considered to be a “show-stopper” for this reactor, but heavy-section welding and post weld heat treatment development will be required for the high-chrome, low alloy steels. As mentioned earlier, through-thickness mechanical properties must be attained. In the case of the $S\text{-}CO_2$ direct design, the production of forgings of sufficient size and thickness will undoubtedly require foreign resources, such as Japan Steel Works. Such a review is incorporated within the NGNP program.

5. HIGH-TEMPERATURE METALLIC COMPONENTS FOR GFR

5.1 Operating Conditions

For selecting high-temperature metallic materials, considerations of the GFR operating conditions are focused on components that operate outside of the intense radiation field. Such components include piping and heat exchangers. Further, high-temperature materials for the power conversion components, such as the turbine, compressors, coolers, and recuperators, are discussed in Section 6. In this sense, the operating conditions of the GFR high-temperature materials differ from the case of the NGNP, where internal metallic support components are subjected to much lower neutron fluences and are included in the category of high-temperature materials.

The anticipated temperatures in the three proposed GFR designs are all relatively lower than those of the NGNP. The reference He-cooled design operates with an outlet temperature of 850º at 7 MPa; the He-S/CO_2 indirect option has an outlet temperature of 600-650°C at 7 MPa with a 550°C secondary at 20 MPa; and the all-S/CO_2 will operate with an outlet temperature of 550°C at 20 MPa. The all-He direct design of NGNP performs with an outlet temperature of 1000ºC at 7.4 ~ 8 MPa. From an operating temperature point of view, the candidate high temperature metallic materials for NGNP can be directly considered for GFR applications.

As to environmental conditions, the "all-He direct" design option of GFR adds concerns for the effects of helium impurity contaminations that could be more severe than the NGNP, as discussed in Section 6 on power conversion and Section 7 on general corrosion considerations. The other two design options, He-S/CO_2 indirect and all-S/CO_2, add significant compatibility challenges at the anticipated service temperatures.

5.2 Preliminary Candidate Materials and Status

Based on the operating conditions of GFR and efforts made in NGNP materials selection, two groups of metallic materials are recommended as primary and secondary candidates, respectively, for high temperature metallic GFR components.

The primary potential candidate materials for high-temperature balance of plant components other than the power conversion system are listed in Table 5.1 with brief status information. Among these materials, Inconel 617 is considered as a leading candidate. The material was developed in the earlier gas-cooled reactor projects, and 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 applied. Alloy 800H is in Subsection NH, and would be the leading candidate for the intermediate temperature range of 600-760ºC. The 316FR stainless steel is not in Subsection NH, but the database is adequate to incorporate the steel should the need arise. The Gr91 and Gr22 (Class 1) steels are currently in Subsection NH.

Table 5.1. Primary Potential Candidate Materials for High-Temperature Metallic GFR Components

Primary Candidates	Nominal Composition	UNS Number	Existing Data Max Temp. °C	Helium Experience	Aging Experience	Section II Physical Props	Design Codes
Inconel 617	45Ni-22Cr-12Co-9Mo	N06617	1100	Yes	Yes	No	Yes
Incoloy 800H	33Ni-42Fe-21Cr		1100	Yes	Yes	Yes	Yes
316FR	16Cr-12Ni-2Mo		700	No	Yes	No	No
Gr91	9Cr-1Mo-V		650	No	Yes	Yes	Yes
Gr22	2 1/4Cr-1Mo		650	Yes	Yes	Yes	Yes

Table 5.2. Secondary Potential Candidate Materials for High-Temperature Metallic GFR Components

Secondary Candidates	Nominal Composition	UNS Number	Existing Data Max Temp. °C	Helium Experience	Aging Experience	Section II Physical Props	Design Codes
Hastelloy X	Ni-22Cr-9Mo-18Fe	N06002	1000	Yes	Yes	Yes	No
Hastelloy XR	Ni-22Cr-9Mo-18Fe		1000	Yes	Yes	No	Yes
CCA Inconel 617	45Ni-22Cr-12Co-9Mo	N06617	1100	No	No	No	No
Alloy 230	53Ni-22Cr-14W-Co-Fe-Mo		900	No	No	Yes	Yes
Gr92	9Cr-1.5W-Mo-V-Nb		650	No	No	Yes	Yes
Gr23	2 1/4Cr-1.5W-V-Nb		650	No	No	Yes	Yes

The secondary potential candidate materials for GFR are listed in Table 5.2. These materials are considered as secondary candidates mainly because their databases have not been developed for inclusion into the high-temperature nuclear code (ASME BVP Sect. III, Subsect. NH). All of these materials, with the exception of CCA617, have extensive databases.

There are a number of outstanding potential candidates that have not been included in Tables 5.1 and 5.2. Their inclusion depends to a large extent on which option is under consideration. Clearly, for any option, the Co-bearing alloys are to be avoided where radiation fields may be present. Thus, alloys 617 and CCA617 may not be first choices for components located in the immediate vicinity of the reactor vessel. Alloy 230 is a good alternative to alloys 617. Hastelloy XR is low in Co, which provides an advantage over Hastelloy X. These alloys may be adequate for the helium option.

For high-temperature heat exchanger and piping for helium service materials, new alloys, such as SAVE 25, 602CA, HR120, and Sanicro29, could be considered. Generally, these alloys are far from being qualified for Sect III construction, but have good promise.

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. Lacking these options, the austenitic and ferritic steels listed in Tables 5.1 and 5.2 remain the primary and secondary candidates for all three options.

