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Program on Technology Innovation: Material Property Assessment and Data Gap Analysis for the Prospective Materials for Very High Temperature Reactors (VHTRs) and Gas-Cooled Fast Reactors (GFRs)

2020 TECHNICAL REPORT

Program on Technology Innovation: Material Property Assessment and Data Gap Analysis for the Prospective Materials for Very High Temperature Reactors (VHTRs) and Gas-Cooled Fast Reactors (GFRs)

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Final Report, October 2020

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ABSTRACT

This report represents one of four major reports prepared by the Electric Power Research Institute on advanced reactor (Generation IV) design materials knowledge gaps and actions. Very high temperature reactor (VHTR) and gas-cooled fast reactor (GFR) designs are highlighted in the report. Two proposed and developing VHTRs, the AREVA/NGNP Alliance Reactor and the X-energy Xe-100 Reactor, as well as the General Atomic GFR, the Energy Multiplier Module (EM^2), are discussed in detail. Both VHTRs are helium-cooled reactors with a gas outlet temperature of 750°C. The AREVA/NGNP system uses the prismatic (graphite blocks) design with TRISO fuel particles, producing 625 MWt, whereas the X-energy system uses a pebble-bed design for the core and with TRISO fuel particles embedded in larger graphite particles to produce 200 MWt with a four-pack module plant. Both VHTRs include three major components: (1) a helium-cooled nuclear reactor, (2) a heat transport system (steam generator), and (3) a cross vessel for helium flow from the reactor to the heat transport system. The GFR EM^2 is a high temperature, direct Brayton cycle, helium-cooled fast reactor with an outlet temperature of 850°C to produce 500 MWt, designated by an advanced small modular reactor by General Atomics.

Materials compatibility issues, as well as advanced materials that might be considered in future designs, are discussed. Specific knowledge gaps and actions for each design are covered.

Keywords

Gas-cooled fast reactor
Helium-cooled reactor
Materials knowledge gaps
Very high temperature reactor

EXECUTIVE SUMMARY

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Product Title: Program on Technology Innovation: Material Property Assessment and Data Gap Analysis for the Prospective Materials for Very High Temperature Reactors (VHTRs) and Gas-Cooled Fast Reactors (GFRs)

PRIMARY AUDIENCE: Scientists, engineers, and developers of advanced reactors interested in the development, deployment, and operation of very high temperature reactors (VHTRs) and gas-cooled fast reactors (GFRs)

SECONDARY AUDIENCE: Utility staff who are interested in the next generation of reactors

KEY RESEARCH QUESTION

The technological diversity of VHTRs and GFRs was examined to develop a better understanding of what concepts might move forward over the next decade. Additionally, key material integrity and development issues affecting each type of reactor were examined.

RESEARCH OVERVIEW

The scientific literature was reviewed to gain an understanding of the designs of VHTR and GFR concepts that have been proposed and to develop an understanding of the materials degradation processes that are associated with each design.

KEY FINDINGS

- Type 316FR material, which will be used in GFRs, has improved properties relative to Type 316 for a projected longer life. Type 316FR is not currently approved in the ASME Code. A reasonably large database of relevant properties has been established, which should be useful in gaining ASME Code approval of this alloy.
- Low alloy and 2.25Cr-1Mo steel mechanical and fracture properties after VHTR and GFR fast neutron irradiation design conditions are not known. Low alloy and 2.25Cr-1Mo steel appear promising for reactor pressure vessel (RPV) applications, but materials degradation acceptance limits under fast fluence are expected to be lower than the 1 x10^{19} n/cm^2 for which low alloy steel is used in light water reactors. End-of-life information under irradiation conditions will be required.
- The use of SA-508/533 material at temperatures above 371°C in the RPV for a 60-year lifetime will require consideration of creep effects.
- Metallic materials will be operated at high temperatures, significantly in excess of those for which the materials are currently code allowed for nuclear construction. High-temperature tensile properties, creep strength, and swelling resistance must also be evaluated for fatigue and creep-fatigue behavior under the conditions anticipated. Creep-fatigue is a significant issue for Grade 91 and other ferritic-martensitic steels.

WHY THIS MATTERS

This report lays the foundation for future testing and assessment of high temperature alloys that will be used for VHTR and GFR materials in future reactors.

HOW TO APPLY RESULTS

The results of the report will be used by the Electric Power Research Institute (EPRI) and other researchers to identify and assess new alloys for use in VHTR and GFR applications.

LEARNING AND ENGAGEMENT OPPORTUNITIES

Members of the EPRI Advanced Nuclear Technology program will be interested in the broad overview of the report, as well as the review of potential alloys considered for VHTR and GFR applications.

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UNIT CONVERSIONS

Temperature Unit Conversions – see Appendix A

Unit	x Factor =	Conversion
MPa	145	psi
GPa	145.0×10^{3}	ksi
kg/s	2.205	lb/s
m	39.37	inch
mm	0.03937	inch
m^2	2.471×10^{-4}	acres
m^3	264.2	gallons
g/cm^3	62.43	lb/ft^3
mbarn	1×10^{-27}	cm^2
W/m-K	0.5782	BTU/h-ft-^{o}F
t (ton)	907.2	kg
MPa($m^{1/2}$)	910.1	psi($in^{1/2}$)

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INTRODUCTION

As recently stated to a committee of the United States Senate: “Advanced reactors, including small modular reactors, hold great promise as a clean, reliable, and secure power source for our nation. The Department recognizes that advanced reactors face challenges to ultimately achieving commercialization. In addition to early-stage research and development the Administration supports prioritized investments in nuclear energy research infrastructure to enable private sector innovation.” [1] Advanced reactors include those designated Generation IV reactors with specific systems part of the Generation IV International Forum (GIF), with the very-high temperature reactor (VHTR) and the gas-cooled fast reactor (GFR) being two of the six included systems. This section of the report presents a summary of reactor system designs, identification of the primary components, the proposed operating conditions, and comprehensive information regarding structural materials requirements for those primary components of two specific VHTR systems and one GFR systems. Based on the available information, the technical gaps that may exist relative to selection and deployment of the materials for those systems are identified. This report provides an overview of the both VHTR and GFR designs (Section 2 and 3), materials compatibility and advanced materials (Section 4 and 5), and a review of materials knowledge gaps (Section 6). The reader will find Section 6 of particular interest as it summarizes gaps and provides actions for closing these gaps for both VHTRs and GFRs.

1.1 Background and Review of Gas-Cooled Fast Reactors

With a view towards achievement of somewhat higher coolant outlet temperatures for higher overall efficiency than that offered by water-cooled systems, gas-cooled reactors (GCR) began operation in about 1956 with the CO_2-cooled Magnox design in the United Kingdom (U.K.) and the UNGG (Uranium Naturel Graphite Gaz) design in France, both Generation I systems [1]. The Magnox system (with outlet temperatures to about 400°C) was subsequently replaced in the U.K. by the CO_2-cooled Generation II Advanced Gas-cooled Reactor (AGR) (with outlet temperatures to about 640°C), which began commercial operation in 1976 [2]. Some AGRs are still in operation (2018). Generation II High Temperature Gas-Cooled Reactors (HTGR) have been in operation producing electricity since the Peach Bottom HTGR began operation in the U.S. in 1966 [3] but were first proposed in 1942 by Professor Farrington Daniels of the University of Wisconsin [3] and published in 1944 while working at what is now Oak Ridge National Laboratory (ORNL) [4]. The Dragon Plant in the U.K. began in 1964 with a coolant outlet temperature of 750°C, but it was a test reactor and did not produce power [5]. Relative to the Generation I GCRs, the HTGRs are characterized by higher outlet temperatures (700-950°C), use of helium as the primary coolant and graphite as the moderator. Table 1-1 (after Ragheb [6]) provides a comparison of operating conditions for a typical HTGR and light-water-cooled reactor (LWR). HTGR designs offer significant safety features in the event of an accident (for example, loss of coolant) because the graphite in the core inhibits a rapid temperature rise, and cooling of the core by natural circulation is enhanced even under depressurized conditions. Of longstanding considerable interest is their suitability for cogeneration of heat and power (CHP) with the resulting possibilities for water electrolysis or various thermo-chemical processes to produce hydrogen for process heat applications. Such cogeneration, then, can lead to an overall higher efficiency of the system in terms of total output.

Beck, et al. [7] provide detailed characteristics (Table 1-2) for HTGRs, Dragon (U.K.), Peach Bottom (USA), AVR (Germany), FSV (USA), THTR (Germany), HTTR (Japan), and HTR-10 (China), as well as for AGRs, a MHTGR, and a GT-MHR. Note that the fuel forms vary among graphite hexagonal blocks, graphite pebbles and graphite compacts in hollow rods.

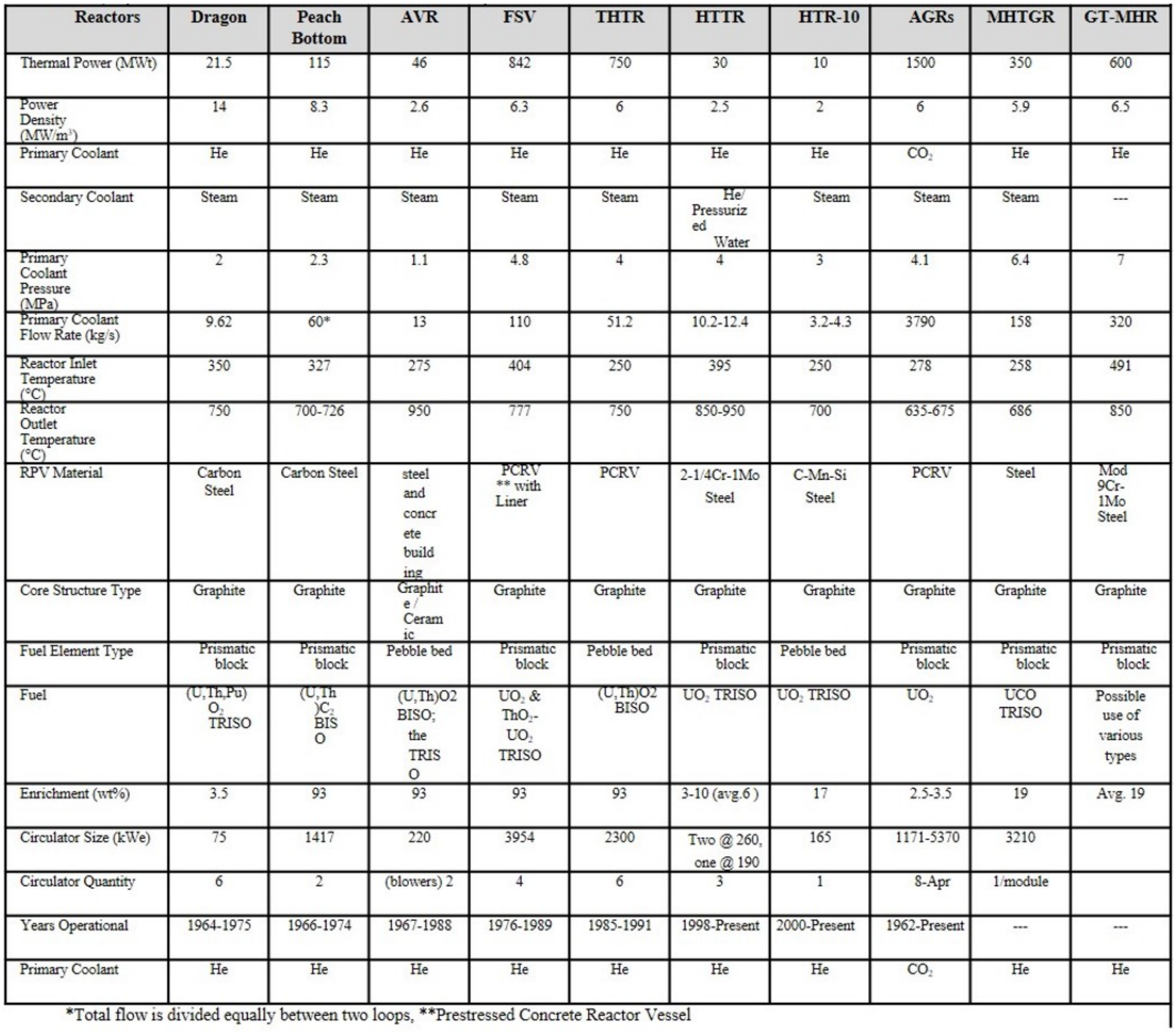
Table 1-1
Comparison of the HTGR and LWR technologies [6]

	High Temperature Gas Cooled Reactor (HTGR)	**Light Water Reactor (LWR)**
Thermodynamic cycle	Gas turbine, Brayton cycle	Steam turbine, Joule cycle
Attainable thermal efficiency	70%	34%
Coolant temperature	He gas: 1,000°C	Steam: 300°C
Operational temperature ranges		
Process heat	30-250°C	–
Power generation	250-600°C	30-300°C
Hydrogen production	600-950°C	–

The Peach Bottom Reactor, for example as described by Chaplin [5], used a "fuel consisted of highly enriched uranium carbide and thorium carbide particles 0.15 mm to 0.40 mm in diameter and coated with pyrolytic carbon to give an overall diameter of 0.25 mm to 0.50 mm. The coated particles were bound together by graphite in a matrix and made into grooved fuel compacts 69.85 mm (2.75 in) in diameter and 38.1 mm (1.5 in) in height. These were stacked on a central graphite spine and encased by a graphite sleeve to make a fuel element 3.658 m (12 ft) in length and 88.9 mm (3.5 in) in diameter. The active length of the fuel in the central region was 2.286 m (7.5 ft). The core was made up of 804 fuel elements arranged in a cylindrical manner to give an active core diameter of 2.792 m (9.2 ft). Graphite guide tubes within the core allowed for the insertion of 36 control rods and 19 shutdown rods as required during various operational states." The fuel design for Peach Bottom was similar to the TRISO (tristructural-isotropic) fuel used in the Dragon reactor, except the TRISO fuel incorporated an additional layer of silicon carbide that was very effective at containing fission products. Whether the fuel particles are imbedded in graphite blocks (prismatic concept) or imbedded in graphite spheres in a packed bed (that is, pebble bed concept), the helium coolant flows through the blocks or packed bed and the heated helium is then passed to a steam generator and then to a turbine, or directly to a gas turbine which eliminates the need for a steam generator. Table 1-3 provides the reactor technical data for the Peach Bottom reactor as an example of one of the first operating HTGRs [5].

Relative to structural metallic materials, Simnad [3] notes that the Peach Bottom steam generator pioneered the use of Incoloy-800 tubes following early failure of tubes made of type 304 stainless steel. The Incoloy-800 tubes appeared to be in excellent condition following shutdown of the reactor after seven years of operation. The Peach Bottom reactor pressure vessel (RPV) was fabricated of carbon steel and cooled by circulating air, typical of these HTGRs with steel vessels. As noted in Table 1-2, two of the reactors incorporated a pre-stressed concrete reactor vessel (PCRV), including Fort St. Vrain, the only commercial HTGR to operate in the U.S. Most PCRVs incorporated a liner of low alloy steel to protect the concrete from the reactor environment [8,9]. It is also notable that the Arbeitsgemeinschaft Versuchsreaktor (AVR) in Germany operated successfully for 20 years and attained the highest temperature (to 1000°C) of any commercial nuclear power reactor. Table 1-2 also includes two experimental HTGRs that are still in operation, the HTTR in Japan, a prismatic design, and the HTR-10 in China, a pebble-bed design. The HTTR has operated at full power since 2001 and has achieved a coolant outlet temperature as high as 950°C. Beck [7] notes (from Fujikawa [10]) that the HTTR is useful to future HTGR development [such as the New Generation Nuclear Plant (NGNP)] because of its current operation and high-temperature capabilities.

Table 1-2
Key specifications for HTGRs [7]



Reactors	Dragon	Peach Bottom	AVR	FSV	THTR	HTTR	HTR-10	AGRs	MHTGR	GT-MHR
Thermal Power (MWt)	21.5	115	46	842	750	30	10	1500	350	600
Power Density (MW/m^3)	14	8.3	2.6	6.3	6	2.5	2	6	5.9	6.5
Primary Coolant	He	He	He	He	He	He	He	CO$_2$	He	He
Secondary Coolant	Steam	Steam	Steam	Steam	Steam	He/ Pressurized Water	Steam	Steam	Steam	---
Primary Coolant Pressure (MPa)	2	2.3	1.1	4.8	4	4	3	4.1	6.4	7
Primary Coolant Flow Rate (kg/s)	9.62	60*	13	110	51.2	10.2-12.4	3.2-4.3	3790	158	320
Reactor Inlet Temperature (°C)	350	327	275	404	250	395	250	278	258	491
Reactor Outlet Temperature (°C)	750	700-726	950	777	750	850-950	700	635-675	686	850
RPV Material	Carbon Steel	Carbon Steel	steel and concrete building	PCRV ** with Liner	PCRV	2-1/4Cr-1Mo Steel	C-Mn-Si Steel	PCRV	Steel	Mod 9Cr-1Mo Steel
Core Structure Type	Graphite	Graphite	Graphite / Ceramic	Graphite	Graphite	Graphite	Graphite	Graphite	Graphite	Graphite
Fuel Element Type	Prismatic block	Prismatic block	Pebble bed	Prismatic block	Pebble bed	Prismatic block	Pebble bed	Prismatic block	Prismatic block	Prismatic block
Fuel	(U,Th,Pu)O$_2$ TRISO	(U,Th)C$_2$ BISO	(U,Th)O2 BISO; the TRISO	UO$_2$ & ThO$_2$-UO$_2$ TRISO	(U,Th)O2 BISO	UO$_2$ TRISO	UO$_2$ TRISO	UO$_2$	UCO TRISO	Possible use of various types
Enrichment (wt%)	3.5	93	93	93	93	3-10 (avg.6)	17	2.5-3.5	19	Avg. 19
Circulator Size (kWe)	75	1417	220	3954	2300	Two @ 260, one @ 190	165	1171-5370	3210	
Circulator Quantity	6	2	(blowers) 2	4	6	3	1	8-Apr	1/module	
Years Operational	1964-1975	1966-1974	1967-1988	1976-1989	1985-1991	1998-Present	2000-Present	1962-Present	---	---
Primary Coolant	He	He	He	He	He	He	He	CO$_2$	He	He

*Total flow is divided equally between two loops, **Prestressed Concrete Reactor Vessel

Table 1-3
Peach Bottom reactor technical data [5]

Parameter	Value
Total Core Height	3.658 m
Active Core Height	2.286 m
Active Core Diameter	2.792 m
Number of Fuel Elements	804
Number of Control Rods	36
Number of Shut Down Rods	19
Reactor Coolant Pressure	2 MPa
Reactor Inlet Temperature	334°C
Core Inlet Temperature	346°C
Reactor Outlet Temperature	734°C
Helium Mass Flow Rate	55.4 kg/s
Thermal Power Output	115.3 MW
Core Power Density	8.24 kW/L
Feedwater Inlet Temperature	218°C
Steam Outlet Pressure	10 MPa
Steam Outlet Temperature	538°C
Steam Mass Flow Rate	45.4 kg/s
Generator Gross Output	46 MW
Plant Net Output	40 MW
Plant Overall Efficiency	34.7%

It is further noted from Fujkawa [10] that a few concerns, such as graphite oxidation and some temperature anomalies, can be considered as lessons for the NGNP. The HTR-10 in China has produced full power since 2003, with an output of 10MWth compared with the 30MWth of the HTTR and has a much lower coolant outlet temperature of about 700°C. Although the HTTR uses a prismatic block fuel element design and the HTR-10 uses a pebble bed design, both use TRISO-coated UO_2 fuel particles. Table 1-2 also shows that the RPV for HTTR was fabricated with 2-1/4-1Mo steel, because it operates at about 400°C (higher than allowed for typical LWRs), while that of the HTR-10 was fabricated with a C-Mn-Si low-alloy steel for lower temperature operation. Although neither has yet been constructed, the Modular High-Temperature Gas-Cooled Reactor (MHTGR) and the Gas-Turbine-Modular Helium Reactor (GT-MHR) are included because they have been parts of major design programs in the USA, such as the Department of Energy New Production Reactor (NPR) program (with eight 350 MWth reactor modules) and a program partially sponsored by the National Nuclear Security Administration (NNSA) [7]. The

South Africa Pebble-Bed Modular Reactor (PBMR) and the General Atomics-designed GT-MHR are both MHTGR type designs. The GT-MHR uses a prismatic core design and a gas turbine to generate power rather than the more conventional steam generator. Notably, the GT-MHR is designed with modified 9Cr-1Mo steel for the RPV.