5.3 High-Temperature Strength Issues

For the materials listed in Tables 5.1 and 5.2, there are no outstanding strength issues. However, the materials were never developed with the intent to provide service in nuclear systems for 600,000 hours. Resolution of this issue and other strength-related issues is incorporated into the R&D effort on the NGNP. On the other hand, if the required service conditions are such that corrosion in the CO_2 is temperature-limiting, the selection of the alternate corrosion-resistant alloys could give rise to strength or embrittlement issues of considerable importance.

5.4 Fabrication Issues

There do not appear to be any fabrication issues. Some concerns may need resolution regarding the heavy-section welding of the nickel-base alloys, but, if required, research is currently underway in other projects to develop the required technology.

5.5 Infrastructure Issues

It is difficult to identify potential issues with infrastructure without knowledge of the designs for the components. It is expected that some of the materials and components will be produced overseas, but there is no reason to believe that the high-temperature metallic components cannot be manufactured. Hence, for the GFR materials scoping R&D phase, no actions are required in this area.

5.6 Regulatory and Codification Issues

Most of the materials in Table 5.1 have excellent service experience and databases. Even so, issues remain to be resolved in their current usages. There do not appear to be any issues that would prevent their use in the GFR from a construction code point of view. However, from a regulatory point of view, it is expected that issues will arise that will need attention. Some of these issues are identified in NUREG/CR-5955 and NUREG/CR-6816 [1,2].

5.7 R&D Plan to Establish Materials Viability and Downselect Candidates

The research and development plan for the high-temperature GFR materials assumes that the efforts on the NGNP will be directly applicable. At this point, it is recognized that the materials listed in Tables 5.1 and 5.2 are also in the NGNP plan. 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. The proposed testing temperatures for candidate GFR materials are listed in Table 5.3.

Table 5.3. Testing temperatures and environments for GFR potential candidate high-temperature alloys

Alloy group	Helium Environment	CO_2 Environment
Nickel base	850	600
High alloy	760	600
Stainless steel		600
Martensitic steel	600	550
Low alloy steel		500

Having established materials and conditions, the first logical step is to assess the experience with the alloys or similar alloys. This information then forms a foundation on which to develop an exploratory testing program to gather the data needed to determine feasibility of the GFR concept. Typically, the kind of exploratory mechanical testing includes creep-rupture, fatigue, crack growth, and combinations of the three. The experimental activities for the scoping phase should not be extensive, but rather sufficient to identify significant trends and assess any unexpected viability issues.

References

[1] R. L. Huddleston and R. W. Swindeman, Materials and Design Bases Issues in ASME Code Case N-47, NUREG/CR-5955 (ORNL/TM-12266), Oak Ridge National Laboratory, Oak Ridge, TN (April, 1993).

[2] V. N. Shah, S. Majumdar, and K. Natesan, Review and Assessment of Codes and Procedures for HTGR Components, NUREG/CR-6816 (ANL-02/36), Argonne National Laboratory, Argonne, IL (June, 2003).

6. POWER CONVERSION COMPONENTS

6.1 Operating Conditions

The GFR reference design power conversion system is very similar to that for the NGNP and essentially identical in terms of components, pressures, and temperatures to that for the GT-MHR. The temperature of the GT-MHR He coolant entering the turbine is ~850°C and the temperature at the recuperator inlet is nominally 500°C. Maximum temperatures in the high- and low-pressure compressors and the intercooler and precooler are very significantly lower (<150°C). The two alternate designs utilize supercritical CO_2 at 20 MPa in their power conversion systems. One design has He primary coolant at 600-650°C transferring heat through an IHX to secondary system supercritical CO_2; the CO_2 enters the power conversion turbine at 550°C (indirect Brayton cycle). The other alternate design utilizes a direct Brayton cycle for power conversion with the primary coolant supercritical CO_2 also entering the turbine at 550°C.

6.2 Preliminary Candidate Materials

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.

The other highest temperatures in the GFR reference design power conversion system will be experienced in the first-stage turbine blades and disks. Typically, the disks of the first three stages are cooled to <650°C; the blades are not cooled and maximum metal temperature is in the range 800-850°C. Wrought Nimonic alloys (Ni with about 20 wt.% Cr with additions of Ti and Al and sometimes Mo) are prime candidate alloys for the disks. An example is Nimonic 80A which was developed for service up to 750°C. A large number of similar alloys with comparable properties are also commercially available. The blade material will almost certainly be a cast Ni-base alloy such as Alloy 713LC or IN-100. It should be noted here, however, that the exact materials selected for the disks and blades will likely be highly dependent on the turbine manufacturer selected as each manufacturer has its own favorite materials based on experience and turbine conditions. Further, the materials R&D plan for the NGNP delegated material choice and qualification of the materials chosen to the turbine manufacturer eventually selected.

The recuperator for the 850°C GT-MHR is currently a modular counter-flow He-to-He heat exchanger with corrugated-plate heat exchange surfaces; that for the 850^{o}C GFR reference design will likely be similar. Both would operate with helium inlet from the turbine at ~500°C. Austenitic 300 series stainless steels are the prime candidates for all portions of the recuperator. Examples are 316L and stabilized steels such as 321 and 347.