Thus, there has been a significant amount of useful operating experience with GCRs and HTGRs, both for experimental research and commercial power applications. McDowell, et al. [11] concluded that the full HTGR potential for electrical generation has not yet been demonstrated in practice, but also note that both pebble-bed and prismatic block reactor cores appear to be practical. Both the Beck and McDowell reports provide discussion relative to "lessons learned" from operation of the various gas-cooled reactors and those relevant to the subject of this section. Although more recent VHTR concepts have proposed outlet temperatures of 1000°C or more, there are no significant conceptual differences distinguishing VHTRs from HTGRs, and so the term VHTR will be used to represent both VHTRs and HTGRs in the remainder of this document.

While both the VHTR and the GFR are helium-cooled systems, the VHTR has a thermal spectrum and the GFR has a fast-neutron spectrum. Thus, while the VHTR incorporates a large amount of graphite as a moderator, the fast-spectrum concept of the GFR limits the use of graphite because it is an excellent moderator and, thus, it must rely on selection and qualification of other materials for the core and internal structural components capable of sustaining operation at high temperatures and high radiation doses. Moreover, a major driving force for the GFR is the closed fuel cycle resulting in efficient use of uranium resources (self-generation of plutonium in the core) and effective management of actinides leading to minimization of long-lived radioactive waste isotopes. Various GFR designs have been proposed since the 1960s in both the USA (for example, General Atomics), Europe (for example, Germany consortium, European Gas Breeder Reactor Association) [12] U.K., and Japan, and include both helium-cooled and supercritical CO_2-cooled (S-CO_2) systems. Although the current proposed GFRs are designed to operate at a somewhat lower temperature (to ~850°C) than some of the proposed VHTRs (to ~950-1000°C), very high temperatures (to ~1600°C) are anticipated under accident conditions in the GFR and irradiation doses to some internal structures are significantly higher in the GFR [13]. As noted by Corwin, et al. [14], many of the materials issues outside of the GFR core region are similar to those for the NGNP (that is, a VHTR); thus, most of the materials discussion for the GFR will focus on the core region. In contrast with the case for the VHTR, no GFRs have ever been constructed. Since there is no operating experience for GFRs, no review of the various previous designs is presented here.

Three specific prior reports have been published that provide results of detailed reviews of the gas-cooled NGNP, including a report on plant phenomena identification and ranking tables (PIRT) [15], a gap analysis report [16], and a lessons-learned report [7]. These reports are used to provide initial identification of the critical components, materials, and unresolved issues for the VHTR system, which will then be updated as appropriate with more recent literature.

1.2 Overview of Current VHTR and GFR Designs

There are many VHTR concepts and designs, both for power production and for co-generation of power and process heat applications. As noted by Buckthorpe [13], the nuclear energy systems identified by the GIF offer significant improvements over current Gen II and Gen III systems with regard to economics; safety and reliability; proliferation resistance and physical protection/ and sustainability. For the two gas-cooled (helium) reactors identified by the GIF, the VHTR and GFR reference designs are identified with outlet temperature/power output of 1000°C/250-300 MWe and 850°C/1200 MWe, respectively. The VHTR has a thermal neutron spectrum and an open fuel cycle while the GFR has a fast spectrum and a closed cycle. Buckthorpe [13] also notes that the GIF members are collaborating on the research and development needed to develop these systems beyond that currently being undertaken by industry. A schematic diagram of a co-generation VHTR [17] is shown in Figure 1-1 with the reactor primary circuit (either the prismatic block or pebble bed type core) connected to a steam reformer/steam generator to deliver process heat, in this case to a hydrogen production plant (the same figure is shown in the more recent Buckthorpe paper [13]).

Various thermochemical processes [for example, sulfur-iodine (S-I) process] or combined thermochemical and electrolysis (such as a hybrid sulfur process), high temperature steam electrolysis (HTSE), or from heat, water and natural gas by applying the steam reformer technology. As noted in a more recent report (2014) from the GIF [18] the main focus in the near future will be on a VHTR with core outlet temperature of 700-950°C, with further R&D on materials and fuels that should enable higher temperature up to and above 1000°C. The following statements from the GIF update document are worth noting here:

> "*Further demonstrations of the safety performance for both the prismatic and pebble bed concepts, at HTTR and HTR-10, emphasize the benefit of the strong negative temperature coefficient of reactivity, the high heat capacity of the graphite core, the large temperature increase margin, and the robustness of TRISO fuel in producing a reactor concept that does not need off-site power to survive multiple failures or severe natural events as occurred at the Fukushima Daiichi nuclear station.*"

The GIF R&D objectives for the VHTR fuels and materials include, 1) qualification of TRISO fuel; 2) ZrC coatings for TRISO fuel; 3) carbon/carbon (C/C) and SiC/SiC composite components; 4) pressure vessel materials; and 5) heat utilization systems materials. The last item includes information that ***internal core structures and cooling systems, such as intermediate heat exchangers, hot gas ducts, process components and isolation valves, all that are in contact with hot helium, can use current metallic materials up to a core outlet temperature of about 700-800°C.***

The Corwin, et al. report [17] identifies a number of candidate materials/alloys for the various high-temperature metallic components of the NGNP: Alloys 316FR, 800H, 617, 230, 347, and XR, as well as CC composites. Because the reactor of the VHTR contains a large volume of graphite, properties of the graphite (including irradiation effects) are critical and the previously characterized graphites used in the past are no longer available, requiring extensive characterization of currently available graphite(s). Material selection issues for VHTR are discussed in more detail in Section 2.

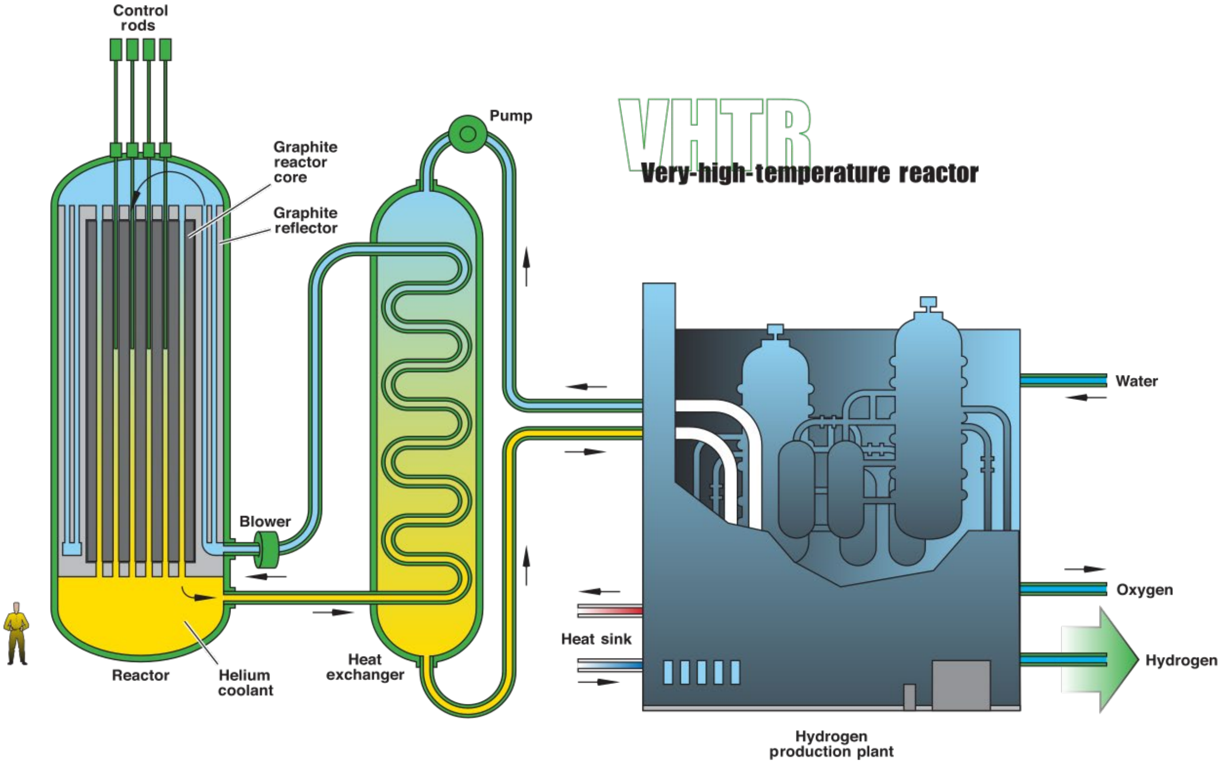


Figure 1-1
Schematic diagram of a typical VHTR [85]

Figure 1-2 from Corwin, et al. [14] shows a schematic diagram of a GFR reference design (2005) that utilizes a direct Brayton cycle turbine for electricity production and, as with the VHTR, provides process heat for thermochemical production of hydrogen (the same figure is shown in the more recent Buckthorpe paper [13]). There are two well-known alternate options, with the first being a helium-cooled system utilizing an indirect Brayton cycle at 850°C and 7 MPa for power conversion, but with a secondary system that uses supercritical-CO_2 (S-C O_2) at reduced temperature of 550°C and 20 MPa. The resulting lower outlet temperatures in the primary circuit (~600-650°C) can reduce fuel, fuel matrix, and material requirements, but still maintain high thermal efficiency (~42%). The second option is a direct Brayton cycle system using S-CO_2 cooling at 20 MPa and with an outlet temperature of 550°C [14]. This option further reduces temperature in the primary circuit while maintaining efficiency and reducing both fuel and materials development costs relative to the reference design [14]. However, a more recent revision of the GIF reference design for GFR is for a 2400 MWth reactor which is capable of break-even breeding [18]. Thus, for the GIF, there really are two reference GFR designs, the larger 2400 MWth concept is to allow for power production and fuel breeding, while the smaller 600 MWth concept is considered an option for a small modular reactor (SMR) that does not need to be a breakeven breeder. That report also notes that the direct power conversion cycle chosen as the original GIF roadmap reference design is changed to favoring an indirect cycle with helium on the primary circuit, and Brayton cycle on the secondary circuit and a steam cycle on the tertiary circuit. The indirect cycle is favored by some, including the GIF, because of its lower technological risk and higher flexibility with respect to the choice of working fluid [18]. In that regard, however, the GIF [12] notes that the development of robust high temperature, high power density refractory fuels and core structural materials are the greatest challenge facing the GFR.

As noted previously, most of the GFR balance of plant will likely utilize materials selected and qualified for the VHTR, although the primary materials challenge for the GFR will be qualification of materials for the core and reactor internals structures due to high temperatures and very high neutron exposures. Also, without the graphite content as in the VHTR, the core can heat up very rapidly with loss of forced cooling due to low thermal inertia. Since the power density is high, the conduction cool-down typical in a VHTR will not work for removal of the decay heat which is critical in accident conditions. The GIF update report R&D objectives include the need for an experimental fast spectrum reactor, but also notes the need for fuel development and out-of-pile experimental facilities for qualification of the main systems [18]. Within the latter item are listed the need for innovative materials and various ceramic materials, and development and qualification of components, including specific blowers and turbomachines, thermal barriers, valves and check-valves, and instrumentation.

Weaver [19] provided specific requirements for the GFR fuel matrix and structural materials as discussed in Section 3 of this report. He also stated that only ceramics can meet those reference values and provided some likely candidates discussed in Section 3. For out of core applications, metallic alloys T122, 800H, and certain oxide dispersion strengthened (ODS) alloys were noted. However, Buckthorpe [13] notes that considerations for the GFR core material would include Cr-Ni based austenitic steels, high-nickel alloy steels, some ferritic-martensitic (F-M) and ODS steels, a few vanadium alloys, and ceramic materials. For the RPV of the GIF reference design, Vasile [12] lists the need for very long term (60 years) creep and creep-rupture properties operating in the 400-550°C range and up to 100 dpa. Candidate RPV materials are also listed and include F-M steels and stainless steel. Potential high temperature bolting grades are also listed.

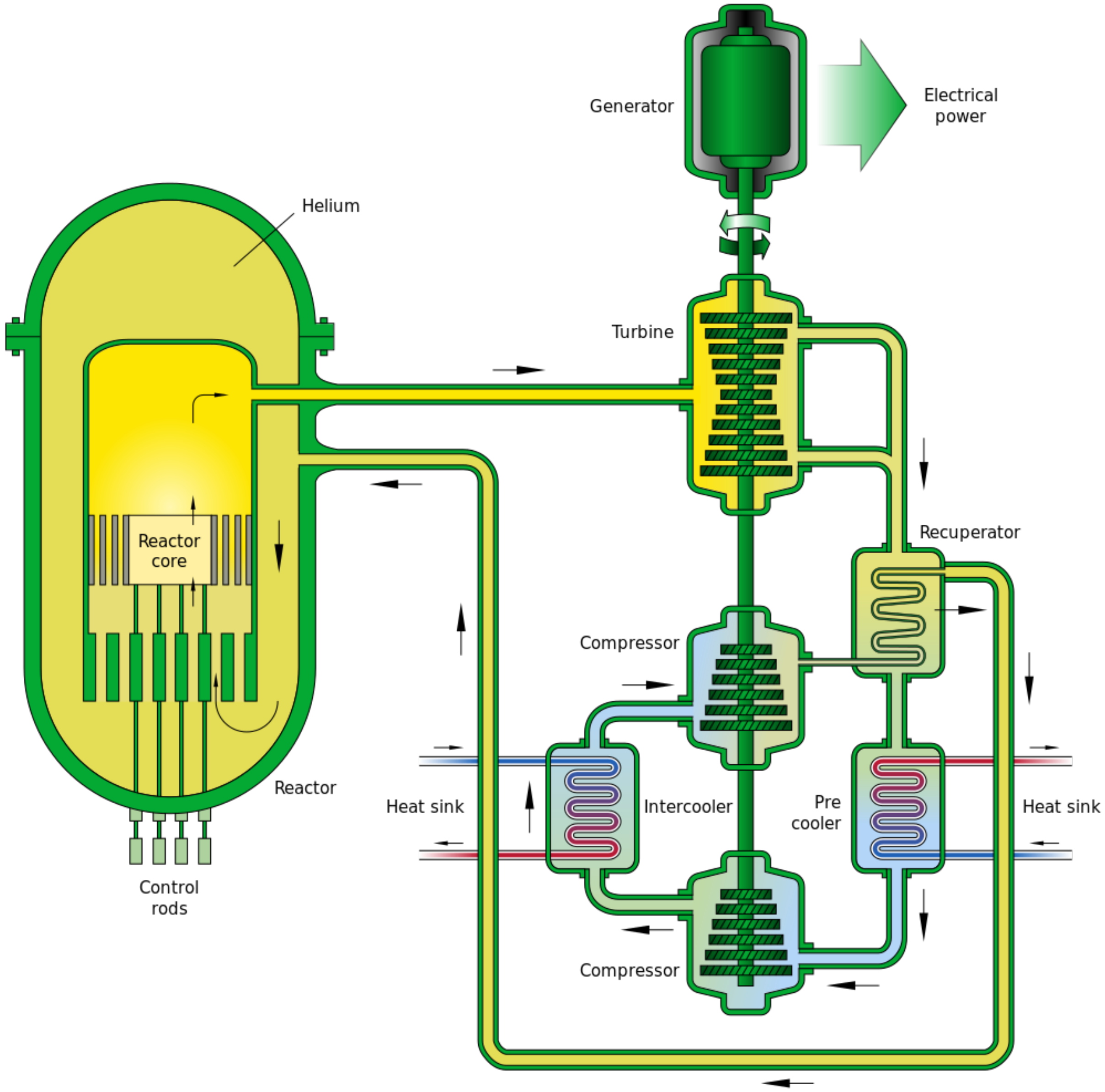


Figure 1-2
Schematic diagram of a GFR [86]

It is noteworthy that the European Sustainable Nuclear Industrial Initiative (ESNII) includes a project to develop a small GFR (ALLEGRO) as a technology demonstration reactor, with objectives to demonstrate the viability to qualify specific GFR technologies [20]. The initial ALLEGRO reactor design with a mixed oxide (MOX) core is for $75MW_{th}$ power and with a helium output temperature of 530°C at 7MPa, but without a power conversion system. Figure 1-3 [20] shows a schematic view of the ALLEGRO reactor. In the long term, it is foreseen that the initial MOX core would be replaced with a ceramic core which would allow for a helium outlet temperature of 850°C. Material selection issues for the GFR are discussed in more detail in Section 3.

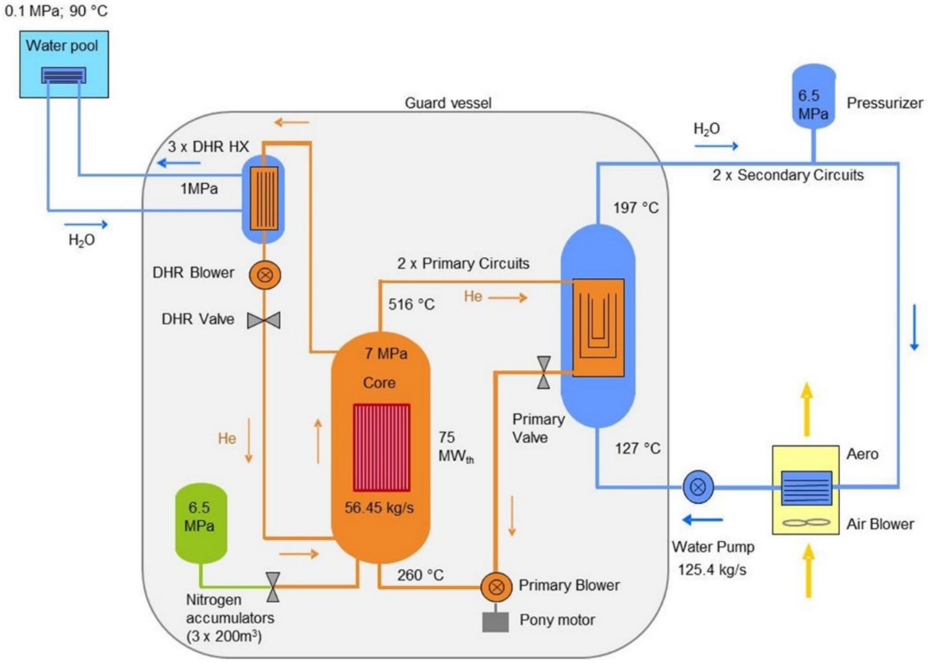


Figure 1-3
Schematic view of the ALLEGRO demonstration gas-cooled fast reactor [20]

1.3 Overview of ASME Code High Temperature Design Rules

The ASME Boiler and Pressure Vessel Code (ASME Code) has major sections that are devoted to the design, construction, and inspection of nuclear reactor components focusing primarily on pressure retaining systems. The following discussion provides the scope of the key sections in the ASME Code as it relates to nuclear reactors and components, which focus on the high temperature reactors discussed in this report.

Section II Materials covers the general requirements for qualified materials in all boiler and pressure vessels covered by the ASME Code. Note that the material designations are consistent with those used by the American Society for Testing and Materials (ASTM), except that an “S” is added to designate that it as an ASME Code approved material; for example, the specification for forging material SA-508/SA-508M is the same as specified as A508/A508M by ASTM. The “M” refers to the metric SI version rather than the U.S. customary version. There are some non-ASTM materials that have been endorsed for Code application, and their originating organizations are acknowledged and reproduced in Section II. There are four parts in Section II: Part A covers the specifications for ferrous materials; Part B covers the specifications for nonferrous materials; Part C covers specifications for weld rods, electrodes, and filler materials following American Welding Society (AWS) specifications; and Part D provides property tables for establishing component thicknesses and approved operating component temperatures as well as design stress intensities. In some materials applications that may exceed the operating temperatures as specified, supplemental test data (including thermal aging, creep and creep-fatigue interaction results at the higher temperatures) can be submitted to the ASME Code body as a proposed Code Case to modify the temperature range of application and/or the associated properties. If the Code Case is approved, the material specification can be adjusted, but the U.S.

Nuclear Regulatory Commission (NRC) must accept the Code Case for use or apply restrictions to allow its use of the material in a constructed reactor. Generally, approved Code Cases (even if not fully endorsed by the NRC) are later incorporated into the Code itself, but the process can take several years.

Section III Rules for Construction of Nuclear Facility Components has requirements for the design and construction of nuclear components as indicated in its name. Section III is broken into divisions where Division 1 is most commonly used for light water reactor components. Within Division 1 through the 2015 Code there were seven subsections: NB (Class 1 components), NC (Class 2 components), ND (Class 3 components), NE (Class MC components), NF (supports), NG (Core support structures), and NH (Class 1 components in elevated temperature service). Note that subsection NH subsequently has been moved to Division 5 (High Temperature Reactors), Subsection HB, subpart B. Subsection NH was originally developed as Code Case N-499 and then added to the Code in Section III. Division 5 was been established for high temperature reactor components, and the ASME Code 2017 Division 5 is broken down into the following subsections (although the slightly modified 2015 NH subsection is included again even though it is part of subsection HB, subpart B):

- Subsection HA—General Requirements
  - Subpart A—Metallic Materials
  - Subpart B—Graphite Materials
  - Subpart C—Composite Materials
- Subsection HB—Class A Metallic Pressure Boundary Components
  - Subpart A—Low Temperature Service
  - Subpart B—Elevated Temperature Service
- Subsection HC—Class B Metallic Pressure Boundary Components
  - Subpart A—Low Temperature Service
  - Subpart B—Elevated Temperature Service
- Subsection HF—Class A and B Metallic Supports
  - Subpart A—Low Temperature Service
- Subsection HG—Class A Metallic Core Support Structures
  - Subpart A—Low Temperature Service
  - Subpart B—Elevated Temperature Service
- Subsection HH—Class A Nonmetallic Core Support Structures
  - Subpart A—Graphite Materials
  - Subpart B—Composite Materials

The interest in this report is primarily focused on Subsection HA, Subpart A, Metallic Materials and Subsection HB, Subpart B, Elevated Temperature Service. Note that special groups have been organized within the ASME Code to continue to support the development needs of high temperature reactors in areas relevant to component materials: Subgroup on High Temperature Reactors, Subgroup on Elevated Temperature Design, Subgroup on Graphite Core Components, and Special Working Group, High Temperature Gas Cooled Reactors.

Subsection HA, Subpart A contains the general requirements associated with metallic components used in the construction of high temperature reactors and their supporting systems. The metallic materials requirements are for new construction and include consideration of mechanical and thermal stresses due to cyclic operation and high temperature creep. Changes in properties that may occur in service as a result of radiation effects or other degradation processes are not covered, although it is important to realize that radiation effects and thermal effects can alter the properties of materials over time. Changes in properties of materials subjected to neutron radiation may be checked periodically by means of material surveillance programs, such as those currently in use in the light water reactor (LWR) nuclear industry. As an example, SA-533 plate and SA-508 forging materials have been used for over 50 years in the LWR nuclear industry and are Code approved for Section III, NB Class 1 components up to a temperature of 371°C (700°F). Other high temperature alloys, as discussed later in this report, may have NB Class 1 similar maximum temperatures, but the Division 5, HB, Subpart B temperatures can be much higher and generally with restrictions on time in service. Note again that the temperature range may also be expanded through Code Cases. Section 2.3 (Materials for Core Internals) of this report discusses additional details of maximum allowable temperatures in Division 5.

In this report, Sections 2.2 through 2.5 for VHTRs and Sections 3.2 through 3.5 for GFRs provide a more complete view of the candidate metallic materials that are being considered for use in the high temperature reactors. There are specific issues for many of these materials primarily related to the operating temperatures of the components and the expected radiation dose. In many cases the current SA-508/SA-533 materials are being considered for many of the components, but the viability of their application at these higher temperatures need to be addressed.

Section V Nondestructive Examination has the requirements for validating and performing nondestructive examination (NDE) as specified in Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components or elsewhere in the Code. The NDE methods are designed to detect surface and internal imperfections in the components being examined. Typical examinations include radiographic, ultrasonic, liquid penetrant, eddy current, magnetic particle, acoustic emission, and visual.

Section IX Welding, Brazing, and Fusing Qualifications provides the approved and documented procedure qualifications for the joining operations using welding, brazing, and other material fusing methods. Relative to Class 1 metallic components, specific requirements for welders and welding operators, as well as brazing operations are delineated to assure high quality components are produced under the ASME Code requirements with the materials approved in Section II.

Section XI Rules for Inservice Inspection of Nuclear Power Plant Components provides requirements for inservice inspection and testing of LWR nuclear power plant systems and components. Although not directly used for high temperature reactors, the general methodology used in Section XI can be applied to future inspection needs and requirements for high temperature reactors.

2

VHTR DESIGN

2.1 VHTR Component Descriptions and Operating Conditions

For this report, the primary VHTR designs for discussion include those currently described by AREVA/NGNP Alliance (prismatic core) and X-energy (pebble bed core). Other similar designs exist worldwide, but these two provide good examples of the two different design concepts. In both cases the core outlet temperature is 750°C, while the thermal/electrical outputs per module are 625 MWth/272 MWe for the Areva/NGNP Alliance case and 200 MWth/75 MWe for the X-energy case. Higher temperatures are not considered herein as such designs will require additional materials development and are not thought to be sufficiently mature at this point in time.

2.1.1 AREVA/NGNP Alliance Reactor

The NGNP Industry Alliance has chosen the AREVA steam cycle high temperature gas-cooled reactor (SC-HTGR) as the reactor design concept of choice for high temperature process steam for industrial applications [21]. Ten commercial organizations are currently members of the NGNP Industry Alliance and all ten companies are affiliates of the Electric Power Research Institute (EPRI) [22]. Three major components comprise their HTGR nuclear heat supply system (NHSS); 1) a helium cooled nuclear reactor, 2) a heat transport system (steam generator), and 3) a cross vessel for helium flow from the reactor to the heat transport system [23], as shown in Figure 2-1 (NGNP Alliance Website—[22]). The NHSS design is of a modular design, with module ratings from 200 MWt to 625 MWt, reactor coolant outlet temperatures from 700 to 850°C, and heat transport systems that can provide steam and/or high temperature fluids. All three components are enclosed in metallic pressure vessels. Heat from the heat transport system is fed to an energy conversion system (for example, a steam turbine generator) which can be interfaced with some industrial process (for example, H_2 production) and/or the electrical grid (not shown in Figure 2-1). Reactor temperature is maintained during shut down for maintenance or refueling by the shutdown cooling system. However, even if all electrical power is lost and the shutdown cooling system and other cooling paths cannot operate, heat transfer through the reactor building structure to the ground is sufficient to maintain reactor temperatures within acceptable limits.

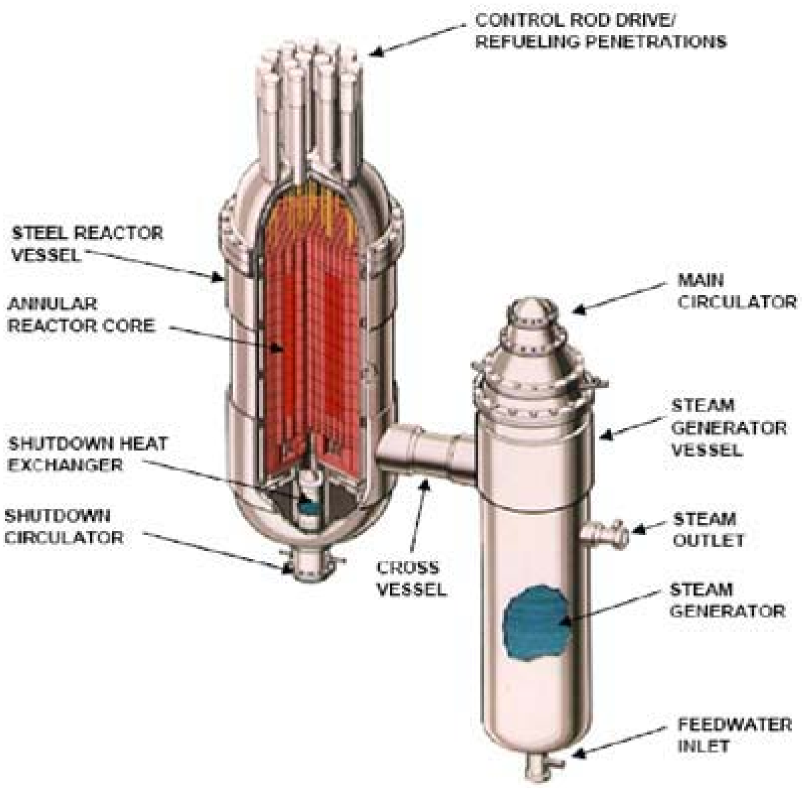


Figure 2-1
Schematic overview of NGNP Alliance Reactor [23]

The reactor uses the prismatic design, with TRISO-coated fuel particles (that is, a uranium dioxide or oxycarbide kernel with a layer of silicon carbide and a final layer of pyrolytic carbon). The particles are formed into fuel compacts and inserted into graphite fuel elements similar to that described previously for the Peach Bottom Reactor. Thus, many dozens of fuel elements make up the large graphite core which resides in the RPV. The graphite can absorb a very large quantity of heat so that it takes many hours or even days to reach peak accident temperatures [23]. Figure 2-2 (Areva Info Kit) shows a flow diagram with temperatures indicated for a typical co-generation plant configuration [21] with, in this case, a helium outlet temperature of 750°C shown for the primary loop.

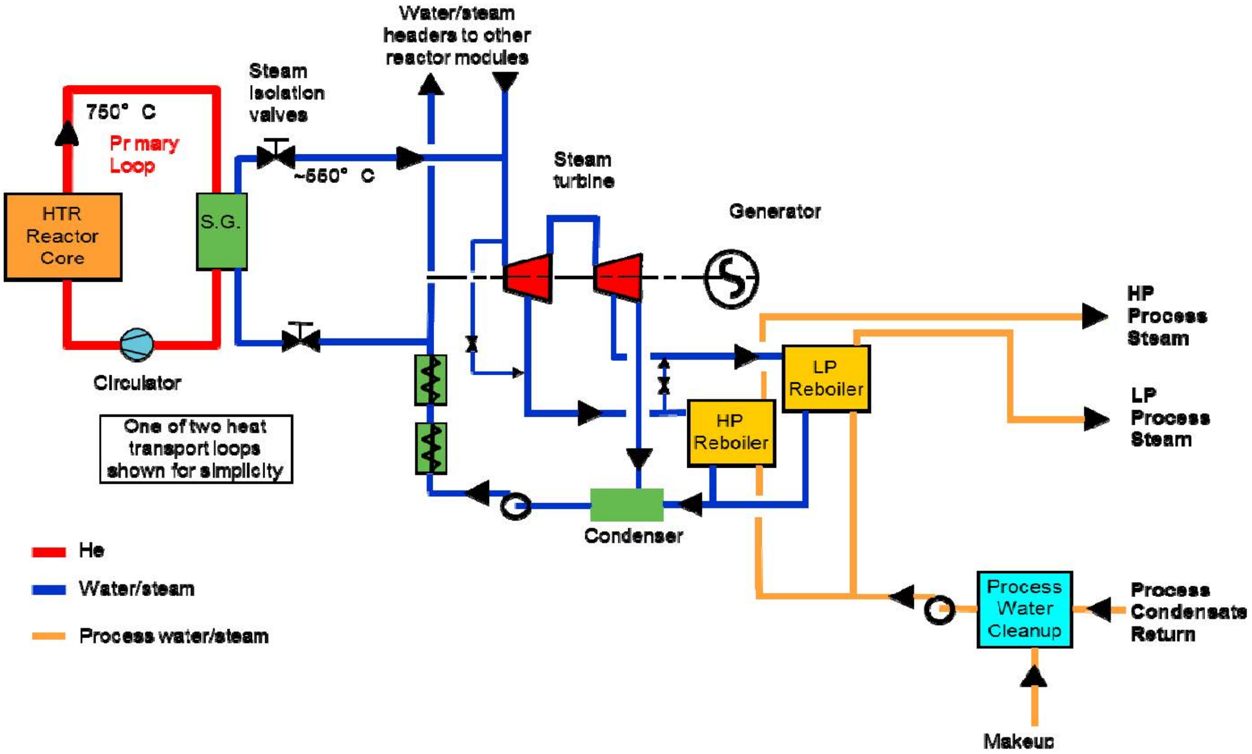


Figure 2-2
Typical Co-generation plant configuration [21]

More detailed operating parameters for the 625 MWt AREVA design [21] are provided in Table 2-1 (Areva Info Kit). As shown, the reactor inlet temperature at 325°C is half that of the outlet temperature at 750°C, resulting in a nominal efficiency of 50%. The table also shows the selection of SA-508/533 steels for the vessel material. Although they do not state the specific grades, SA-508 and SA-533 are both low alloy steels (forging and plate, respectively) used for light-water reactor (LWR) pressure vessels. Those steels, usually the forging grade, are specified for the RPV in other advanced high temperature reactors that specify cooling of the RPV for operation at about 300°C as typical of LWRs. The AREVA design specifies that the reactor vessel and metallic internals would be cooled with core inlet gas, which is listed at 325°C [21] (Table 2-1). Figure 2-3 (Areva Info Kit) shows a schematic diagram of the nuclear reactor indicating the directions of the helium gas flow and various sub-components of that part of the system [21]. The reactor system is based on the previous AREVA ANTARES design [24].

Table 2-1
Nominal operating parameters for AREVA HTGR [21]

Fuel Type	TRISO Coated Particle
Core geometry	102 column annular 10 blocks high
Reactor power	625 MWt
Reactor outlet temperature	750°C (1382°F)
Reactor inlet temperature	325°C (617°F)
Primary coolant pressure	6 MPa (870 psia)
Vessel material	SA 508/533
Number of loops	2
Steam generator power	315 MWt (each)
Main circulator power	4 MWe (each)
Main steam temperature	566°C (1050°F)
Main steam pressure	16.7 MPa (2422 psia)

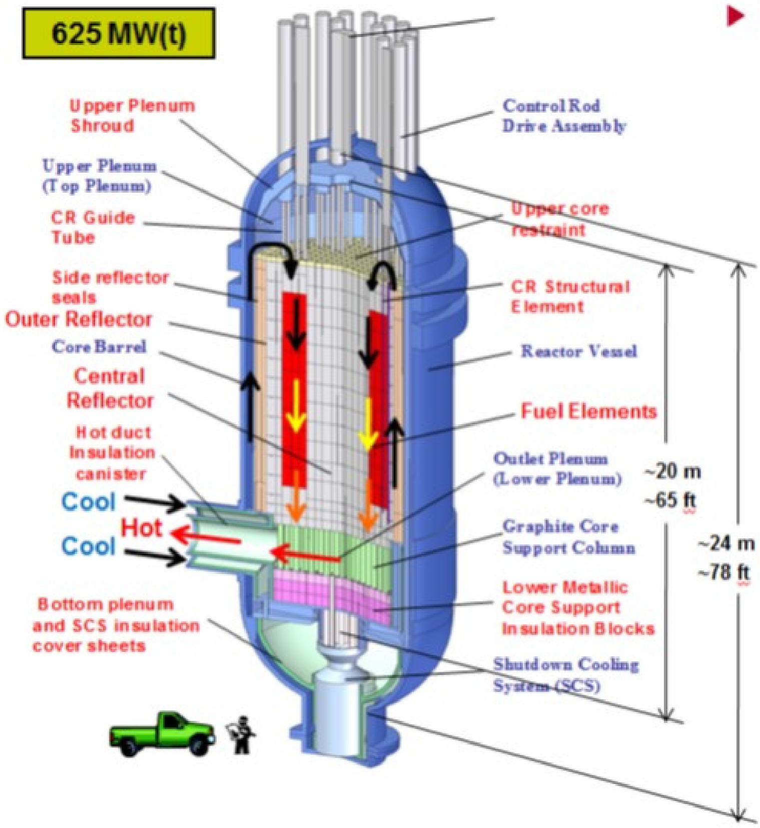
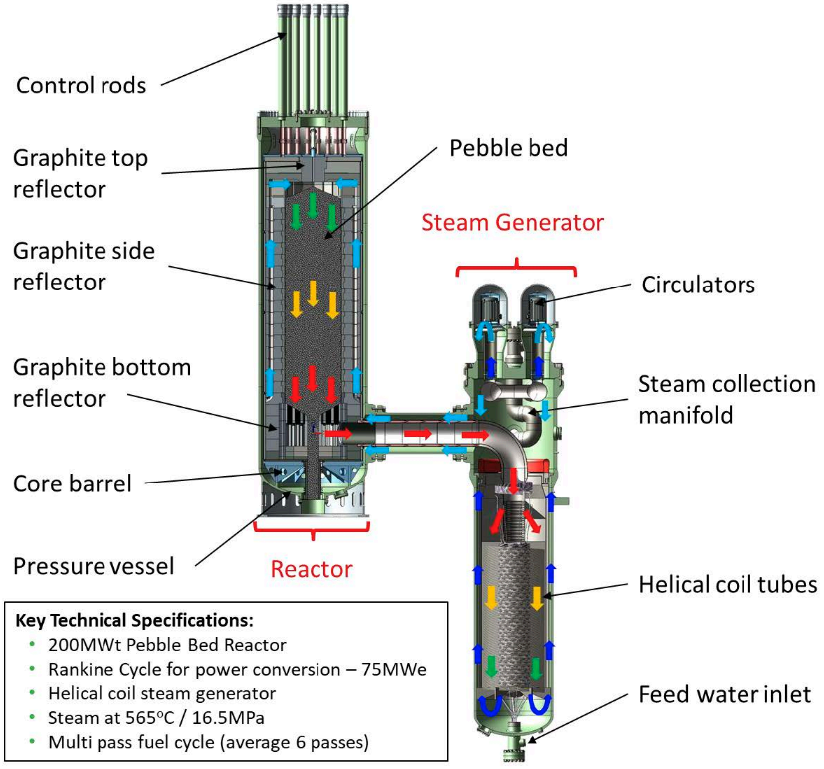


Figure 2-3
Schematic diagram depicting sub components of the AREVA 625 MWt reactor [21]

2.1.2 X-energy Xe-100 Reactor

The Xe-100 is a 200MWth (~75MWe) helium cooled nuclear reactor with pebble-bed design that features a low enrichment uranium (LEU) fuel cycle [25, 26]. The standard Xe-100 "four-pack" plant generates approximately 300MWe and will fit on as few as 13 acres. The Xe-100 is a CO_2-free nuclear thermal power source that can be utilized for power generation, process heat applications, and water desalination, while all the components for the Xe-100 will be road-transportable and will be installed, rather than constructed, at the project site to streamline construction [25]. The fuel is composed of uranium oxy-carbide (UCO) kernels (15.5% LEU) encased in carbon and ceramic layers (TRISO particle) and embedded in 6cm graphite pebbles. There are about 18,000 TRISO coated particles in each graphite pebble, and about 220,000 pebbles in the core of each Xe-100 reactor. The inherent safety benefits of the pebble fuel are stated by X-energy as, 1) carbon and ceramic layers prevent release of radioactivity; 2) TRISO particles maintain individual integrity independent of their particles; 3) graphite surrounding the TRISO particles moderates the reactor; and 4) UCO TRISO coated particles act like pressure vessels to retain fission products. The reactor has a helium outlet temperature of 750°C and pressure of 6MPa, with helium circulators. The steam generator is a helical coil design and operates at a steam pressure of 16.5MPa and steam temperature of 565°C. It is of relatively small size and modular construction, and includes an on-line fueling capability which allows for continuous operation, giving it advantage over prismatic designs that have an 18 to 20- month fuel cycle [26]. The core includes a core barrel and helium pressure boundary, contained in a pressure vessel of 17.5m x 4.85m diameter. The heat source is based on pebble bed technology which X-energy states has a proven meltdown proof core [25], and the heat transfer is via a proven helical coil steam generator. The Xe-100 reactor and steam generator are shown in Figure 2-4 [26] with a table indicating key technical specifications. The Xe-100 development involves a strategic partnership between X-energy and multiple partners, including nuclear industry organizations and four U.S. Department of Energy National Laboratories.



Key Technical Specifications:
- 200MWt Pebble Bed Reactor
- Rankine Cycle for power conversion – 75MWe
- Helical coil steam generator
- Steam at 565°C / 16.5MPa
- Multi pass fuel cycle (average 6 passes)

Figure 2-4
Schematic diagram of X energy Xe-100 reactor and steam generator [26]

The process flow diagram for the Xe-100 is shown in Figure 2-5 (X-energy at NRC) and indicates the various thermal values in the nuclear island components as well as in the conventional island for production of electricity.