The blades and disks in the GT-MHR power conversion system high- and low-pressure compressors operate at about 110^{o}C and a Ti alloy with 6%Al and 4%V is the primary candidate alloy. This should also be acceptable for the GFR reference design system. Finally, the precoolers and intercoolers (He-to-water heat exchangers) of both the GT-MHR and GFR reference would operate with maximum He temperatures of 150°C and water temperatures of 60^{o}C. A titanium-stabilized 300 series stainless steel, 321, is the primary candidate alloy for the GT-MHR design.

The materials for the power conversion system components in the two alternate designs should be identical as the projected operating conditions for both are essentially identical. It would be expected that use of the candidate materials for the reference design would be conservative because of the much lower inlet temperatures in the turbine (550°C) and the recuperator (400°C). However, there are two complicating factors that will be discussed below in subsections 6.4 and 6.8 on *High-Temperature Strength Issues* and *Other Issues, resp*.

6.3 Status

All of the candidate materials discussed above are commercially available and recognized by the ASTM and the ASME. Each of the materials has a long history of satisfactory service in a number of industrial applications.

6.4 High-Temperature Strength Issues

The candidate structural alloys for the various components of the GFR reference design power conversion system have sufficient high temperature strengths for their anticipated service. The most highly stressed component will be the turbine with the maximum temperature approaching 850°C. Materials for this application must have sufficient resistance to creep and to both high- and low-cycle fatigue. The turbine and turbine inlet shroud candidate materials discussed above possess these attributes. The only other high-temperature component is the recuperator (500°C). All of the 300 series stainless steels mentioned as candidates for this application have strengths sufficient for the component.

The recuperator and the turbine in the alternate supercritical CO_2 designs operate at 400°C and 550°C, respectively. The only significant strength issue may be relative to the turbine materials. Although no details of the design are available for examination at present, it is expected that the turbine will be substantially higher-speed and more compact relative to the existing He turbine designs. First indications are that the turbine for a 300MWe turbine design with supercritical CO_2 could be enveloped in 1-m length by 1-m diameter; the current equivalent He-turbine is approximately 2-m in length by 2-m in diameter operating at 4400 rpm. Given the anticipated higher turbine speed, both static creep loads and fatigue stresses and cycles could be more severe in the turbine operating in supercritical CO_2. This cannot be quantified at present, since mature designs for CO_2 turbines do not exist. However, the much lower temperature and the smaller diameter in the supercritical CO_2 turbine should compensate for potentially significantly higher static loads. Temperature compensation of fatigue loadings is less certain.

6.5 Fabrication Issues

There are no fabrication issues for the materials/components of the GFR power conversion system that have not already been considered for the NGNP power conversion system. None of the issues are major but include such things as demonstrating manufacturing processes (forming and welding) for the turbine shroud and proving the capability for manufacturing very high quality large, thin sheets of 300 series stainless steel for the recuperator.

6.6 Infrastructure Issues

No significant modifications or additions to commercial infrastructure will be needed to provide the materials and components for the GFR reference design or the two alternate designs using supercritical CO_2 other than developing industrial capabilities for production of supercritical CO_2

turbines. While this may not be exceptionally challenging from a technical point of view, at the current time, there are no manufacturers of commercially available supercritical CO_2 turbines.

6.7 Regulatory And Codification Issues

Codes and standards needed for the GFR reference design and alternate designs have not yet been established. Materials data requirements and design methods will depend on whether these components are constructed to ASME Sect. III and associated code cases or to alternative sections. Regardless of which construction rules are utilized, the ASME and regulatory bodies will require that the designer consider all likely modes of failure including those related to degradation from thermal aging and environmental effects. This is not anticipated to be activity required during the scoping phase of the GFR materials R&D program. Identification of regulatory and codification issues for the qualification phase expect to subsequently follow will be identified.

6.8 Other Issues

The oxidation potential provided by impurities expected in the He coolant of the GFR reference design could be higher than that experienced in the He coolant of the GT-MHR and the NGNP. The graphite present in these latter two systems reacts with oxidizing species to minimize the oxidant level in the He exiting the reactor core. With respect to power conversion system components, this would likely be of potential concern only for materials at the highest temperatures in the turbine. Further, relatively simple engineering solutions are available to control oxidant levels.

Possibly the most important issue relative to the materials for the GFR alternate design power conversion systems is that of their compatibility with supercritical CO_2. Maximum temperatures in these systems will be ~550°C; preliminary calculations suggest that operating temperatures must be limited to this maximum to have any chance of achieving lifetimes as long as 10-years. However, this is probably acceptable for turbine blades, etc., which are generally inspected/replaced on a 10-year or shorter cycle. Longer lifetimes would be anticipated for materials applied in other power conversion system components operating at lower temperatures. An additional consideration in the direct Brayton cycle supercritical CO_2 design is the possibility that radiolytic species coming from the reactor core may negatively influence compatibility. This, however, is likely not a critical feasibility issue and, further, cannot be addressed with confidence at this stage of design.

6.9 R&D Plan to Establish Material Viability and Down-Select Candidates

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 as indicated in Table 6.1. 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.

Table 6.1. Power Conversion System Materials Compatibility Test Matrix for Alternate GFR Designs

Test Temp. °C	Power Conversion System Component			
	Turbine Inlet Shroud	Turbine Blades	Turbine Disks	Recuperator
350				X
400				X
450			X	X
500	X	X	X	X
550	X	X	X	
600	X	X	X	
650	X	X		

7. GENERAL MATERIALS COMPATIBILITY CONSIDERATIONS IN GFR ENVIRONMENTS

7.1 Operating Conditions

The GFR reference design, like thermal-spectrum helium-cooled reactors such as the GT-MHR and the PBMR, uses a direct-cycle helium turbine for electricity generation and can use process heat for thermochemical production of hydrogen. This reference design shares many materials' requirements in common with the NGNP. However, the temperatures and composition of the environment are somewhat different. One alternate design also uses helium-cooled system with an indirect Brayton cycle for power conversion. The secondary system of this alternate design utilizes supercritical CO_2 at 550°C and 20 MPa. A second optional design is a supercritical CO_2 cooled (550°C outlet and 20 MPa), direct Brayton cycle system. From a corrosion viewpoint, the pressure vessel will operate in air and the internals of reactor will operate in either in helium or supercritical CO_2 environments.

For the helium-cooled reactor, it is expected that:

- Inlet/outlet temperatures will be 550/850°C;
- Surface temperatures of materials in the core in contact with the coolant during normal operation will be in the range of 800 to 1000°C; and
- Surface temperature of materials in the core in contact with the coolant under accident conditions will be in the range 1400 to 1600°C for approximately 6 hours (time required for the temperature to rise from normal operating to accident peak and return to near normal operating temperature).

For the supercritical CO_2-cooled reactor, it is expected that:

- Inlet/outlet temperatures will be 550/650°C;
- Surface temperatures of materials in the core in contact with the coolant during normal operation will be approximately 650°C; and
- Surface temperature of materials in the core in contact with the coolant under accident conditions will be approximately 1000°C for approximately 6 hours (time required for the temperature to rise from normal operating to accident peak and return to near normal operating temperature).

7.2 Helium Environment

The interactions between structural materials in controlled-impurity helium atmospheres associated gas cooled reactors have been the subject of numerous investigations [1]. The results of these studies conducted by various organizations in USA, Germany, England, Norway, Japan, and other places have demonstrated the importance of small changes in impurity levels, high temperatures and high gas flow rates. Metallic materials can be carburized or decarburized, and oxidized internally or at the surface. These corrosion reactions, depending on the rate, can affect long-term mechanical properties such as fracture toughness.

The simulated advanced HTGR helium chemistries used in various test programs are shown in Table 7.1. Because of the low partial pressures of the impurities, the oxidation/carburization potentials at the metallic surface of a gas mixture are established by the kinetics of the individual impurity catalyzed reactions at the surface. As shown, the main impurities are H_2, H_2O, CO and CH_4. The hot graphite core in an HTGR is assumed to react with all free O_2 and much of the CO_2 to form CO, and with H_2O to form CO and H_2. In addition, in cooler regions of the core, H_2

reacts with the graphite via radiolysis to produce CH_4. Because of the change in surface temperatures around the reactor, and associated changes in reaction mechanisms and rates of reaction on bare metal versus on scaled surfaces, reaction rates and order of reactions are important.

Table 7.1. Composition helium environments (advanced HTGR) used in past tests

Program	H_2 (µatm)	H_2O (µatm)	CO (µatm)	CO_2 (µatm)	CH_4 (µatm)	N_2 (µatm)	He (atm absolute)
NPH/HHT	500	1.5	40		50	5–10	2
PNP	500	1.5	15		20	<5	2
AGCNR	400	2	40	0.2	20	<20	2
NPH: Nuclear process heat HHT: High temperature helium turbine systems PNP: Prototype Nuclear Process Heat AGCRNR: Advanced Gas Cooled Nuclear Reactor							

Because of there being little or no graphite in the proposed GFR reactor, the composition of the helium environment may be somewhat different from those for which materials test data are available. Assuming zero graphite, the GFR environment should contain near zero levels of CH_4, less CO_2 and CO, about the same amount of nitrogen, and more moisture and oxygen than previous helium cooled reactors. However, the materials' surface temperatures are within the range of previous tests. Because it is possible to treat a side stream of the helium environment to reduce the oxygen and moisture, it is very likely that the GFR helium environment can be controlled to compositions very similar to that of previous reactors, if desired. As such, the materials' performance issues are mostly known.

The overall stability of the proposed helium environment must be evaluated in order to ensure that testing proposed in various sections of the program are performed in environments that have consistent chemical potentials. In addition, the corrosion of metals and nonmetals 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.

7.3 Supercritical CO_2 Environment

The chemical potential of the alternate supercritical CO_2 environment will, at least from a thermodynamic viewpoint, be oxidizing. It is also possible that under certain conditions, the environment may be carburizing. The long-term performance of materials under the oxidizing and/or carburizing conditions must be established for the supercritical CO_2 environment at temperatures relevant to the GFR, where little data currently exist. Corrosion of metals and nonmetals will be evaluated to establish baseline data. These tests will be performed at temperatures to include at least 50°C above the proposed operating temperature. In addition, the spalling, transport, and deposition of radiological corrosion products must be evaluated for the direct supercritical CO_2 Brayton cycle system.

7.4 Testing to Establish Feasibility

Helium

It is expected that the materials performance needs for the GFR in helium will be largely covered by the work needed for the VHTR 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.

Supercritical CO_2

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 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.

Test Program

The tests proposed in this section are in addition to environmental mechanical properties and thermal-physical properties testing proposed in other sections of this feasibility study.

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 that the expected exposure temperatures.

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 an 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.

Because choices of materials are still be modified, the proposed test matrix contained in Table 7.2 will be identified by materials application rather than specific materials.