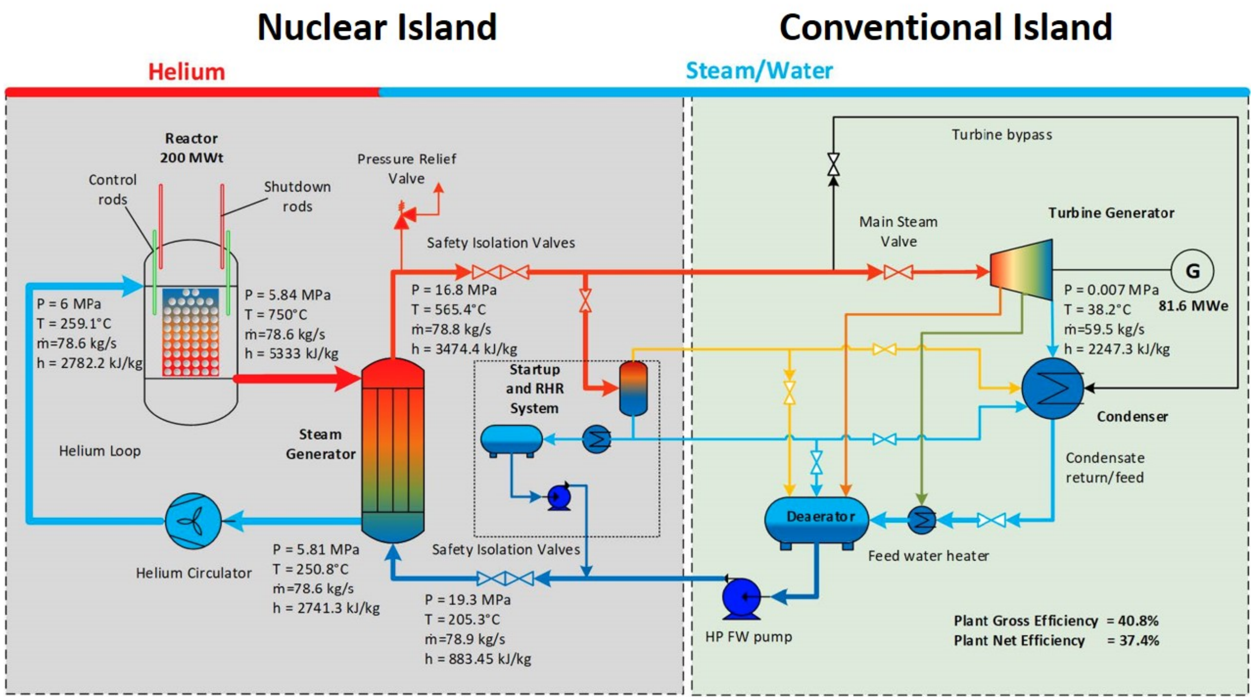


Figure 2-5
Xe-100 energy balance process flow diagram [26]

Table 2-2 [26] provides the material selections for the various portions of the RPV; the choice of SA-508 forgings (similar to those for LWR RPVs) implies that the RPV will be cooled to about 300°C, although that is not explicitly stated in the X-energy document cited [26]. The RPV is 17.5m x 4.85m diameter and is claimed to be road transportable [25]. All the materials selections shown in Table 2-2 are within existing ASME specifications. The reactor's graphite reflector will be selected to comply with ASME Section III, Division 5, and X-energy states an irradiation program is currently underway at ORNL.

Table 2-2
Material selections for Xe-100 RPV components [26]

Component	Material
Bottom Head, Manway Shell, Crossover Vessel	SA-508, Grade 3, Class 1
Vessel Flange, Top Head CRDM Housings	SA-508, Grade 3, Class 2
Top Head Fasteners	SB-637, Alloy 718
Center Manway Fasteners Bottom Manway Fasteners	SA-540, B24, Class 1

2.2 Components and Sub-Components Materials Issues

As discussed in Sections 2.1.1 and 2.1.2, the AREVA and X-energy VHTRs, respectively, are designed with the same helium outlet temperature, 750°C. Previous assessments for Gen IV VHTRs focused on reactor designs with outlet temperatures as high as 1000°C [14]. From the materials perspective, the outlet temperature of 750°C will certainly reduce the challenge for identifying potential materials, especially for the metallic reactor internals and steam generators. As stated recently in the GIF Annual Report [27] (2017) relative to the HTR-PM in China: *"The reactor outlet temperature will be 750°C, which is well within the limits of the current state-of-the-art for materials and components, yet suitable for the generation of high-quality steam of 566°C."* Additionally, the apparent decision by the designers of both reactors to cool the RPV and cross vessel mitigates the need for the use of higher alloy steels, such as 9Cr-1MoV, for those components. Although many of the components and sub-components for each reactor are identified in the figures in Sections 2.1.1 and 2.1.2, the actual operating temperatures, irradiation doses, and so on, are not specifically available for this report. Except for the reactor core differences associated with the prismatic (AREVA) and pebble-bed (X-energy) designs, it is assumed in this report that the components and sub-components for both reactors are similar in that they have a reactor, a cross vessel, and a steam generator. Previous comprehensive assessments (for example, Corwin, et al. [14,17], Ball and Fisher [15], Ball, et al. [16]) of Gen IV VHTRs for the U.S. Department of Energy and for the U.S. NRC identified potential materials for the various components and performed plant phenomena and identification ranking exercises. Those studies, as well as others referenced as appropriate, are used here to inform this review. The major categories of interest are the graphite for core support and reflectors, the materials (nonmetallic or metallic) for control rods, core restraints, liners, and insulation, metallic alloys for internal structures [(core barrel, inside shroud, core support floor, upper core restraint, shutdown cooling system (SCS) shell and tubes], the RPV and cross vessel, the steam generator and heat exchangers, recuperator, piping and valve materials.

The major classes of materials and potential component applications identified in the NGNP PIRT exercise for high temperature materials performed for the Nuclear Regulatory Agency (NRC) are shown in Table 2-3 [28].

With reference to the PIRT exercise for the NRC [15], Longoni, et al. [11] defined classes of high temperature materials by maximum temperature:

- Ferritic steels (450°C)
- Ferritic/martensitic steels (650°C)
- Austenitic stainless steels (800°C)
- Nickel-based superalloys (1050°C)
- Inter-metallics (1250°C)
- Refractory Alloys (1400°C)
- Ceramics/Composites (1600°C)

Such a classification provides a very useful guide for initial selection of materials, with the selection then being dependent on other factors of operating conditions, for example, coolant environment, radiation exposure, and so on, as well as recognition of potential advances in development of advanced materials within the categories.

Table 2-3
Major classes of materials expected to be used in the NGNP [28]

Material Type	Examples of Materials	Potential Component Application
Low-alloy steel*	SA-508 steel SA-533B steel *2-1/4 Cr-1-MoV steel* *9 Cr-1-MoV steel*	Reactor pressure vessel *and piping*
Stainless steel	304 stainless steel 316 stainless steel 347 stainless steel	Core barrel Ducting
High alloys	Inconel 617 Haynes 230 Incoloy 800H Hastelloy X and XR Inconel 740	Core barrel Intermediate heat exchanger Piping Bolting Control rods Turbomachinery
Nanostructured and oxide dispersion strengthened alloys	MA 956 PM 2000	Support structures
Nonmetallic composites	Carbon-carbon (C-C) SiC-SiC	Control rods Core restraints Liners for hot ducts and insulation
Nonmetallic materials (ceramics)	Alumina Silica Kaowool	Insulation
Graphite**	SGL NB-10 NB-12	Fuel, core support, reflectors

* The 9Cr-1MoV steel listed primarily for the higher outlet temperature of 950°C; the lower Cr steel is listed as an alternative to the SA-508 and SA-533 if there are concerns regarding adequate cooling of the RPV.

** Newer graphite grades are discussed in Section 2.3.

2.3 Materials for Core Intervals

Table 2-4, taken from Corwin, et al. [17], shows the components and conditions for selection of the NGNP metallic materials, somewhat generically based on a number of VHTR concepts in 2005. Table 2-3 is based on the much higher helium outlet temperatures of 950°C; thus, as stated earlier, the substantially lower outlet temperature of 750°C for the AREVA reactor and the X-energy reactor will clearly reduce the challenge for materials selection. Also, there are steam generators in the current designs, not IHXs or traditional piping, and the steam outlet temperature is 565°C for both systems.

Table 2-4
Conditions affecting materials selection metallic NGNP components with 950°C outlet [17]

Condition	SCS Tube	Core Barrel	Core Support Floor	SCS Shell	Inside Shroud	Upper Core Restraint
Normal	*600°C*	*600°C*	*600°C*	*600°C*	*600°C*	*600°C*
Temperature*						
Maximum	*600°C*	*700°C*	*700°C*	*1200°C*	*1200°C*	*1200°C*
Temperature*						
Loading	Thermal	Core	Core	Own	Own	Own
	Stress	Weight	Weight	Weight	Weight	Weight
	LCF/HCF					
Environment	Helium	Helium	Helium	Helium	Helium	Helium
Issues	Pressurized			Off normal	Off normal	Off normal
	Water, SCC			Helium	Helium	Helium
Radiation	Not	Negligible	Negligible	Negligible	Negligible	Negligible
Issues	Significant	<0.005DPA	<0.005DPA	<0.005DPA	Avoid Co	Avoid Co
Aging Issues	Some	Some	Some	None, if CC	None, if CC	None, if CC
				Composite	Composite	Composite
Joining Issues	Some	Some	Some	N/A, if CC	N/A, if CC	N/A, if CC
				Composite	Composite	Composite
Manufacturing	None	None	None	Major, if CC	Major, if CC	Major, if CC
Issues				Composite	Composite	Composite

Table 2-4 (continued)
Conditions affecting materials selection metallic NGNP components with 950°C outlet [17]

Condition	**SCS Tube**	**Core Barrel**	**Core Support Floor**	**SCS Shell**	**Inside Shroud**	**Upper Core Restraint**
	316FR	800H	800H	CC	CC	CC
Candidate	800H	316FR	316FR	Composite	Composite	Composite
Materials				Alloy 617	Alloy 230	Alloy 230

Condition	**IHX Indirect****	**Hydrogen HX****	**Hot Duct**	**Bellows**	**He Circulator**	**Primary to Secondary Piping****	**Recuperator**
Normal	*750˚C*	950˚C	600˚C	600˚C	600˚C	950˚C	600˚C
Temp*							
Maximum	*750˚C*	950˚C	700˚C	700˚	600˚C	950˚C	600˚C
Temp*							
Loading	Thermal	7 MPa	Own	Fatigue	Fatigue	7 MPa	
	Transients	Cycles	Weight		Creep fat.		
Environment	Helium	Helium,	Helium	Helium	Helium	Helium	
Issues		Heat				Heat trans	
		Transfer				Fluid	
		Fluid					
Radiation	None	None	None	None	None	None	None
Issues							
Aging	Some	Some	Some	Some	Some	Some	Some

Table 2-4 (continued)
Conditions affecting materials selection metallic NGNP components with 950°C outlet [17]

Condition	IHX Indirect**	Hydrogen HX**	Hot Duct	Bellows	He Circulator	Primary to Secondary Piping**	Recuperator
Issues							
Joining	Some	Some	Some	Some	Some	Some	Some
Issues							
Manufacturing	Major	Some	Some	Major	Some	Major	Some
Issues							
	Alloy 617	Alloy 617	Alloy	Alloy	316FR	Alloy 617	347 SS
Candidate		Alloy 230	800H	800H		Alloy XR	
Materials			316FR	316FR			

* Temperatures indicated are for 950°C helium outlet. A 750°C outlet will reduce temps by 100 to 200°C.

** IHX is replaced by a steam generator (temps revised to 750°C) and there is no conventional piping in the AREVA and X-energy VHTR.

Another table reproduced from the same report is shown in Table 2-5 which provides a list of potential candidate materials for high-temperature metallic components of the NGNP. Although the table was developed for systems with an outlet temperature of 950°C, the materials are still relevant for the VHTRs with outlet of 750°C. (albeit that some higher temperature performing alloys may not be needed due to the actual lower operating temperatures).

Table 2-5
Potential candidate metallic materials selection for high-temperature metallic NGNP components [17]

Nominal Composition	UNS No.	Common Name	Code Max Temp	Data Max Temp	Helium Experience
Ni-16Cr-3Fe-4.5Al-Y		Haynes 214		1040	
63Ni-25Cr-9.5Fe-2.1Al	N06025	VDM 602CA	980	1200	
Ni-25Cr-20Co-Nb-Ti-Al		Inconel 740		815	
60Ni-22Cr-9Mo-3.5Nb	N06625	Inconel 625	900		
59Ni-23Cr-16Mo-Fe-Al	N06059	VDM 59	760		
53Ni-22Cr-14W-Co-Fe-Mo	N06230	Haynes 230	980	1100	
Ni-22Cr-9Mo-18Fe	N06002	Hastelloy X	900	1000	Yes
Ni-22Cr-9Mo-18Fe		Hastelloy XR		1000	Yes
46Ni-27Cr-23Fe-2.75Si	N06095	Nicrofer 45	815		
45Ni-22Cr-12Co-9Mo	N06617	Inconel 617	980	1100	Yes
Ni-33Fe-25Cr	N08120	HR-120	900	930	
35Ni-19Cr-1 1/4Si	N08330	RA330	900		
33Ni-42Fe-21Cr	N08810	Incoloy 800	980	1100	Yes
33Ni-42Fe-21Cr	N08811	800HT	900	1100	
21Ni-30Fe-22Cr-18Co-3Mo-3W	R30566	Haynes 556	900	1040	
18Cr-8Ni	S30409	304H SS	815	870	Yes
16Cr-12Ni-2Mo	S31609	316H SS	815	870	Yes
16Cr-12Ni-2Mo		316FR		700	
18Cr-10Ni-Nb	S34709	347H SS	815	870	
18Cr-10Ni-Nb		347HFG	730	760	
18Cr-9Ni-3Cu-Nb-N		Super 304	815	1000	
15Cr-15Ni-6MnNb-Mo-V	S21500	Esshete 1250	700	900	
20Cr-25Ni-Nb		NF 709		1000	
23Cr-11.5Ni-N-B-Ce		NAR-AH-4	815	1000	

The tables of ASME Code Section II, Part D provide allowable design stresses for nuclear components for temperatures up to 375°C for ferritic alloys and to 425°C for austenitic alloys. Table 2-5 is of particular interest in that it provides materials information related to overall ASME Code applications, including Code Sections applicable to fossil energy components that provide allowable design stress for some materials to 980°C. For nuclear construction, metallic core support structures must conform to ASME Section III, Subsection NG which does not provide allowable design stresses to such high temperatures. It is also important relative to Code requirements whether a component is designated for Class 1 or Class 2 application (note that the Owner of a nuclear power plant shall be responsible for applying system safety criteria to classify the equipment in the nuclear power plant). It is also assumed that the metallic materials will be limited to those in Section II, Part D, so that additional Code-approved materials are needed for high temperature service.

As discussed in Section 1.3 of this report, Division 5 of Section III includes only six materials for Code application at higher temperatures than given in Section II; they are (with maximum allowed temperatures): type 304SS (815°C), type 316SS (815°C), Alloy 800H (750°C), 2¼ Cr-1Mo (650°C), 9Cr-1MoV (650°C) and Alloy 617 (954°C). Note that the use of Alloy 617 (UNS N06617) was recently approved for both low temperature (to 425°C) and high temperature (to 954°C) service construction within Section III, Division 5 of the ASME Code via Code Cases N-872 and N-898, respectively [83, 29]. Many of the materials listed in Table 2-4 were also listed as the primary candidates for application in heat exchangers and core internals in an INL report on reactor down-selection [30]. Alloy 617, Alloy X, and Alloy XR were stated as the leading potential metallic candidates for VHTRs in the earlier Gen IV Integrated Materials Technology Program Plan when the target service temperature was above 760°C. However, this requirement and the need for the very high temperature materials has been mitigated with the lower outlet temperature of 750°C for the two VHTRs reviewed in this report. Thus, Alloy 800H is certainly a leading candidate for service at 600 to 750°C, especially since it is already included in ASME Code Section III, Division 5. For temperatures at 600°C and lower, the list of potential candidates is longer, with the other alloys in Section III, Division 5 being leading candidates. Of course, as stated earlier, while operation temperature is critical for material selection, coolant compatibility and resistance to irradiation are also important for long-term deployment.

In the 2014 Technology Roadmap Update for Generation IV Nuclear Energy Systems [18], it was stated: “Internal core structures and cooling systems, such as intermediate heat exchangers, hot gas ducts, process components and isolation valves, that are in contact with hot helium, can use current metallic materials up to a core outlet temperature of about 700 to 800°C.” Tables 2-4 and 2-5 list Type 316FR stainless steel as a potential material for many components. This material is a low carbon and medium nitrogen stainless steel developed for fast breeder reactor structural materials. Creep rupture experiments at 550 and 600°C were performed for up to 30,000h for comparison with conventional medium carbon and low nitrogen type SUS316 steel (SUS is the Japanese Industrial Standards designation) for stainless grades. The 316FR had higher rupture strength (see Figure 2-6) and ductility than SUS316 for long-term creep. Microstructural investigations revealed important differences with Laves phase precipitation mainly on the grain boundaries at 550 and 600°C and also in the matrix after 10,000 h creep at 600°C in 316FR, but the Laves phase on the grain boundary did not decrease rupture ductility. The solid solution hardening by nitrogen was effective for a long period of time, because of the extremely small number of nitrides precipitated during creep. Thus, the 316FR had higher rupture strength and ductility than SUS316 due to its higher phase stability during high temperature exposure. The Type 316FR material is not currently approved in the ASME Code, but a relatively large database should make Code approval less arduous [17].

Many of the components will require some form of insulation to improve thermal performance, although, again, the requirements are mitigated from the previous 950°C outlet design to the 750°C outlet design. High temperature fibrous insulation in the cross vessel, the upper plenum shroud, the SCS helium inlet plenum, and the turbocompressor are among the applications. They are generally assumed to be installed for lifetime operation. Such insulation has been used in the Ft. Saint Vrain reactor, the AVR, and others. Information provided in Corwin, et al. [17] lists Kaowool and Quartz-et-Silica fibrous mats as ones for which experience exists. For applications where significant structural support is not required, that same report lists Al_2O_3 and SiO_2 mixed ceramic fiber mats contained between metallic or carbon-carbon composite cover plates as potential materials. Graphite-based insulation is also an alternative, but the maximum temperature ratings for these insulations are 1000°C and higher, indicating their use in the AREVA and X-energy reactors should not be an issue needing resolution. It is also pointed out in that report that the canisters that contain the insulation will be exposed to the same temperatures, although they are not listed as structural materials, so selection of canister materials should be chosen from those in Table 2-5 and, moreover, from the currently approved materials in Section III, Division 5.

The PIRT analysis exercises of Ball, et al. [15] for the NRC ranked various subcomponents and phenomena as to their importance and available knowledge. This was followed by a gap analysis [16] to identify the issues needing additional research and development or analysis for application in the NGNP. However, the project was performed for an NGNP with helium outlet temperature of 950°C, thus, many of the issues identified as being of high importance may not be of such high importance, or even relevant, for a reactor with an outlet temperature of 750°C since the lower operating temperatures may allow for selection of materials for which the required data are already available. The gap analysis identified items needing near term needs and those for intermediate and long-term needs. Additionally, the INL White Paper [31] essentially updated those studies for the revised outlet of 750°C. The generic discussion of components, critical performance issues and candidate materials for VHTR's that is presented below is based significantly on the information presented in those previous documents. To provide a logical flow for the discussion of the materials' selection issues and gaps the discussion is presented for the specific components of the system and is organized by sub-component and identified issues.