Table 7.2. Materials test matrix

Materials application	Environment
High dose tolerant metals	Helium
Ceramic internal	Supercritical CO_2
Inert fuel matrix ceramics	Supercritical CO_2
Metallic internal	Supercritical CO_2
Pressure vessel cladding	Supercritical CO_2
Lift-off/plating experiments	Supercritical CO_2
Ceramic internal	In-reactor supercritical CO_2
Metallic internal	In-reactor supercritical CO_2
Pressure vessel cladding	In-reactor supercritical CO_2

References

1. O. F. Kimball and D. E. Plumblee, *Gas/Metal Interaction Studies in Simulated HTGR Helium,* HTGR -85-064, General Electric Company, Schenectady, New York, June 1985.

8. HIGH TEMPERATURE DESIGN METHODOLOGY

8.1 Introduction

The impact and requirements of high temperature design methodology (HTDM) and possible codification needs will vary for each of the three proposed GFR designs. Earlier sections in this report adequately cover these conditions. Relevant to HTDM and codification, metallics and non-metallics have similar and different design and codification requirements. HTDM and codification of materials and components that operate inside vs. outside the high radiation field will differ also. Likewise, HTDM requirements for power conversion components will differ. Several materials may be used in more than one design, although use conditions may differ; consequently, the HTDM requirements may vary accordingly. Regardless, the basic framework for HTDM will be the same for all materials and designs.

8.2 Non-metallic Components

Non-metallics may be used in the reactor core and internals for components such as reflector, control rod guides, and upper and possibly lower support plates. As in the NGNP plans, although uncertain, no codification and HTDM is believed to be required for composites, ceramics, carbon, and graphite components for the GFR. NGNP does include plans to address codification issues for graphite as it is used more extensively in the VHTR. Otherwise, the use of carbon for GFR is limited since carbon is a strong neutron moderator. An ASME code section that addresses requirements to use ceramics or composites for the GFR are encompassed partially in the NGNP plans along with additional efforts discussed in Section 2 of this report. Both the NGNP and GFR plans will address composites and ceramic material requirements under the umbrella of ASTM standards. Consequently, HTDM and codification issues for the GFR will be restricted to the requirements of metallic materials.

8.3 Metallic Components

Metallics are considered for use in the power conversion, reactor core, reactor internals, pressure vessel, piping and heat exchangers. Each is addressed separately as follows.

Power Conversion Components
ASME Section III codification is not believed to be required for power conversion components. As in the NGNP qualification program, the materials R&D plan for delegated materials selection and qualification will be made by the turbine manufacturer; notwithstanding, the assessment of viability of preliminary candidate materials for use in supercritical CO_2 is included in the GFR plans as stated earlier.

Pressure Vessel, Piping, and Heat Exchanger
The GFR HTDM and codification requirements for pressure vessel, piping, and heat exchangers are included in the NGNP plans. The nature of the GFR will result in significantly higher doses of radiation to core and reactor internals than the NGNP designs. Although the GFR pressure vessel will experience a higher dose level than the NGNP pressure vessel, the primary candidate pressure vessel materials response is reasonably well understood at the doses anticipated. Similarly, the operating conditions of piping and heat exchangers, where intense radiation exposure is not present, are within the envelope of the NGNP designs. No additional work will be required in this area to establish GFR viability unless alternate materials are required.

Core Components and Reactor Internals
Significantly higher doses of radiation to core and reactor internals will occur relative to NGNP components. This in itself, even in cases where the same materials as proposed for the NGNP designs will be used, requires substantial R&D to assess viability. Further, estimated normal and off-normal operating temperatures are much higher than in the NGNP designs. This is a significant challenge.

Reference Design
The use of low swelling austenitic stainless steels and advanced versions of 9 Cr ferritic-martensitic steels may be candidates for the lower support plate, portions of the core barrel, as discussed in Section 3. Modified 9Cr-1Mo is already code approved for normal operating conditions of pressure vessel materials in ASME Section III, Subsection NH up to 538°C, see Table 8.1; NGNP plans include efforts to increase the approved temperature limits to 600-650°C. However, irradiation effects and the elevated temperatures need to be investigated. Subsection NH does not address radiation effects, but requires the owner, designer, and operator to address. Similarly, there is a substantial database on 304 and 316 stainless steels, including some data in helium and extensive data in irradiation environments. Both are code approved to 704°C as shown in Table 8.1. The high doses, up to 100 dpa, will require additional mechanical properties R&D. Additional efforts to verify that no issues for high-temperature design and codification exist will be required, i.e. efforts to address ratcheting, creep, creep-fatigue, weldments, mutiliaxial effects, etc., at elevated temperatures and at high levels of irradiation.

For other components such as the reflector, control rod guides, and upper support plate, non-metallics will be required. Ceramics are most likely to be used for core internals. However, the use of refractory alloys, intermetallics, or ODS ferritic-martensitic alloys postulated in Section 3 have not been ruled out. Since the use of carbon in the core will be minimal, levels of oxygen contamination are expected to be moderate to high, coupled with the 60-year lifetime, it is unlikely that refractory alloys can perform. NGNP plans include a small effort to address viability of these alloys. Again, this effort does not include irradiation levels indicative of the GFR; hence, additional validation efforts that address ratcheting, creep, creep-fatigue, weldments, flaw sensitivity, etc. will be required.

Table 8.1. Subsection NH Materials and Maximum Temperatures

Material	Temperature (°C)	
	Primary stress limits and ratcheting rules	Fatigue curves
304 stainless steel	816	704
316 stainless steel	816	704
2 1/4 Cr – 1 Mo steel	649[a]	1100
Alloy 800H	760	760
Modified 9 Cr – 1 Mo steel	649[a]	538

[a]Time above 593°C limited to 1000 h.

Alternate Design 1
This He/S-CO_2 design does include significantly lower temperatures and will prove to have a significant advantage on stress allowables, i.e. materials in the reference design will be less challenged. However, the use of stainless steels and ferrittic-martensitic steels for other internal

components is not possible. Validation efforts for the GFR reference design will cover requirements for this design.

Alternate Design 2
The S-CO_2 direct cycle offers substantially lower temperatures, 400-550°C for the lower support plate and core barrel, and 400-700°C for the reflector and control rod guides. The lower temperature range is close to the limits of ASME Section III, Subsection NB (371°C for ferritics and 427°C for austenitics) where stress allowables are considered time-independent, and surely covered by Subsection NH as discussed earlier. Hence, in that respect the materials are far less challenged than in the previous design concepts. However, the S-CO_2 environment is a critical factor that is not taken into account. Virtually no data exists in this environment, not to mention in the irradiated condition. Significant efforts will be required to validate compatibility of austenitics, ferritics, ODS, and intermetallics materials in these environments, as discussed in sections 3 and 7. Similarly, scoping efforts will be required to address compounded effects of environment, irradiation, creep, fatigue, creep-fatigue, flaw tolerance (crack growth and fracture toughness). These requirements are obviously in addition to those of the other GFR designs and the NGNP designs.

8.4 HTDM Framework and Applicability

The framework for HTDM is taken from the current ASME B&PV Code Section III, Subsection NH; this framework considers both time independent and time dependent material behavior. The same framework is planned to be extended and modified for 617 and 9Cr-1Mo [1] at temperatures where the materials exhibit no distinction between time dependent plasticity and creep. The framework has already been successfully extended to the durability of composites for automotive applications by Corum et al. [2-4] Clearly, the applications and family of materials are quite different than the typical metals used in reactors; however, time independent and time dependent material behavior and failure modes existed.

A brief summary of the approach encompasses the following, though the content need not be restricted to that which is listed:

- Elastic and creep properties for design analysis
- Design allowables for static loading
    - Short-time tensile stresses
    - Time-dependent tensile stresses
    - Compressive and biaxial stresses
    - Membrane and bending stresses
    - Treatment of increment of changing loads
- Design limits for cyclic loadings
    - Fatigue design curve
    - Effects of temperature
    - Fluid effects
    - Varying stress amplitudes
- Damage tolerance assessment
    - Circular holes
    - Circular holes and cracks
    - Impact damage
    - Strength and stiffness degradation

The ability to apply the framework of Subsection NH to automotive composites speaks to the validity of the approach. Similar approaches may be used for ceramics and composites. Obviously, statistical approaches that account for distributions and variation in properties, flaws, etc. can and should be employed; Corum et al. used statistical approaches to quantify size effect in randomly reinforced chopped-strand composites on stiffness and strength [4]. Modifications can be implemented to address effects of environment and irradiation as needed.

8.5 Required Experimental and Analytical Activities

Assuming that the bulk of HTDM needs for GFR will be covered by activities already planned for the NGNP, the following tasks will remain to establish viability.

- Evaluate methods, existing data, and assist in planned test activities of pressure vessel materials and metallic core internals and reactor internals specific to GFR to gain material (creep, fatigue, creep-fatigue) properties required for HTDM.
- Evaluate the results of testing for GFR, propose a method to address variation in material properties of pressure vessel material with thickness for high temperature design (section NH).
- Evaluate the need and assess the available damage models and life prediction approaches (creep, creep-fatigue) to address 60 year design service life (aging effects) with available data, and extrapolation of data for such long periods, for both base and weld metals (pressure vessel, core and reactor internal materials). Develop or propose appropriate models for high temperature design.
- Analyze and simulate component-like parts under representative loadings, irradiation exposure and times for high temperature service. Determine if issues arise regarding ratcheting, multiaxial effects, creep, and creep-fatigue; develop high temperature design methods and rules to avoid deleterious issues.
- Participate in required ASME Code meetings to guide and implement HTDM activities.

References

[1] Next Generation Nuclear Plant Materials Selection and Qualification Program Plan, INEEL/EXT-03-01128, Revision 0, November 7, 2003.

[2] J.M. Corum et al., *Durability-Based Design Criteria for an Automotive Structural Composite: Part 1. Design Rules*, ORNL-6930, Oak Ridge National Laboratory, Oak Ridge, Tenn., February 1998.

[3] J.M. Corum, *Durability-Based Design Criteria for a Quasi-Isotropic Carbon-Fiber Automotive Composite*, ORNL/TM-2002/39, Oak Ridge National Laboratory, Oak Ridge, Tenn., March 2002.

[4] J.M. Corum, , *Durability-Based Design Criteria for a Chopper-Carbon-Fiber Automotive Composite*, ORNL/TM-2003/86, Oak Ridge National Laboratory, Oak Ridge, Tenn., May 2003.

9. SUMMARY OF MATERIALS STUDIES, ASSOCIATED SCHEDULES, AND FUNDING REQUIRED TO DEMONSTRATE VIABILITY OF GFR

9.1 Summary of GFR Development Costs

The needed materials development tasks, schedules, and costs to assess the viability of the GFR are detailed in Section 9 of this report and summarized in Table 9.1.