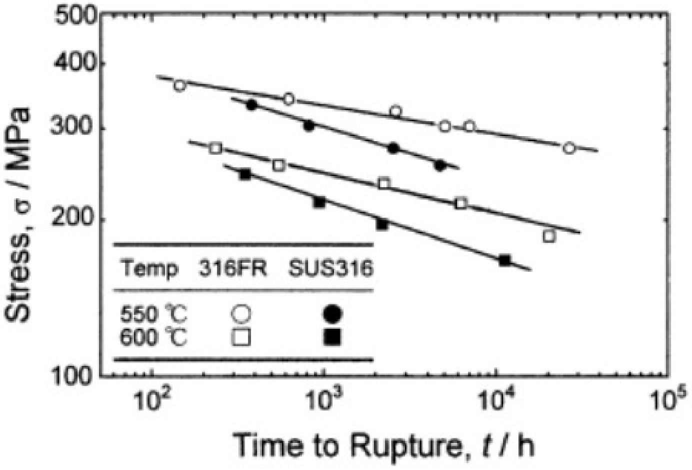


Figure 2-6
Creep rupture strength of 316FR and SUS316 at 550 and 600°C [84]

2.3.1 Control Rods and Guide Tubes (Metallic)

Given the outlet temperature of 750°C, the control rod guide tubes operate below 400°C during normal operation [31]. The leading candidate materials are Hastelloy X and 800H, with the specific material selection based on off-normal temperature expectations. Radiation-induced degradation, embrittlement, swelling, and radiation creep of 800H are noted in the PIRT as remaining issues for long term application. There is limited information on swelling and ductility at moderate doses, but none on radiation creep. However, as discussed by Rowcliffe, et al. [32], the components for which Ni-base alloys (such as 800H) are being considered will accumulate over the design lifetime maximum doses of only a few displacements per atom (dpa); “consequently, many of the radiation effects of interest (radiation hardening, flow localization, void swelling and irradiation creep), are unlikely to have any significant impact on alloy performance.”

On the other hand, because of the high operating temperatures and the potential for very high off-normal temperatures, the generation of helium and the subsequent migration and clustering to form gas bubbles needs to be carefully assessed relatively to the potential for grain boundary embrittlement under sufficiently high stress [31]. A technical maturity assessment report by INL [32] stated: “Metallic control rod drive tubes and seals, however, may fail in the event of the most severe loss-of-forced-cooling events, with subsequent depressurization of the core. While this is not expected to cause significant fuel failure, circulating radiological inventory would be released and expensive core repairs would be necessary. Qualification of new alloys or even the use of carbon or SiC composites for the guide tubes may be needed.”

2.3.2 Control Rods and Guide Tubes (Non-Metallic)

If ceramic-composite materials are selected, the limits of fluence need to be known for both C-C and SiC-SiC materials to assess the potential radiation-induced degradation. Due to long-term exposure to low partial pressure of oxygen, more rapid oxidation during air ingress may occur and information is needed.

2.3.3 RPV Internals (Metallic)

The leading materials include type 316 stainless steel and 9Cr-1MoV (Grade 91) steel, but Grade 22 of 2.25Cr-1Mo may also be a candidate for the core barrel given that its allowable stress is similar to that of the Grade 91 steel up to about 430°C [31]. Emissivity of the core barrel is ranked as highly important, but this aspect is highly dependent on the overall reactor system design, heat removal systems, and so on. Although Type 316L stainless steel is a potential candidate for the core barrel, emissivity of that material is not very high and emissivity is an important issue for the core barrel in helping with heat removal during conduction cooldown events. Fast neutron fluence for core support structures and internals are expected to be much lower than 1×10^{19} n/cm². That value of fluence is well below that for which irradiation effects in austenitic stainless steels would be an issue. However, this value needs to be confirmed, especially if steels like 2.25Cr-1Mo are selected since irradiation effects can be much more significant in these ferritic materials at such levels of fluence.

Creep and other time dependent properties for the core barrel were identified by the PIRT as issues for long term consideration and, though such issues were considered for incorporation of the candidate materials into Section III, Division 5, consideration needs to be given to those properties for a 60-year design lifetime.

2.3.4 RPV Internals (Non-Metallic)

Insulation of the core restraint structures are subject to radiation-induced degradation and oxidation. Effects of long-term exposure to low partial pressure of oxygen and more rapid oxidation are well known. There is C-C composites' experience from other industries (for example, aerospace, and so on) that should be assessed, but, as mentioned for other non-metallic components, the relevant issues are highly dependent on the specific component design. There are some ceramic insulations such as baked carbon and fused or sintered quartz that appear to offer good insulating properties, but it is not known if the current reactor designs propose to use such insulations.

Radiation effects on potential fibrous insulation with regard to their thermal stability and degradation of performance are not well known. There is very limited information available and, if these materials are proposed for application, those issues need to be assessed.

2.4 Materials for Reactor Pressure Vessel

The materials selection issues for the RPV include operating temperature (for performance and thermal aging), irradiation conditions, coolant compatibility, vessel size (for fabrication and transport), and accident conditions. Further, as noted in the PIRT performed for the NRC, fatigue, emissivity, and creep were assessed, but the primary material was 9Cr-1MoV steel. Although many previous assessments for VHTRs have evaluated the use of higher temperature capable steels (such as 9Cr-1MoV [17], both the AREVA/NGNP Alliance and X-energy VHTR designs have chosen to cool the RPV and cross vessel to enable construction of the vessel shells with low alloy steels typically used for LWR vessels, meaning forging grades within the SA-508 specification or plates within the SA-533 specification, and welds. For the Xe-100 RPV, X-energy specifies SA-508, Grade 3, Class 2 for some of the RPV components (vessel flange, top head CRDM housings); that specification has the same chemical requirements as those for Class 1 but with about 30% higher specified minimum yield strength. X-energy also specifies

SB-637, Alloy 718 for top head fasteners, and SA-540, Grade B24, class 1 for center and bottom manway fasteners. These materials are not included in Section III, Division 5 so properties at the anticipated temperatures need to be assessed. The AREVA/NGNP Alliance design does not provide the specification details for the forging or plate, but the chemical composition and mechanical property requirements for SA-533 Type B Class 1 plates used in many LWRs are nominally the same as those for SA-508, Grade 3, Class 1 forgings. The choice of plate or forging for the RPV shell will most likely be based on vessel diameter and fabricator capability, but it is normally preferable to use ring forgings to eliminate the need for longitudinal (axial) welds in the RPV shell.

In order to evaluate potential fatigue issues, operational details are need, although high cycle fatigue in LWRs has not been an issue and is not expected to be an issue for the VHTRs. However, the potential for fatigue issues needs to be assessed following more complete system design, and when there is an extensive database for the RPV steels.

Relative to conditions during accidental temperature excursions, high temperature properties are important if the RPV temperature is anticipated to exceed the normal limit of 371°C. If Section III, Division 5 is deemed applicable, the temperature and time limits that are specified consider strength, fatigue, creep, and stress to rupture. As stated in the INL White Paper [31], the reactor design will conform to the Division 5 limits so that further analysis of those properties will not be required. However, a note of caution for designing the reactor vessel in accordance with Division 5 is that creep data analysis for extrapolating to 60-year of operation by Jetter, et al. [34] resulted in their statement that the SA-533 material will require consideration of creep effects when evaluating expected performance against strain limits and creep-fatigue damage in accordance with the requirements of NH (now Division 5).

More than 50 years of LWR experience with these RPV alloys (SA-508 and 533) has seen improvements in weldability and toughness [34], in addition to improved resistance to thermal aging and radiation resistance through limitations in chemical composition, specifically carbon, sulfur, phosphorus, and copper [35]. The AREVA/NGNP Alliance RPV is similar in size to that of a large boiling water reactor and, with internal pressures of 6-7 MPa, the wall thickness will be on the order of 3 to 4 in. thick; thus, in the absence of specific citing or other unknown issues, fabrication (shop fabrication, welding, and heat treatment) and transport of the vessel are not seen to be major issues. X-energy also specifies that their RPV is road transportable.

As mentioned earlier in Section 1.3, the former ASME Code Case N-499 of Section III, Subsection NB was annulled in 2012 and its provisions incorporated into Section III, Division 5, High Temperature Reactors. This portion of the Code can be applied provided the RPV wall temperature can be maintained at or below 700°F (370°C), except for limited time-temperature excursions as specified in Article HBB-II-3000 for Level B, C, and D events. Specifically, Article HBB-II-3000 permits temperature excursions to 800°F (425°C) for a total accumulation of 3000 hours, for an accumulated time of 1000 hours between 800°F and 1000°F (425°C and 540°C), and only three excursions anticipated in the latter range. However, Article HBB-II-300 applies specifically only to SA-533 Type B Class 1 and SA-508 Grade 3 Class 1 and their weldments; it does not apply to SA-508 Grade 3 Class 2 material which is specified for the vessel flange and top head CRDM housings in the X-energy Xe-100 RPV.

As pointed out in Corwin, et al. [17], emissivity of the RPV is an issue for the passive heat removal system to function properly. As mentioned in Section 2.1.1, the AREVA/NGNP Alliance claims that, even if all electrical power is lost and the shutdown cooling system and other cooling paths cannot operate, heat transfer through the reactor building structure to the ground is sufficient to maintain reactor temperatures within acceptable limits. For the Xe100 reactor, it is claimed that the structural graphite provides a heat removal path and heat capacitance during loss of forced flow events, but there is insufficient information at this time to know the anticipated maximum RPV temperature. The PIRT exercise ranked emissivity as high importance and with low knowledge for the 9Cr-1MoV steel, but the INL also emphasized its importance for the SA-533/SA-508 steels due its effect on radiating heat energy during a conduction cooldown event. Thus, the issue of emissivity needs to be assessed based on the design details for each VHTR system and "measurements of emissivity will be required after oxidation in air and helium to determine the most appropriate values [31].' Moreover, it is not known if either reactor design may incorporate some high emissivity coating on the RPV.

Given the active RPV cooling case for both reactors and the specification of low alloy steels used in hundreds of operating LWRs, material performance issues for a 60-year design life are potential effects of neutron irradiation, thermal aging, and material compatibility with the coolant.

For the Xe-100 RPV, SA-540, B24, Class 1 (a low Cr-Ni high-strength steel, essentially 4340) is specified for the manway fasteners, while SB-637, Alloy 718 [a precipitation-hardening nickel base (~50% Ni) Ni-Cr-Mo-Nb high-strength alloy] is specified for the RPV top head fasteners. The SB-637 is a specification for high-temperature service, indicating that the RPV top head is not cooled to the same temperature as the RPV shell.

The estimated neutron irradiation doses are not specified for either of the VHTR designs, but previous evaluations for the NGNP have indicated that most designs anticipate a somewhat lower fast neutron fluence relative to that of many LWRs, that is, on the order of $\sim 1\text{-}10 \times 10^{22}$ n/m^2 (>1 MeV), although the estimated fluences for the two VHTRs are not specifically known. Given an RPV temperature of about 300°C, changes in mechanical or physical properties due to irradiation effects at such a fluence level would be expected to be insignificant based on current experience with the low alloy steels selected. Even in the case of a material with the maximum allowed contents of radiation-sensitive elements (that is, Cu, Ni, Mn, P, Si) and a fast neutron fluence of 1×10^{23} n/m^2 (>1 MeV), the highest predicted irradiation-induced transition temperature shift using the three embrittlement trend curves usually employed in the USA is less than 40°C as shown in Figure 2-7: NRC Reg. Guide 1.99-Rev. 2 [37], ASTM E900-15 [38], and the Eason, Odette, Nanstad, Yamamoto (EONY) correlation [39] of 10CFR50.61a [40]. Such a conclusion, however, can only be made once the neutron fluence estimations for the specific reactors are available.

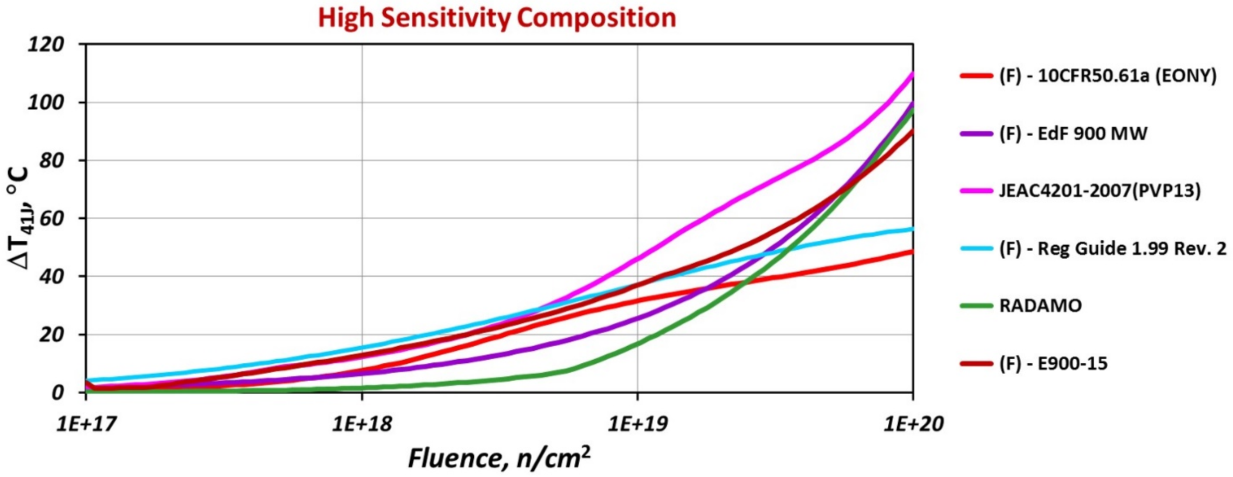


Figure 2-7
Comparison of trend curves for SA-508 Grade 3, Class 1 forging with maximum allowed chemical compositions of Cu, Ni, Mn, P, Si

Thermal aging effects for the selected low alloy steels and welds in the range of 320°C are not considered to be an issue for operating times of about 30 years as shown in Figure 2-8 [36] for effects of aging time at various temperatures on ductile-brittle transition temperature (DBTT) shifts [36]. For RPV steels with typical composition and aging temperatures of 280 to 320°C, the database indicates DBTT shifts from 0 to 30°C for aging time to about 30 effective full power years ~263,000 h). However, for a proposed 60-year design life (equivalent to about ~500,000 h, depending on the plant's capacity factor), there are no available data. As discussed in Nanstad, et al. [36], the mechanisms leading to thermal embrittlement are complex, with chemical composition, heat treatment, and aging temperature as significant variables that are further confounded by simultaneous effects of irradiation. Because the RPV operating temperature for VHTRs may be higher than 320°C and with the potential for very high thermal exposure in accident conditions, this issue is not resolved until more detailed expected thermal exposure over the plant's life is known.

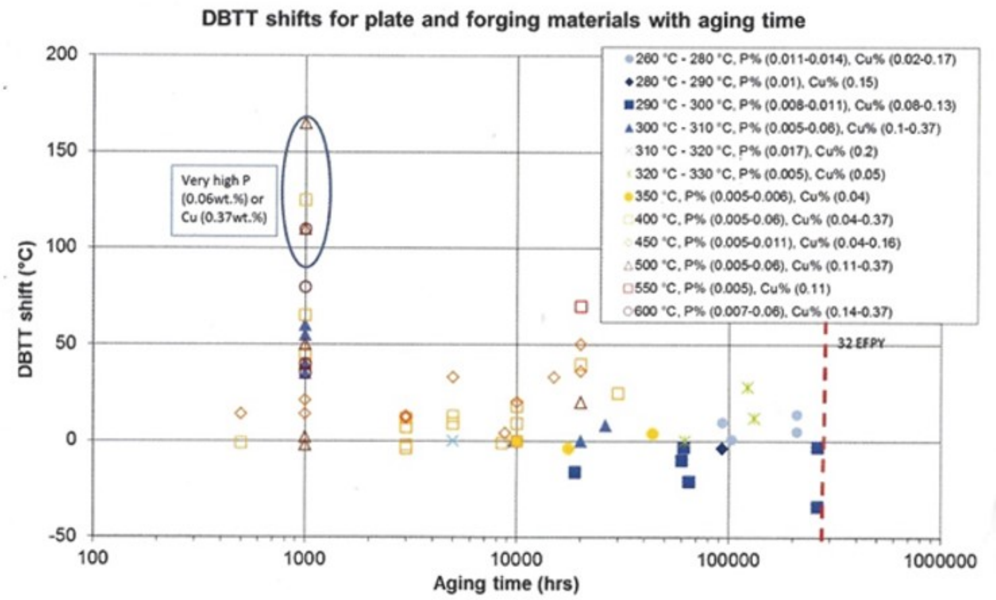


Figure 2-8
Ductile-Brittle transition temperature shifts for SA-508 and 533 plate and forging materials with aging time [36]

Although the potential effects of oxidation from the helium coolant during normal operation as well as during hot transient conditions were raised by INL, it was concluded that those effects are not expected to be significant and no new data should be required [41].

2.5 Materials for Power Conversion System

2.5.1 Power Conversion Vessels, Turbomachinery and Circulators

All turbomachinery and pressure boundary materials operating at steam temperatures of ~565°C are available from commonly used fossil energy power systems; see, for example, Viswanathan and Baker [42], and Viswanathan, et al. [43].

For various issues aspects associated with creep and fatigue, rotational stress, and so on, in the power conversion systems (PCS), such issues need evaluation based on the actual design. An assessment is also needed to evaluate potential for missile generation and debris generation from turbomachinery and the associated consequences.

Primary coolant contamination: Assessment of actual design to evaluate potential for contamination from bearings. If significant, additional environmental studies of carburizing coolant on high temperature metallic components may be required.

2.5.2 Piping

The only "piping" is the cross-duct between the RPV and the steam generator shell; thermal aging concerns for this piping is the same as for the RPV. Fatigue loading is expected to be low, but evaluation of HAZ in weldments should be made if a high number of cycles are expected.

Regarding environmental degradation of insulation, all designs for hot duct liners are expected to utilize metallic materials (for example, Hastelloy X or 800H) rather than ceramics or composites. Thus, an assessment of the actual design is recommended to resolve the relevance of potential issues associated with insulation debris generation within the primary circuit [15].

2.5.3 Steam Generator Vessel and Cross Vessel

Materials of construction will likely be the same SA-508/SA-533 grades of steel as for the RPV. Because the steam generator (SG) vessel and cross vessel are not significantly affected by conduction cooldown events or irradiation, the factors of highest concern are mechanical properties for long term operation and corrosion resistance. Thus, emissivity and thermal aging require consideration for these two components and consideration of a surveillance program for thermal aging is recommended for the vessels.

Regarding fatigue, potential response to such issues as thermal-hydraulic vibrations, crack initiation and subcritical crack growth for this component expressed in the PIRT [15], the loading conditions in the SG and shroud are not known. As expressed for the RPV, design performed in accordance with the Division 5 limits is expected to be sufficient, with a caution about extrapolation of creep to 60 years.

2.5.4 Steam Generator

SG tubes are likely to be fabricated from 800H for use in the high temperature region (~620°C maximum) and possibly from 2.25Cr-1Mo in the cooler region (~400°C maximum). Division 5 of Section III only includes use of the annealed 2.25Cr-1Mo, not V-modified or the Normalized and Tempered versions of this steel. There are system concerns related to the possible failure of tubes due to steam pressure in the secondary circuit being greater than the helium pressure in the primary circuit. A considerable knowledge base exists for the properties of 2.25Cr-1Mo alloys, but at the time of the white paper work [31] additional development was proceeding on 800H, including tests in HTGR helium.

As stated in the INL White Paper, preliminary results of studies sponsored by ASME Standards and Technology, LLC have determined that there is currently sufficient information available to extend Code qualification of 800H to a maximum use temperature of 850°C and for 500,000 hours design life for lower temperatures, but these data have not yet been developed in Division 5 applications. It is also important to note, however, that Alloy 800H has been used in high temperature components of constructed and operated HTGRs (for example, FSV, AVR, THTR) and for over 20 years in the case of the AVR. Additionally, many experimental studies have been performed over the decades to evaluate creep rate and fatigue of this material (both low-cycle and high-cycle) at temperatures to 1000°C. They further note that the results of the experiments form the foundation for the Code provisions in Division 5 for the use of Alloy 800H [31]. Thus, although thermal aging has been designated a potential issue for 60-year application, the amount of actual operating experience combined with extensive research and development results provides a relatively firm basis for selecting alloy 800H for many of these high temperature components in the VHTR. However, it is possible that type 316L stainless steel could be selected for the tubes given the maximum allowed operating temperature of 815°C.

The PIRT analysis concluded that dissimilar metal joints between 800H and 2.25Cr need design review, as does potential for fretting at supports. Moreover, there are concerns regarding long-term loading with significant unexpected thermal gradients and other stresses (for example, graphite particle build-up) to compel evaluation of the actual design for potential plastic collapse, and also for erosion, within the steam generator.

Dissimilar metal welds between 800H and 2.25Cr-1Mo may be incorporated in the designs and specific consideration is needed especially for corrosion questions where alternating wet-dry conditions may exist [31].

2.5.5 Valves

Isolation valve failure was ranked with high importance and low knowledge by the PIRT. Not much is known about He-leak tightness in large valves, but the designer must evaluate depending on design.

2.5.6 Reactor Cavity Cooling System (RCCS)

Change in RCCS panel emissivity with time was ranked as an issue for further review by the PIRT, but it is not clear if the RCCS panel emissivity is needed to maintain fuel temperature below the limit during an accident condition. Thus, this topic needs evaluation based on the specific reactor design.

2.6 Materials for Graphite Components

Graphite is one of the key critical materials in the VHTR systems as it is used as a containment for the fuel and as a nuclear moderator. It is also a structural material for the AREVA system with a prismatic core design. The graphite components need to maintain core geometry, to passively remove core heat during off-normal events, and control chemical attack by limiting oxidation for events involving ingress of water or air gas mixtures [31,44]. Graphite is attractive as a moderator for many reasons, including that it can reach very high temperatures that will allow for the "doppler effect" to come into play which will shut down the reactor; implying a passively safe nuclear system [44]. Graphite has been used for decades as a moderator due to its nuclear properties, such as a high neutron scattering cross section and low neutron absorbent cross section. Moreover, it also has good structural strength and machinability. Its thermal properties allow for a large mass of graphite to act as a heat sink so that thermal transients are considerably slower than in other systems. Graphite is, however, sensitive to neutron radiation effects on its mechanical properties and dimensional stability, with an example shown in Figure 2-9. At extremely high fluence levels, the accumulation of pores and microcracks effectively leads to a loss of the material integrity or cohesion, designated the cohesive life limit. The figure also demonstrates the high dependence on irradiation temperature. Although there are many differences in the manufacturing processes for different grades, graphite is basically made with a filler coke and pitch binder, with the coke being either petroleum or coal-tar-derived and an isotropic or near-isotropic material is desirable for nuclear applications to minimize irradiation-induced dimensional changes and resultant lower internal stresses in the graphite structures. The reader is referred to Marsden, et al. [44] for discussion of graphite manufacturing details.

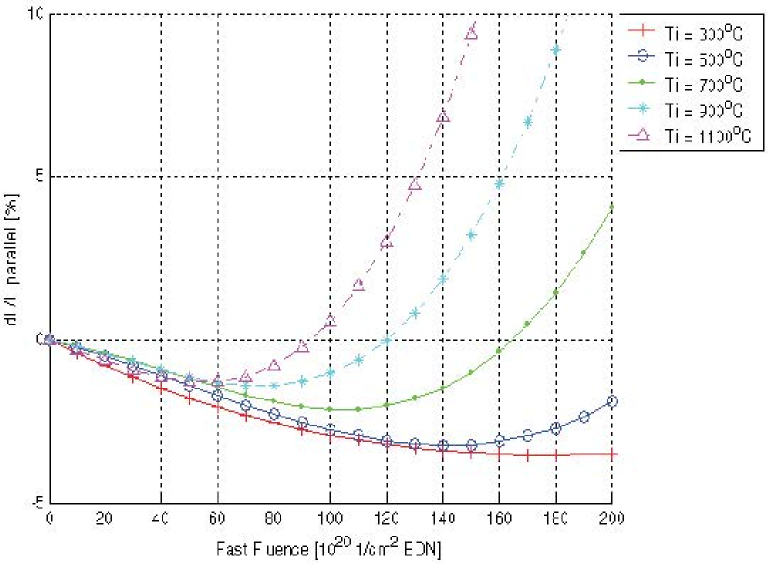


Figure 2-9
Typical irradiation-induced dimensional changes in reactor graphite, parallel direction [45]

Various grades of graphite have been used for reactors, with one of the well-known grades, H-451, no longer available. Table 2-6 was taken from Corwin, et al. [17] and shows the various candidate graphite grades for the components of both prismatic and pebble bed systems considered in 2005 and with updated information [30] or this report. These changes are also reflected by the fact that three of the above indicated candidates, NBG-17, NBG-18, and IG-110, are currently under evaluation within the VHTR Graphite R&D Program at INL [31] or comparison with the Graftech PCEA. Also, the NBG-17, NBG-18, and Toyo Tanso IG-110 are also listed as candidate grades in the INL White Paper, which was published in 2012 [31]. The table showing old and current grades is included here to demonstrate the evolving nature of graphite availability. It is recognized that components that cannot be designed for the full plant life (such as reflector graphite components in high flux regions) need to be designed to be replaceable [31] but there are some permanent reflector components normally designed for the plant lifetime.

The grades of graphite ultimately deployed in the reactors must meet the requirements of the appropriate ASTM International specifications (for example, ASTM D7219, D7301) and a radiation effects database must be developed for the physical properties (density, coefficient of thermal expansion, thermal conductivity), mechanical properties (Young's modulus, strength, radiation induced creep, fracture toughness), dimensional stability, and oxidation properties. The properties most relevant for ideal nuclear reflector graphite are shown in Table 2-7. Moreover, the ASME Code Section III, Division 5 includes rules for design, construction, examination, and testing of Graphite Core Components and Graphite Core Assemblies. There is a substantial database for the afore-mentioned Grade H-451, including effects of irradiation, which allows for other grades to be selected in preliminary designs with enough data to verify that the behavior is similar to that of previously "qualified" grades such as the H-451. It is notable that the Toyo Tanso Grade IG-110 was used in the Japanese HTTR for fuel blocks and in the Chinese HTR-10 pebble bed reactor [17]. Other graphite grades listed by Marsden, et al. [44] include those that have been irradiated in various European Framework programs: MG-1, SFC-1, MG-2, FG-1, and FG-2.

Table 2-6
Candidate grades of graphite for the core components of the VHTR

NGNP Concept	Component Description	Candidate Grades*
Prismatic Block	Fuel Element and Replaceable Reflector	Graftek PCEA SGL Carbon NBG-17 Toyo Tanso IG-110 *or -430*
Prismatic Block	Large Permanent Reflector	Graftek PGX SGL Carbon HLM
Prismatic Block	Core Support Pedestals and Blocks	Graftek PCEA SGL Carbon NBG-10 or-17 or-18 Mersen 2114 Carbone USA 2020 Toyo Tanso IG-110 or-430
Prismatic Block	Floor Blocks and Insulation Blocks	Graftek PCEA SGL Carbon NBG-18
Pebble Bed	Reflector Structure	Graftek PCEA SGL Carbon NBG-18 Toyo Tanso IG-110
Pebble Bed	Insulation Blocks	Graftek PCEA SGL Carbon NBG-10 or-18

* Note: Items in red are no longer candidates, items in blue are newly recommended; Graftech PCEA and NBG-10 are no longer available, Carbone Lorraine USA became Mersen in 2010. IG-430 is no longer recommended.

Thus, there are numerous grades of graphite that have been identified as potential candidates for both the AREVA and X-energy VHTRs, but the selection for the various components will depend on the continued development of adequate databases. In that regard, X-energy has an irradiation program underway at ORNL for their reactor graphite reflector material and has developed lifetime analysis tools for predicting graphite lifetime using the guidelines in accordance with ASME Section III, Division 5 [26].

Table 2-7
Ideal requirements for reflector graphite [31]

Property	Required Range	Reason	Performance Attributes
Density	1.7–1.9 g/cm^3	High density is indicative of lower porosity, provides for more effective neutron moderation/reflection per unit volume, and in general, also indicates higher strength.	Neutron efficiency Structural integrity
Neutron absorption cross-section	<5 mbarn	Required for neutron efficiency of the core. The limiting neutron absorbency is that of pure carbon (~3.5 mbarn).	Neutron efficiency
Thermal conductivity at room temperature	>100 W/m-K	Indicative of a high degree of graphitization and typically the level required for effective heat transfer in HTGR applications.	Heat transport
Purity (total ash content)	<300 ppm	Required to minimize activation and reduce susceptibility to catalytic oxidation. It is possible to manufacture graphite with much higher purity levels using a dedicated purification step. The selected and specified purity may vary depending on the function of the components. This decision will be based on a cost-benefit analysis.	Component activity levels during replacement and/or disposal. Graphite Oxidation under normal and accident conditions.
Tensile strength	>15 MPa (tensile)	Adequate strength is required for structural component integrity. Higher strengths are achievable with isostatic- molded, fine grain graphite, but these typically possess lower fracture toughness. This is a trade-off that must be taken into account in the design.	Structural integrity
CTE (20 to 500°C)	3.5 to 5.5 × 10^{-6} K^{-1}	A higher value is indicative of the coke isotropy and hence isotropy of the graphite. This implies that the graphite will have better dimensional stability when subjected to fast neutron irradiation. However, lower CTE can be beneficial in terms of thermal stress.	Structural integrity
CTE isotropy ratio	<1.10	Indicative of the bulk graphite isotropy.	Structural integrity

Table 2-7 (continued)
Ideal requirements for reflector graphite [31]

Property	Required Range	Reason	Performance Attributes
Dynamic elastic modulus	8–15 GPa	Higher modulus is typically associated with a higher strength material, but increased sensitivity to thermal stresses. Thus, values at the lower end tend to be more beneficial.	Structural integrity
Dimensional changes with irradiation	Minimal shrinkage over the applicable fluence range and minimal differences in the with-grain and against-grain directions.	This is mainly a function of temperature and fluence but is strongly dependent on the graphite grade. Dimensional changes strongly influence the level of internal stresses generated in core components when subjected to fast neutron irradiation and are critical in determining their useful life.	Structural integrity

The PIRT exercise referred to earlier also assessed phenomena related to nuclear graphite for the VHTR with outlet of 950°C. At that time, the phenomena with high importance and low knowledge related to irradiation-induced changes in creep, coefficient of thermal expansion, mechanical properties (strength, toughness), effects of creep strain, and blockage of a coolant channel due to graphite failure and/or graphite spalling. Other issues of high importance and medium knowledge related to irradiation-induced changes in various elastic constants, thermal conductivity, dimensional stability, and statistical variation of nonirradiated properties. Although the outlet temperature for the current VHTRs is 200°C lower, most of the identified issues are still relevant because the available grades of graphite have changed and there are insufficient data to assess behavior for a 60-year design lifetime. Moreover, as stated earlier, there is a substantial database for the Grade H-451, including effects of irradiation, which allows for other grades to be selected in preliminary designs with enough data to verify that the behavior is similar to previously “qualified” grades such as the H-451. It is also noted in the INL White Paper [31] that “experience indicates that materials produced using similar source materials and processing will possess similar as-manufactured properties and will exhibit similar trends in behavior under irradiation.” Irradiation effects on material properties (expansion/contraction, thermal conductivity), and consistency of graphite quality and performance over the service life, are the primary issues associated with graphite for the current VHTR designs.

In summary, the technology gaps for graphite identified by the PIRT panel [15,16] and also reflected in the INL White Paper [31] are as follows:

- Graphite supply (coke sources, graphite vendors)
- Confirmatory data for new grades being considered for the NGNP
- Irradiation creep data and effect of creep on properties of candidate NGNP graphites
- Consensus design codes and materials testing standards
- Extension of current theoretical models to higher doses and temperatures

- Development of improved understanding and models for neutron irradiation induced displacement damage in graphite
- Development of whole-core structural models
- NDE methods for use in and out of core and
- Graphite analytical models for oxidation, property changes, and dimensional changes and creep induced by irradiation

2.7 Summary of VHTR Material Properties Review

This review considered two proposed and in-development VHTRs, the AREVA/NGNP Alliance Reactor and the X-energy Xe-100 Reactor. Both are helium-cooled reactors with a gas outlet temperature of 750°C. The AREVA system uses the prismatic (graphite blocks) design with TRISO fuel particles, while the X-energy system uses a pebble-bed design for the core with TRISO fuel particles embedded in larger graphite particles. Three major components comprise their systems; 1) a helium cooled nuclear reactor, 2) a heat transport system (steam generator), and 3) a cross vessel for helium flow from the reactor to the heat transport system. Although many details of operating temperatures, irradiation doses, and so on, are not available for the many individual components of the systems, the information that is available for each of them as well as detailed information from previous studies of VHTRs have been utilized to evaluate the gaps in knowledge that need either more detailed data to make a valid assessment of designs or need additional research and development for deployment.

The GIF Annual Report (2017) [27] stated that, because the outlet temperatures are limited to 750°C, current state-of-the art materials and components are applicable. Relative to the subject of heat utilization systems materials for the VHTR, the GIF Roadmap (2014) [18] stated that internal core structures and cooling systems, such as intermediate heat exchangers, hot gas ducts, process components and isolation valves, that are in contact with hot helium, can use current metallic materials up to a core outlet temperature of about 700 to 800°C. That document further noted that efforts in the United States have focused on developing the data needed to extend Alloy 800H for use up to 850°C and Alloy 617 for use up to 950°C, and that within the next four years, all of the data needed to codify these materials will be provided to the ASME.

This review has identified issues that need resolution for the VHTRs, either due to insufficient information to judge the issue, or issues that have been noted to not have sufficient knowledge. In many cases, the most prominent factor is the need for sufficient data/information for a 60-year design lifetime.

- Type 316FR material has improved properties relative to type 316 for long lifetimes; it is not currently approved in the ASME Code, but a fairly large database should make Code approval relatively easy.
- If composite materials are selected for internal components, the actual fluence and the limits of fluence need to be known for both C-C and SiC-SiC materials to assess the potential radiation-induced degradation. Due to long-term exposure to low partial pressure of oxygen, more rapid oxidation during air ingress may occur and additional information is needed.

- Fast neutron fluence for core support structures and internals are expected to be much lower than 1×10^{19} n/cm², but this value needs to be confirmed, especially if 2.25Cr-1Mo is selected to benchmark the alloys.
- Creep and other time dependent properties for the core barrel were identified by the PIRT [28] as issues for long term consideration and, though such issues were considered for incorporation of the candidate materials into Section III, Division 5, specific consideration needs to be given to those properties for a 60-year design lifetime.
- Radiation effects on potential fibrous insulation as affecting degradation and thermal stability are not well known. There is very limited information available and, if proposed for application, those issues need to be reviewed.
- For the Xe-100 RPV, X-energy specifies SA-508, Grade 3, Class 2 for some of the RPV components (vessel flange, top head CRDM housings); that specification has the same chemical requirements as those for Class 1 but with about 30% higher specified minimum yield strength. X-energy also specifies SB-637, Alloy 718 for top head fasteners, and SA-540, Grade B24, Class 1 for center and bottom manway fasteners. These materials are not included in Section III, Division 5, so the anticipated temperatures need to be assessed.
- Fatigue for the RPV is not expected to be an issue, but the potential for fatigue issues needs to be assessed following more complete system design for a 60-year lifetime.
- If the RPV is expected to exceed 371°C, SA-508/533 material will require consideration of creep effects for a 60-year lifetime when evaluating strain limits and creep-fatigue damage in accordance with the requirements of Section III, Division 5.
- Section III, Division 5 does not include SA-508 Grade 3 Class 2 material which is specified for the vessel flange and top head CRDM housings in the X-energy Xe-100 RPV, and maximum anticipated temperatures for those components are not known.
- For the RPV and cross vessel, the issue of emissivity needs to be assessed based on the design details for each VHTR system and “measurements of emissivity will be required after oxidation in air and helium to determine the most appropriate values [31].” Moreover, it is not known if either reactor design may incorporate some high emissivity coating on the RPV. Insulation and cooling methods to keep the vessels below 371°C are not described.
- For various aspects associated with creep and fatigue, rotational stress, and so on, in the PCS, such issues need evaluation based on the actual design. An assessment is also needed to evaluate potential for missile generation and debris generation from turbomachinery and the associated consequences.
- Potential thermal aging of the low-alloy steels for the RPV, cross vessel, and steam generator requires consideration based on the design anticipated temperatures. Consideration of a surveillance program for thermal aging is recommended for the vessels.
- The PIRT analysis concluded that dissimilar metal joints between 800H and 2.25Cr-1Mo steel, if incorporated in any components, need design review, as does potential for fretting at supports. Specific consideration is needed especially for corrosion questions where alternating wet-dry conditions may exist. Moreover, there are concerns regarding long-term

loading with significant unexpected thermal gradients and other stresses (for example, graphite particle build-up) to compel evaluation of the actual design for potential plastic collapse, and also for erosion, within the steam generator.

- Change in RCCS panel emissivity with time was listed as an issue for further review by the PIRT for NRC, but it is not clear if the RCCS panel emissivity is needed to maintain fuel temperature below the limit during an accident condition in either VHTR. Thus, this topic needs evaluation based on each specific reactor design.

The list of technology gaps for graphite identified in this review, those by the PIRT panel [15, 16], and also reflected in the INL White Paper [18] for long term operation are as follows:

- Irradiation effects on material properties (creep, expansion/contraction, thermal conductivity), consistency of graphite quality, and performance over the service life are the primary issues associated with graphite for the current VHTR designs.
- There are numerous grades of graphite that have been identified as potential candidates for both the AREVA and X-energy VHTRs, but the selection for the various components will depend on the continued development of adequate databases.
- Availability of graphite supply (coke sources, graphite vendors) for long lifetime designs (for example, 60 years) is an issue.
- Consensus design codes and materials testing standards need continuing development.
- Development of improved understanding and models for neutron irradiation-induced displacement damage in graphite, and development of whole-core structural models are needed.
- Development of non-destructive examination/inspection methods for use in and out of core is an identified need.
- To enhance predictive reliability, continued development of graphite analytical models for oxidation, property changes, and dimensional changes and creep induced by irradiation is needed.

3

GFR DESIGN

3.1 GFR Component Descriptions and Operating Conditions

There have been various designs over the decades for GFRs, including the Energy Multiplier Module (EM^2) by General Atomics (GA) [46]. The GA design is a high temperature direct Brayton cycle helium cooled fast reactor with a core outlet temperature of 850°C to produce 500 MW thermal/240 MW electric, designated an advanced small modular reactor (SMR) by GA. A direct-drive thermal power conversion system (PCS) (also designated power conversion unit, PCU) with a closed-cycle gas turbine and a residual heat removal system is shown schematically in Figure 3-1 [47]. It includes one chamber and connected to the reactor through a chamber that includes the direct reactor auxiliary cooling system (DRACS) and fission product vent system (FPVS), constituting the primary coolant system, to achieve 53% net thermal efficiency [47].

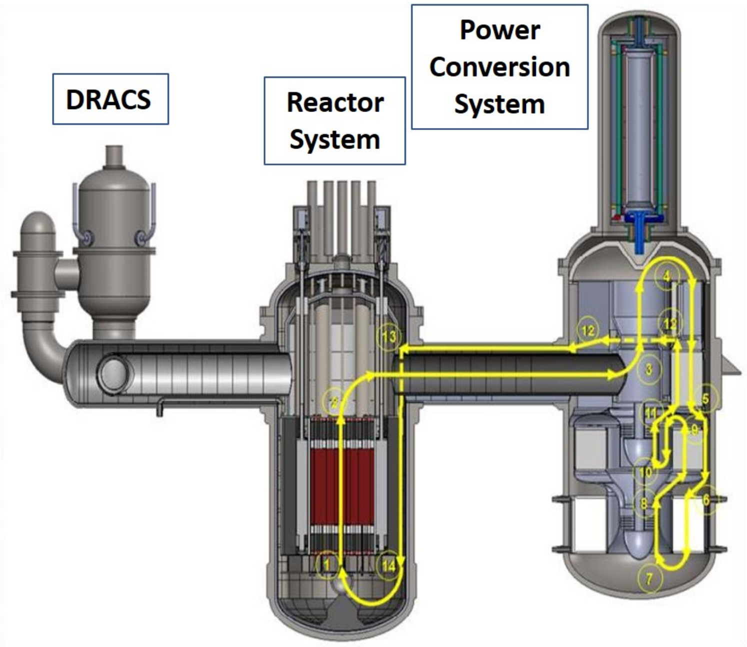


Figure 3-1
Primary coolant flow paths for EM^2 during normal operation [47]

The hot helium from the core flows through an inner concentric duct in the cross duct to the PCS, and then pressurized helium flows back through the outer cross duct annulus and into the reactor [48]. For the first-generation plant, the fuel consists of about 22.2 t of LEU starter and about 20.4 t of used nuclear fuel, which is roughly 1% U^{235}-1% Pu, mixed actinide (MA) and 3% fission products, with the rest being U^{238}. GA calls it a “convert and burn” core design because it converts fertile isotopes to fissile and “burns” them in situ over a 30-year core life. General Atomic claims there is no need for uranium enrichment after the first-generation reactor, since the discharge from the first-generation reactor is used for the succeeding generation, as shown in Figure 3-2 [46]. This is in contrast with the 2014 GIF roadmap [18] that proposed a 2400 MWth design because the 600 MWth reactor in the original roadmap could not meet the breakeven-breeding requirement.