Table 9.1 Summary of Funding Requirements for the GFR Materials R&D Viability Program

Task	FY05	FY06	FY07	FY08	FY09	FY10	TOTAL
Ceramic Internals	2,000	4,800	6,450	7,400	7,500	6,500	34,650
Metallic Internals	1,100	3,600	5,700	6,800	5,900	4,900	28,000
RPV	900	900	900	900	500	500	4,600
High-Temperature Metallic Components	0	460	700	600	550	350	2,660
Power Conversion System	200	450	750	750	750	300	3,200
Materials Compatibility	0	1,200	3,400	6,200	5,000	2,900	18,700
High-Temperature Design Methodology	50	200	600	1,250	1,350	1,150	4,600
TOTAL	4,250	11,610	18,500	23,900	21,550	16,600	96,410

The total cost estimate for development of the needed materials for the GFR is about $96 million dollars. The funding specifically required for the GFR materials studies can be significantly reduced if (1) existing university facilities are used, (2) the costs are shared with our international GIF partners, and/or (3) the costs are shared with other Generation IV reactor development programs. Note that these costs are for "viability" research and development as defined in the Generation IV Roadmap (GIF 2002). Viability research and development examines the feasibility of key technologies and is that R&D necessary for proof of the basic concepts, technologies, and relevant conditions.

9.2 Ceramics for Core and Internals Applications

The cost breakdown below makes a number of approximations as to the level of effort necessary to carry out the GFR ceramics scoping work. The primary assumption is that in the first phase of GFR, which will be carried out over through 2010, the goal is to prove the viability of a few materials in each class. Once this Phase 1 proof of principal is carried out, a set of candidate ceramics materials can be recommended for Phase 2 with a high degree of confidence.

Table 9.2 Funding Requirements for GFR Ceramic Core and Internals Components

Task	FY05	FY06	FY07	FY08	FY09	FY10	TOTAL
Insulating Ceramics							
Dimensional Stability Under Irradiation			400	500	500	500	1,900
Environmental Effects		100	200	200	200	100	800
Structural Ceramics							
Mechanical & Physical Properties Tests	500	700	700	1400	1500	1800	6,600
Thermal/Dimensional Properties Under Irradiation		400	600	600	600	600	2,800
Environmental Effects		400	500	500	500	300	2,200
Ceramic Composites							
High Dose Thermomechanical and Dimensional Properties	800	1,800	2,000	2,000	2,000	1,800	10,400
Processing and Properties	600	800	800	800	800	600	4,400
Environmental Effects		100	200	200	200	100	800
ASTM Standards Development	50	50	50	50	50	50	300
Carbon Composites							
Baseline Materials Testing and Characterization		300	450				750
Full Scale Testing and Verification				400	400		800
Irradiated Materials Evaluations			300	500	500	500	1,800
Environmental Effects		100	200	200	200	100	800
ASTM Standards Development	50	50	50	50	50	50	300
TOTAL	2,000	4,800	6,450	7,400	7,500	6,500	34,650

Specifically, assessments of the materials compatibility of insulating and structural ceramics as well as ceramic and carbon composites will be made with helium and S-CO_2 environments. Mechanical and thermophysical properties screening studies as a function of irradiation will be performed, including standards formulations for testing of the structural composites.

9.3 Metallic Materials for Core and Internals Applications

A comprehensive assessment of the current status of advanced austenitic stainless steels and ferritic/martensitic steels will be carried out focusing on materials with proven potential for swelling resistance and satisfactory high temperature mechanical behavior. A limited set of selected materials will form the basis of a program to evaluate mechanical behavior, long-term microstructural stability and radiation resistance, focusing on pertinent GFR conditions. Following an evaluation of the radiation effects data base on Ni alloys, low dose neutron irradiation experiments will be carried out to assess various strategies for reducing the susceptibility to grain boundary embrittlement. The potential application of refractory metal alloys will be assessed and work carried out to investigate the potential for current and improved nano-structured Fe- and Ni-base alloys with properties specifically tailored to GFR conditions.

Table 9.3 Funding Requirements for GFR Metallic Internals

Task	FY05	FY06	FY07	FY08	FY09	FY10	TOTAL
Refractory metal alloys assessment	200	300					500
Assess advanced austenitic and F-M steels and Ni alloys; select candidate alloys	300	600	600				1,500
Determine baseline mechanical properties and long term microstructural stability	300	700	400	200			1,600
Assess environmental effects on mechanical properties of candidate alloys			400	600	800	600	2,400
Assess irradiation effects on candidate advanced austenitic and F/M steels		1200	3000	4200	3600	3000	15,000
Assess irradiation effects on candidate radiation-resistant nickel-based materials		400	1000	1400	1200	1000	5,000
Assesement and development of Fe, Ni and RM base nano-structured materials	300	400	300	400	300	300	2,000
TOTAL	1,100	3,600	5,700	6,800	5,900	4,900	28,000

9.4 Reactor Pressure Vessel Materials

Although many of the research needs for the GFR RPV will be undertaken within the research scope of the NGNP, there are some differences between the operating conditions that will require GFR-specific research. Moreover, there are significant uncertainties regarding those conditions for the three different GFR designs, primarily related to the greater radiation exposure of the

Table 9.4 Funding Requirements for GFR RPV

Task	FY05	FY06	FY07	FY08	FY09	FY10	TOTAL
Baseline Materials	400	400	400	400			1,600
Aging	100	100	100	100	100	100	600
Irradiation Effects	400	400	400	400	400	400	2,400
TOTAL	900	900	900	900	500	500	4,600

vessel materials. Most of the ferritic-martensitic steels being considered have good radiation resistance in the anticipated temperature regime and to the anticipated radiation dose. However, specific radiation experiments will be required at the specific design conditions to ensure that the potential candidate materials will perform adequately under GFR conditions.