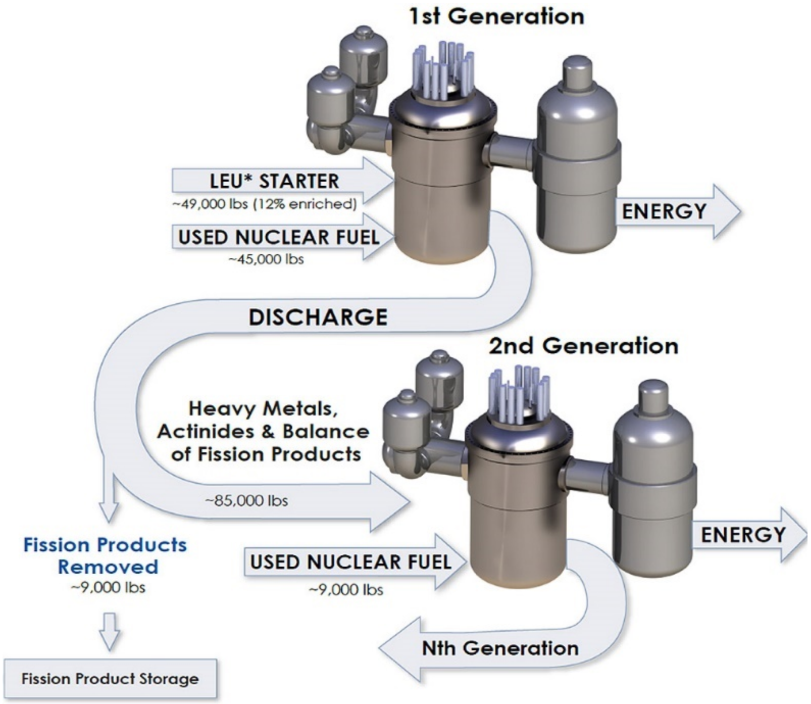


Figure 3-2
Schematic diagram illustrating the initial fuel charge, production of fuel and fission product removal in succeeding generations [46]

Out of each discharge, about 38.5 t is used in the succeeding generation while about 4 t of fission products are removed [46]. Interestingly, the entire containment is designed to be below grade and sealed for the 30-year core period. The reactor design is based on a uranium carbide (UC) annular fuel pellet, silicon carbide (SiC) cladding and structure, and a zirconium silicide (Zr_3Si_2) reflector [48]. The core incorporates SiC-SiC composite material for resistance to very high temperatures that could be attained under accident conditions and high irradiation doses. A schematic depiction of the core construction is shown in Figure 3-3 from Choi and Schleicher [49], showing the zirconium silicide and graphite reflectors. The different colors indicate different depletion materials due to fission gases. The reflector consists of an inner section of SiC-SiC–enclosed Zr_3Si_2 blocks and an outer section of graphite blocks [49].

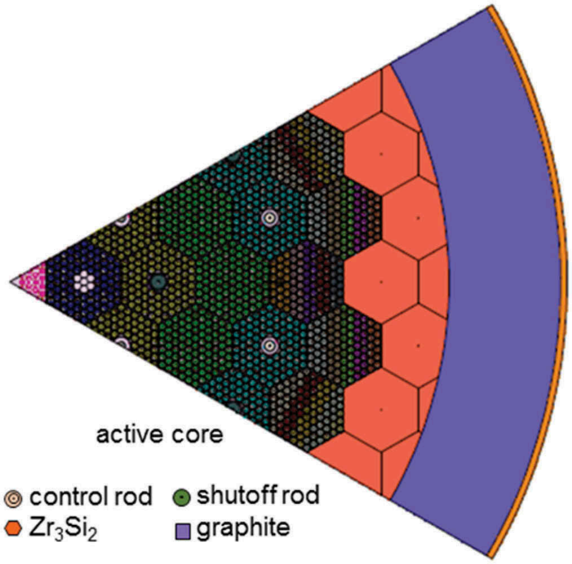


Figure 3-3
Schematic diagram of EM^2 active core [49]

All structural components are made of SiC-SiC composite. The reactor core characteristics are provided in Table 3-1 [48]. A demonstration reactor with about 1/4th the power of the full-scale reactor will be constructed and operated to obtain core physics, fuel integrity, and safety system performance data to support the EM^2 design certification; moreover, the design work will include experimental tests of in-core materials under high temperature and high irradiation [48]. Choi, et al. [48] specifically noted that the EM^2 design relies on non-conventional nuclear materials and GA is performing high-risk R&D programs that include characterization of structural materials under high neutron irradiation, with some irradiation experiments already performed in the High-Flux Isotope Reactor (HFIR) at ORNL and in the Breazeale Nuclear Reactor of Pennsylvania State University (PSU); also, fuel irradiation tests are underway in Norway's Halden research center. Schleicher and Bertch [47] stated that: "because there is no precedent for the EM^2 core and PCU designs, GA believes that a one-unit prototype plant is required to reduce the technical and licensing risk to an acceptable level before embarking upon a commercial plant. This would accomplish identifying and resolving unforeseen problems, providing a test basis for retiring the residual Phase 1 risks, providing the bases for qualifying the fuel for long life, and providing the bases for a 10CFR52 Design Certification." In the later paper, Choi, et al. [48] stated that the demonstration reactor would be licensed under 10CFR50.

Figure 3-4 from Schleicher and Bertch [47] shows a cutaway view of the EM^2 reactor system showing the flow path of helium in the reactor system with the various components in the reactor system. Schleicher and Bertch note that the RPV and the cross-duct vessels are constructed from SA-533, Grade B steel and are internally insulated. As discussed in Section 2 for the two VHTR systems, the use of RPV steels typical of those used in LWRs, imply that the RPV and cross duct vessels will be cooled to enable operation around 300-325°C.

Table 3-1
Comparison of EM^2 core characteristics [48]

	Demo	Full-Scale
Reactor power (MWth)	120	500
Core volume (m^3)	1.8	8.6
Core inlet temperature (°C)	550	550
Core outlet temperature (°C)	850	850
Coolant pressure (MPa)	13.3	13.3
Average power density (W/cm^3)	66.6	58.3
Peak linear power (kW/m)	78.8	71
Uranium loading (t)	8.6	41.1
Average fissile content (wt.%)	10.5	7.68

For the PCU vessel shown in Figure 3-5 from Schleicher and Bertcsh [47], all components are in contact with the primary helium, and include two heat exchangers (HX), a recuperator and a pre-cooler. The recuperator is a helium-to-helium heat exchanger constructed from Inconel 617, while the precooler is a helium-to-water HX constructed from 2.25Cr-1Mo steel in a diffusion bonded configuration [47]. The turbo-compressor system is mounted with a stiff cartridge frame, also shown in Figure 3-4, which allows for removal for repairs or even replacement. The simpler direct-drive system also means reduced construction costs, since elements such as the steam generators, the main steam system and the feedwater-condensate system are eliminated.

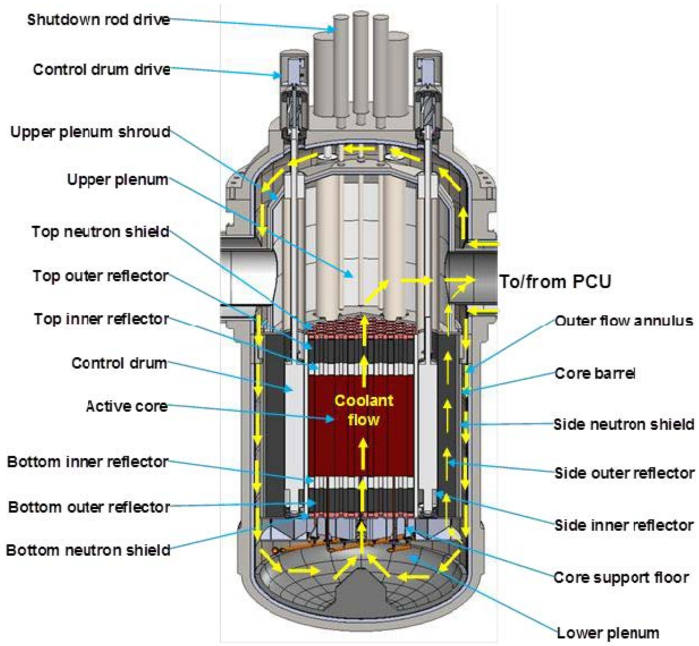


Figure 3-4
Cutaway of EM^2 reactor system showing helium flow path and system components [47]

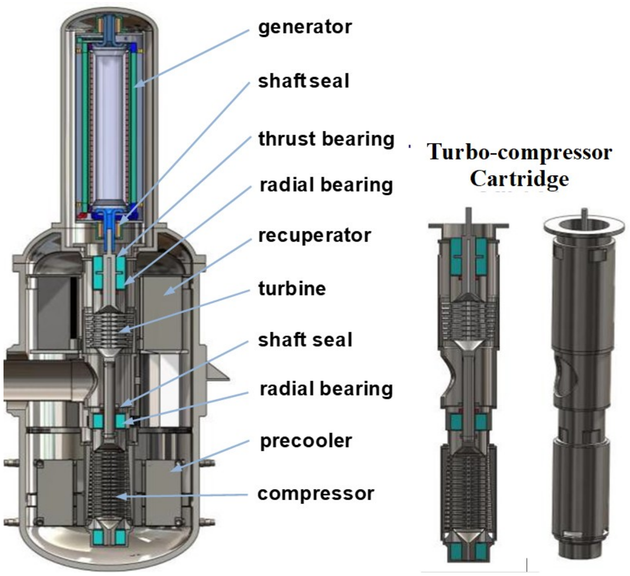


Figure 3-5
Principal elements of the EM^2 power conversion unit [47]

3.2 Components and Sub-Components Issues

Table 3-2 is constructed from one in Corwin, et al. [17] that provided parameters for three different GFR concepts, including the helium direct concept (for example, the EM^2) shown in the table. The temperatures and radiation doses in the table are based on the GEN IV reference design in 2005, but the core characteristics shown in Table 3-1 for the full scale EM^2 reactor are very similar to those for the GEN IV reference GFR in an INL publication by Weaver [19]. Both the Corwin, et al. and Weaver publications emphasize that the ~600 MW_t design enables "modular" design (that is, small vessel, small core, and so on), that can be transported to the site, and can utilize the VHTR balance-of-plant development, thus, minimizing R&D costs. Relative to the VHTR design, the power density in the GFR is about two orders of magnitude greater, the coolant density coefficient adds reactivity during re-pressurization accidents, and there are no large blocks of graphite to provide thermal inertia to slow rapid heat-up of the core following loss of forced cooling. The design incorporates a decay heat removal system GA designates as the DRACS which is shown in Figure 3-1 (two DRACS are actually incorporated in the system) [49]. Thus, materials must be utilized and demonstrated to be acceptable regarding neutronics and performance at the high temperatures and very high levels of neutron exposure. This need for development and alternate materials is also emphasized in the GIF roadmap [18]. Of course, all materials must be qualified, including those for fuel and cladding, supporting structures, 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. The anticipated radiation levels are not known for the EM^2 reactor, but such estimates will be critical for the ultimate material selections for many of the components in the reactor system.

As noted previously, the temperatures and radiation doses for the GFR are much higher than those for the VHTR and present significant challenges. It is important to note that the design for this reactor is, understandably, very much evolving; for example, the reflector was initially an inner section of canned Be_2C and outer section of graphite but has been changed to an inner section of SiC-SiC–enclosed Zr_3Si_2 blocks and an outer section of graphite blocks [49]. Table 3-2 was taken from Corwin, et al. [17] but some of the temperatures have been changed based on available information for the EM^2 reactor. The changed values are noted in bold italics. Neither the radiation doses nor the off-normal temperature have been changed and are used here as approximations since they were provided for the GFR reference design in 2005. Moreover, the EM^2 reactor design has made advancements in design and designation of materials of construction that likely significantly affect those estimates.

The fuel cladding is 21 mm in diameter and 1 mm thick, with multiple fuel rods assembled into a hexagonal assembly and with three assemblies making up a tri-bundle. The cladding is made of SiC-SiC, which is highly resistant to both temperature excursions and neutron damage, with the maximum fast neutron fluence stated as 1.07×10^{24} n/cm^2 [49,50]. The peak fuel temperature is estimated to be 1400°C, which is below the sintering temperature (1600°C to 2000°C) and far below the melting point of the cladding [49]. The average cladding temperature is ~1000°C around which the swelling of SiC is minimal [51]. In the case of accident (off-normal) conditions, analysis results show that the peak fuel temperature remains below 1920°C during reactivity-induced accidents [49]. Compared with the values in Table 3-2, the average cladding temperature is lower and the off-normal temperature is higher. Weaver [19] provided specific requirements for the GFR fuel matrix and structural materials as shown in Table 3-3, although the derivation of those values is not clear. He also stated that only ceramics can meet those reference values.

Table 3-2
Operating conditions for the reference GEN IV GFR vessel, core, and internals* [17]

Component	Normal Conditions		Off-Normal Conditions	Notes for EM^2 Reactor
	Temperature	Peak Dose	Temperature	
Fuel Matrix- Cladding	1200°C	15-20 dpa/yr, total 60 dpa	Up to 1800°C	*Uranium carbide (UC) annular fuel pellet, β-Silicon carbide cladding. Clad sees 1920°C for 10 min. and cools rapidly.*
Spacers/Wire Wrap	*550*-1000°C	15-20 dpa/yr, total 60 dpa	Up to 1600°C	
Fuel Subassembly	*550*-1000°C	15-20 dpa/yr, total 60 dpa	Up to 1600°C	*SiC-SiC composite*
Fuel Subassembly Duct	*550*-1000°C	15-20 dpa/yr, total 60 dpa	Up to 1600°C	*SiC-SiC composite*
Reflector	*550*-850°C	Up to 150 dpa	Up to 1100°C	*Zr_3Si_2 and Graphite- Normal operating temperatures are conservative; the high end may be less.*
Control Rod Guide	*550*-1000°C	Up to 200 dpa	Up to 1600°C	*SiC-SiC composite*
Upper Support Plate	850°C	Up to 100 dpa	Up to 1200°C	Normal operating temperatures assume the gas is well mixed at the core exit.
Lower Support Plate	*550°C*	Up to 100 dpa	Up to 750°C	
Core Barrel	550-850°C	80-100 dpa	Up to 1100°C	
Pressure Vessel	<371°C	< 1 dpa to 40 dpa	Up to 1100°C	*SA-533 Grade B Class 1 plate -Doses for that material must be much lower than shown as discussed in Section 3.4.*
Fuel Matrix- Cladding	1200°C	15-20 dpa/yr, total 60 dpa	Up to 1800°C	*Uranium carbide (UC) annular fuel pellet, β-Silicon carbide cladding. Clad sees 1920°C for 10 min. and cools rapidly.*
Heat Exchangers	*Up to 850°C*			*Assume Inconel 617*

Table 3-2 (continued)
Operating conditions for the reference GEN IV GFR vessel, core, and internals* [17]

Component	Normal Conditions		Off-Normal Conditions	Notes for EM^2 Reactor
	Temperature	Peak Dose	Temperature	
Cross Duct Vessel	*<371°C*			*Assume SA-533 Grade B Class 1 plate or SA-503 Class 3*
Recuperator	*850°C*			*He-to-He HX - Inconel 617*
Precooler	*<650°C*			*He-to-H_2O HX - 2.25Cr-1Mo*

* Note that all items in bold italics have been changed from the original with estimates based on inlet/outlet temps of 550/850°C, RPV and assume Cross vessel of SA-533 steel.

A well-known graph of temperature-irradiation dose operating windows for various structural materials by Zinkle and Busby [52] is shown in Figure 3-6. These authors emphasize the rather limited operating temperature regimes where such materials can be utilized in a neutron environment. At low temperatures radiation hardening causes reduced ductility and creates conditions requiring larger safety margins in design, but it also causes reduced fracture toughness in some cases which restricts applications to higher temperatures. For the high temperature regime, decreases in thermal creep strength or high temperature helium embrittlement are limitations. The authors point to a key strategy for designing high-performance radiation-resistant materials, that are associated with introduction of a high, uniform density of nanoscale particles in the microstructure. Such microstructural features serve to enhance strength by hindering dislocation motion and provide point defect recombination centers to provide good radiation damage resistance [52]. Thus, the entire operating window is important for application not just the upper temperature limit.

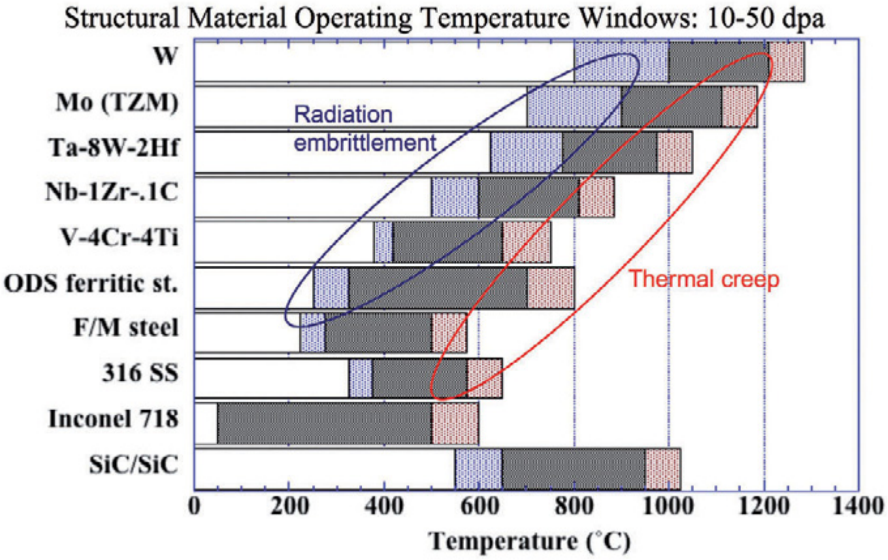


Figure 3-6
Structural Material Temperature and Irradiation Dose Operating Windows (dark shading). The blue and red regions represent uncertainty bands [52].

Table 3-3
GFR fuel matrix and structural material reference requirements [19]

Requirement	Reference Value
Melting/decomposition temperature	>2000°C
Radiation induced swelling	< 2% over service life
Fracture toughness	> 12 MPa $m^{1/2}$
Thermal conductivity	> 10 W/mK
Neutronic properties	Materials allow low core heavy metal inventory and maintain good safety parameters

3.3 Candidate Materials for Core Components and Reactor Internals

The GIF update report includes R&D objectives which highlight the need for an experimental fast spectrum reactor, but also notes the need for fuel development and out-of-pile experimental facilities for qualification of the main systems [18]. Within the latter item are listed the need for innovative materials and various ceramic materials (UC, PuC, SiC, ZrC, Zr_3Si_2), and development and qualification of components, including specific blowers and turbomachines, thermal barriers, valves and check-valves, and instrumentation. That document also describes the ALLEGRO experimental reactor (Figure 1-3) project that will be the test bed to develop and qualify high temperature, high-power density fuel required for a full scale GFR. Although the currently designated outlet temperature of the ALLEGRO is not sufficient to evaluate most structural materials, it is foreseen that the initial MOX core would be replaced with a ceramic core which would allow for a helium outlet temperature of 850°C.