9.5 High-Temperature Metallic Materials

Since the operating temperature conditions for the GFR metallic high-temperature structural materials are expected to be within the limits of existing ASME construction codes, the needs for viability research will largely focus on environmental testing to assist in the estimation of corrosion allowances and the assessment of the impact of corrosion on component performance. To these ends, static and dynamic testing in representative environments will be required. It is expected that the research effort on helium contamination effects in the NGNP program will be adequate to assess the viability of the GFR concepts. The CO_2 effects are unique to the He-S/CO_2 indirect and all-S-/CO_2 options, however, so some exploratory creep-rupture, creep crack growth, fatigue, and creep-fatigue testing in CO_2 will be needed. An estimate of the funding for the exploratory testing of metallic candidate materials is provided below.

Table 9.5 Funding Requirements for GFR High-Temperature Metallic Components

Task	FY05	FY06	FY07	FY08	FY09	FY10	TOTAL
Assess CO_2 effects on creep-rupture and creep crack growth		220	300	200	200	100	1,020
Assess CO_2 effects on fatigue and fatigue crack growth		240	400	400	350	250	1,640
TOTAL		460	700	600	550	350	2,660

9.6 Power Conversion System Materials

Potential materials for the alternate concept power conversion systems' turbines and recuperators 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.

Table 9.6 Funding Requirements for GFR Power Conversion Materials

Task	FY05	FY06	FY07	FY08	FY09	FY10	TOTAL
Baseline Materials Assessment	100	200	250	250	250		1,050
Environmental Exposure in CO_2	100	150	150	150	150		700
Post Exposure Materials Assessment		100	350	350	350	300	1,450
TOTAL	200	450	750	750	750	300	3,200

9.7 Materials Compatibility

Tests are needed to establish the viability of materials performance in the proposed GFR environments, both helium without graphite and supercritical CO_2. Test will be performed to determine the possibility of helium gas cleanup. If cleanup is possible, the helium environment, most likely, will be similar to the previous test environments and hence, data from the previous test programs can be used to support viability determinations. In addition, compatibility tests are needed to ascertain the performance of materials that were not previous evaluated. Because of the lack of information, a larger suite of tests are needed for the supercritical CO_2 environment. Besides materials compatibility information, lift-off/plating studies of corrosion products are required. The latter studies require the use of loop that can attain the appropriate velocities of supercritical CO_2 at test temperatures. The tests proposed in this section are in addition to mechanical or physical properties testing in the specific gaseous environments already included in other sections of this feasibility study. Additionally, tests of both the chemistry produced in the S-CO_2 by in-core radiolysis and assessment of its effects on candidate materials will be required.

Table 9.7 Funding Requirements for GFR Materials Compatibility

Task	FY05	FY06	FY07	FY08	FY09	FY10	TOTAL
Ê Helium							
Helium loop (recirculating, low velocity)		200	300				500
Helium side stream cleanup studies			300	400			700
Helium corrosion studies				200	400	300	900
Supercritical CO_2							
Supercritical CO_2 corrosion test loop (low velocity)		300	200				500
Corrosion performance of proposed materials			200	400	400	200	1,200
Supercritical CO_2 lift-off test loop (high velocity)		200	600	900	300		2,000
Lift-off and plating performance of materials				300	500	600	1,400
Supercritical CO_2 in-reactor loop (low velocity)		500	500	500			1,500
Supercritical CO_2 in-reactor loop corrosion studies			500	1500	1800	1200	5,000
Supercritical CO_2 in-reactor loop chemistry studies			800	2000	1600	600	5,000
TOTAL	0	1,200	3,400	6,200	5,000	2,900	18,700

9.8 High Temperature Design Methodology

The GFR HTDM and codification requirements for pressure vessel, piping, and heat exchangers are included in the NGNP plans. However, the metallic pressure vessel will experience higher dose levels than the NGNP design, as will the reactor vessel internals, core, and core internals. Further, HTDM for NGNP did not include any efforts for core internals or core supports. Hence, the following research must be conducted to assess the viability of materials for the GFR.

Detailed inelastic analysis must be conducted. This will help designers assess the limitations of the vessel internals materials with respect to time-independent, time-dependent, ratcheting limits, accelerated creep damage, creep-fatigue, creep buckling, flaw sensitivity (fracture toughness) and multiaxial effects. Further, the same issues must be examined for possible deleterious effects due to the high radiation levels. Scoping tests will be conducted and compared with analytical and numerical predictions or irradiated vs. unirradiated material. These efforts will apply to 2 1/4Cr, the modified 9Cr alloys, and may be extended to one of the best candidates the class 12Cr or 3Cr alloys. Similar efforts will be needed to asses the viability of ODS, intermetallics and the ferritic-martensitic alloys for core components and reactor internals.

Table 9.8 Funding Requirements for GFR High Temperature Design Methodology

Task	FY05	FY06	FY07	FY08	FY09	FY10	TOTAL
Supplemental Testing		150	300	800	800	600	2,650
Analytical Methods Development			250	400	500	500	1,650
Codes and Standards Interactions	50	50	50	50	50	50	300
TOTAL	50	200	600	1,250	1,350	1,150	4,600