3.3.1 Core Components

As mentioned earlier, Weaver [19] stated that only ceramics can meet the material reference requirements shown in Table 3-3. In that report he listed SiC, ZrC, TiC, ZrN and TiN as likely candidates. From Choi and Schleicher [48], the fuel cladding and all core structural components of the EM^2 reactor will be constructed with SiC-SiC composites. For the EM^2 reactor, Choi and Schleicher [49] explicitly state: “The core uses all-ceramic materials (that is, UC, SiC-SiC, C-C, and so on) for the fuel, core structures, control and shutdown rods, reflector, upper plenum thermal barrier, internal vessel insulation cover plates, and hot duct assemblies, which are resistant to high-temperature operation.” They also state that mechanical load of the cladding due to pellet-cladding mechanical interaction (PCMI) is minimal owing to fission gas venting [48]. In the same document, they claim that the SiC-SiC maintains its strength up to 1700°C and retains greater than 75% of its strength at 2000°C. That statement is not referenced but Figure 3-7 from Snead, et al. [51] shows that the strength of SiC vs. temperature (K) is quite dependent on the type of SiC. Furthermore, for the Hot-pressed/HIP version, the strength retention at slightly over 1750K (~1500°C) has retained about 60% of its room temperature strength. On the other hand, it also shows a sintered/CVD version actually gained about 20% strength at the same high temperature. Thus, given the peak temperature of 1920°C, very high temperature strength needs to be confirmed for the specific SiC specified. From the same Handbook, a plot of fracture toughness vs temperature for various SiC materials (non-irradiated) shows that the fracture toughness increases with test temperature to about 1500°C, but the highest value measured was less than 6 MPa√m, about half the value specified by Weaver. The loading conditions on the various components are not known, thus, it is not clear if such low fracture toughness values present an issue that needs to be resolved for the SiC composites. A potential issue regarding diffusion of fission products through the clad or through micro-cracks is noted by Vasile with the recommendation to use an internal refractory metal liner (W-Re) for prevention [12].

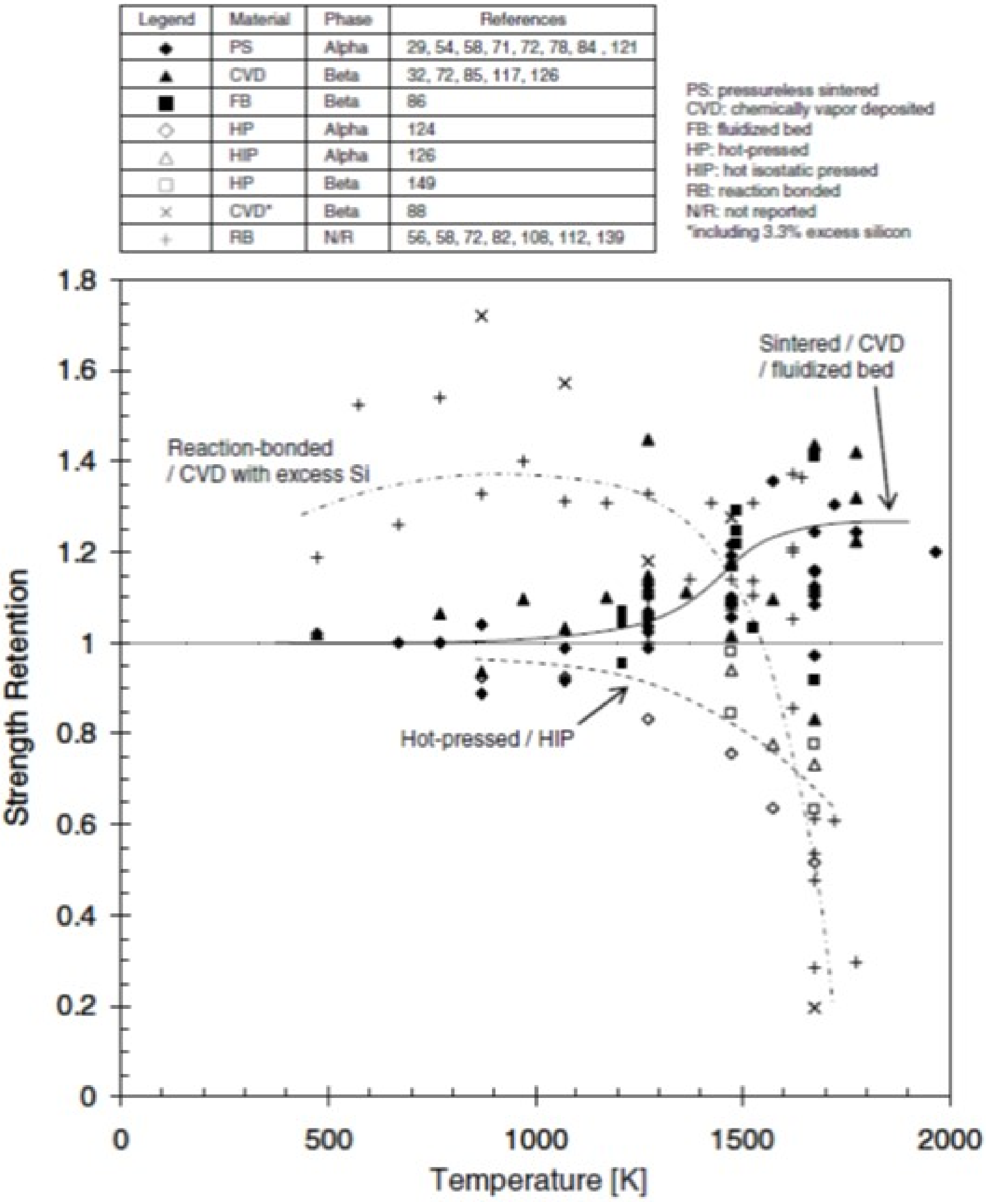


Figure 3-7
Retention strength vs temperature of SiC [51]

Choi and Schleicher [49] report that SiC-SiC composite and hybrid preceramic polymer, chemical vapor-infiltrated (H-SiC) joint components have been irradiated to 4.5 dpa at 700°C in the HFIR and show virtually no change in structure with irradiation. They recognize that 4.5 dpa is far less than the expected EM^2 lifetime value of ~350 dpa but point out that it is past the 1 to 2 dpa point in which changes in irradiated properties appear to saturate, citing Katoh [53]. In the cited paper by Katoh, details of physical, thermal, and mechanical properties of the Hi-Nicalon Type S, CVI SiC-matrix composite following neutron irradiation in HFIR to doses as high as ~74 dpa and various temperatures to as high as 1073 K (800°C) are reported. Figure 3-8 shows the test results for relative ultimate flexural strength vs neutron dose (dpa), indicating a decrease of strength with increasing dose, but less decreases with increasing irradiation temperature. The figure also provides results for relative Young's Modulus from the same experiment, including some data from the literature, indicating only slight decreases in that property when irradiated at 800°C. The authors did note that the materials tested, "represent the very first generation of the nuclear-grade SiC composite that had not necessarily been optimized for balanced properties," implying that recent developments in nuclear grade SiC composites will exhibit even less degradation to irradiation. In the same experiment, swelling of the material was also measured and exhibited only about 1% at 800°C and ~74 dpa.

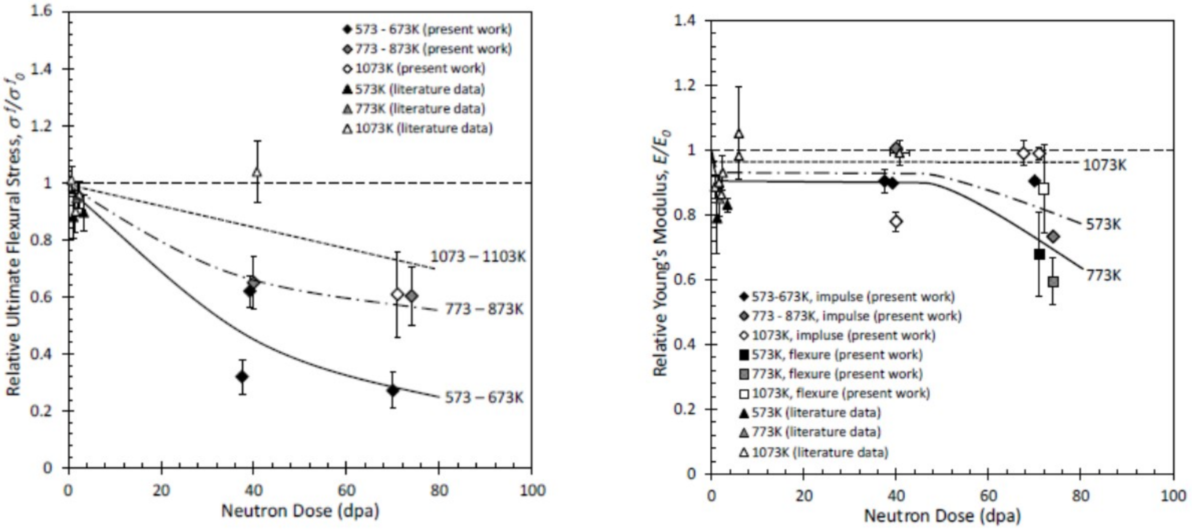


Figure 3-8
Effects of irradiation and temperature on flexural strength and Young's Modulus of Hi-Nicalon Type S, CVI SiC composite [53]

In spite of the encouraging observations from various experiments on SiC at relatively high temperatures and high irradiation doses, the extreme conditions likely to be imposed on some of the SiC-SiC components in the GFR far exceed those results, and further experiments are needed to provide confidence for a 60-year design lifetime. This is not a unique observation, as similarly statements were made about "high temperature resistance silicon carbide cladding" in a report for GFR activities in Europe [54].

General Atomics has also fabricated Zr_3Si_2 reflector samples and conducted irradiation testing in the Breazeale Nuclear Reactor of the Pennsylvania State University. The low irradiation test reached fast neutron fluence of 3.8×10^{16} n/cm^2 without failure (this fluence is many orders of magnitude lower than reflector samples will see as shown in Table 3-2, but the specific doses have not been stated). Thus, continuing activities to obtain irradiation effects data for the Zr_3Si_2 reflector material are needed.

3.3.2 Out of Core Reactor Internals

The normal operating temperatures for the three primary out of core internals, lower and upper support plates, and core barrel, are in the range of 490 to 850°C according to Table 3-2. The EM^2 design for out of core applications, metallic alloys T122, 800H, and certain oxide dispersion strengthened (ODS) have been noted, but very high radiation exposures, even for out of core components in the GFR, are problematic. However, Buckthorpe [13] notes that considerations for the GFR would also include Cr-Ni based austenitic steels (for example, M316, CW 316 Ti, CW-15-15Ti), high-nickel alloy steels (for example, PE16) and some other ferritic-martensitic and ODS steels (for example, EM12, FV548, W.Nr.1.4914) as well as many of the ceramic materials mentioned by Weaver [19] and possibly some vanadium alloys.

3.3.3 Lower Support Plate

The lower support will see lower temperatures (~750°C maximum, for limited time) than the other two components, so a low swelling stainless steel (for example, type 316) already in the ASME Code is a potential candidate material. Also, the stainless steels mentioned above by Buckthorpe [13] are potential candidates, in particular the 15/15Ti that has exhibited acceptable deformation limits at and slightly above 100 dpa, [55] and the high temperature multi-stabilized and ultrafine precipitation strengthened (HT-UPS) steel with only about 15% decrease in yield strength from room temperature to 700°C [56]. The Type 316FR stainless steel discussed in Section 2.3 and with creep-rupture data shown in Figure 2-6 appears also to be a good candidate based on improved high temperature creep rupture properties at 600°C. However, if 100 dpa is an accurate estimate of the radiation dose, swelling and loss of ductility may be potential limitations, depending on the loading conditions. The ferritic-martensitic (F-M) steels with 9 to 12% Cr or even oxide dispersion strengthened (ODS) steels are certainly potential candidates simply due to their lower thermal expansion, higher thermal conductivity, and improved void swelling resistance under fast neutron irradiation [57].

There are ongoing developments with F-M steels (and others) through application of computational thermodynamics and the use of thermo-mechanical treatment (TMT) that show promise to significantly increase strength and ductility at high temperatures, for example, 650°C as shown in Figure 3-9. The figure compares two experimental F-M steels (9Cr-2WMnV) with a current F-M steel, normalized and tempered 9Cr-2W (Grade 92) [58]. Heat treatment conditions include both standard normalized and tempered and hot-roll TMT conditions. The ASME Code Section III, Division 5 includes Grade 91 (9Cr-1MoV) steel to 650°C, but advanced steels such as those in Figure 3-9 represent the kinds of future advances that will enable even higher temperatures by taking advantage of the favorable aspects offered by F-M steels in high irradiation environments. All of these advanced alloys that offer high temperature tensile properties, creep strength, and swelling resistance, must also be evaluated for fatigue and creep-fatigue behavior under the conditions anticipated. Creep-fatigue is a significant issue for

Grade 91 and other F-M steels and is accounted for in Section III, Division 5 of the ASME Code. Recent changes in the Code have increased the creep-fatigue design window and decreased the stress relaxation conservatism for Grade 91 steel. Thus, knowledge of the anticipated loading conditions for each component is needed to optimally deploy such materials.

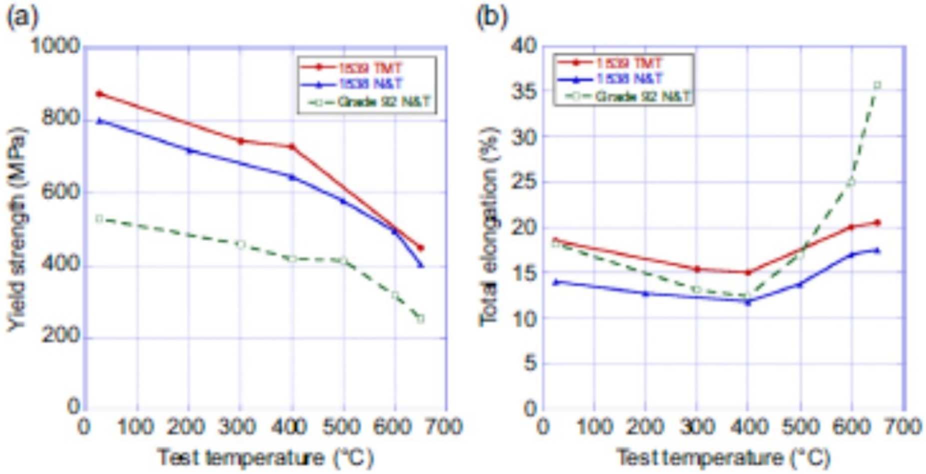


Figure 3-9
Comparison of (a) yield strength and (b) total elongation vs test temperature for two experimental F-M steels compared with Grade 92 N&T steel [58]

3.3.4 Upper Support Plate and Core Barrel

The upper support plate and core barrel will see about the same temperature and irradiation conditions, with normal operation at 850°C, short-term off-normal conditions as high as 1100 to 1200°C, and irradiation doses as high as 100 dpa. These conditions represent very major challenges for any structural materials. For normal service at temperatures above 800°C and at high irradiation doses, Figure 3-6 shows potential metallic candidates as Nb-, Ta-, Mo-, and W-based alloys, as well as SiC-SiC composites. Figure 3-10 from Zinkle, et al. [59] shows various high nickel and nickel-base alloys with reasonably good yield strength and creep rupture strength at temperatures to at least 1000°C, but the figure does not account for radiation exposure. In that case, as those authors noted, Alloy 800H and the general class of high nickel (including Alloy 617) and nickel-based superalloys suffer from neutron embrittlement at very low doses. The authors contrast this behavior with the exceptional irradiation stability of SiC composites, concluding that SiC composites would offer the potential for lifetime components as opposed to the need for expensive replacements using metallic components. Additionally, C-C composites are often considered for very high temperature applications due to their high mechanical and thermal properties. Experimental data have shown that C-C composites can withstand low neutron doses, but they have limited dimensional stability at high doses [14, 60], in fact, somewhat lower than graphite [61].

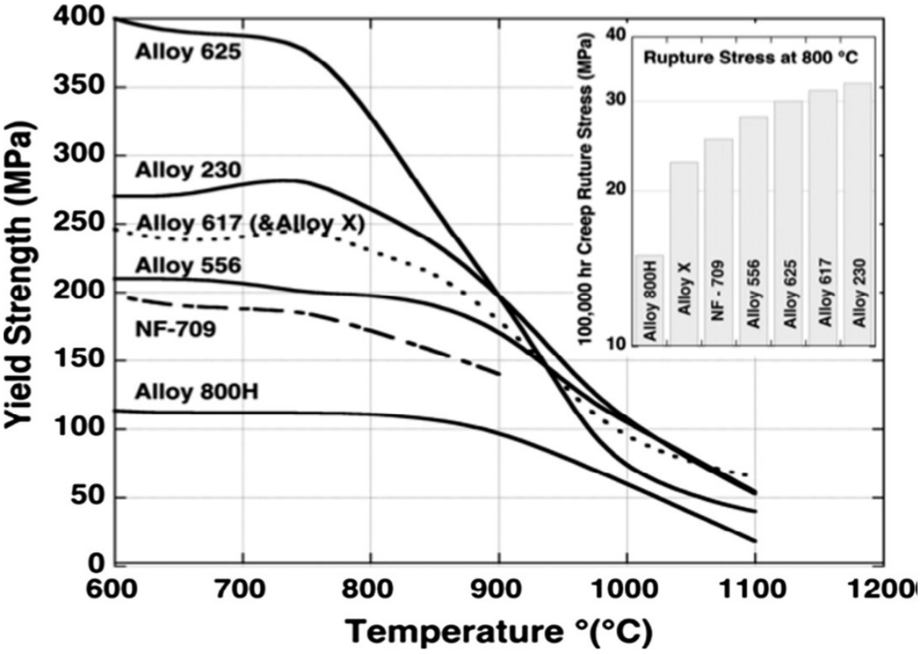


Figure 3-10
Yield strength and 100,000 h rupture strength of candidate in-core gas-cooled reactor structural alloys and vendor-supplied property data [59]

The class of refractory alloys have definitive advantages over many of the other metallic alloy systems discussed, with high melting temperatures, excellent high-temperature strength, generally good thermal creep resistance, and high void swelling resistance relative to face-centered-cubic (fcc) candidate materials [62]. Figure 3-11 [62] shows the excellent ultimate strengths exhibited by many common refractory alloys, while similar graphs of creep properties (not included here) also show quite good creep response in the 1000 to 1300°C regime.

However, refractory alloys have significant feasibility issues associated with their use in Generation IV nuclear reactors, such as low-temperature radiation embrittlement even at low dose, poor oxidation resistance, and fabrication and joining difficulty. Relative to irradiation response, irradiation below about 30% of the melting point results in significant embrittlement [63]. Thus, this limits application of these materials when the operating temperature is below ~600 to 900°C. The actual temperatures for the EM^2 components are not known here, but it appears the core barrel, for example, is exposed to inlet helium at 550°C and outlet helium at 850°C. As noted by Zinkle and Wiffen in 2004 [63], there were large uncertainties in the allowable temperature window for high temperature refractory alloys due to a lack of mechanical properties data (fracture toughness, helium embrittlement of grain boundaries) on irradiated material. Since that statement, the available irradiation data on refractory alloys has not substantially increased. Figure 3-12 from Muroga provides the recommended design stress operating windows for the indicated refractory alloys as proposed by El-Genk and Tournier [64] for all but the V-4Cr-4Ti from Zinkle [65].

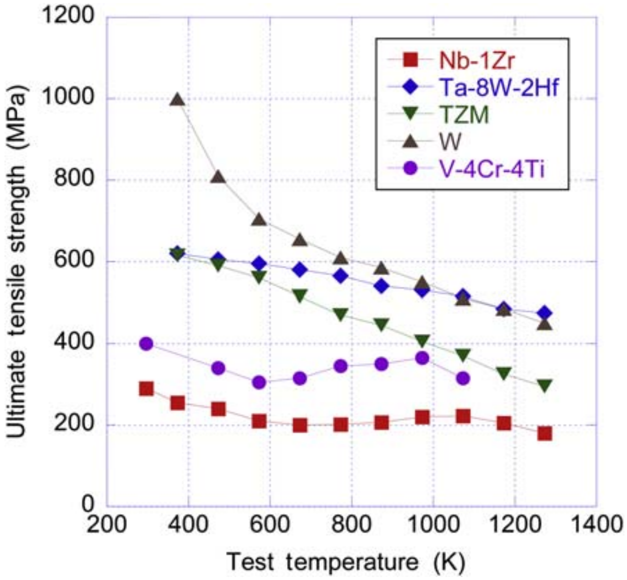


Figure 3-11
Ultimate strengths for common refractory alloys in non-irradiated condition [62]

In addition to sensitivity to oxidation, Muroga [62] emphasizes that refractory alloys are also subject to corrosion and degradation of mechanical properties from impurity contamination and states: "..unrealistic impurity control requirements make the use of refractory alloys in gas or water coolants almost unfeasible, unless an anticorrosion coating is developed." Corwin, et al. note that without some type of protection of the metal, oxygen content in the system must be maintained well below ~1ppm [17]. However, although there have not been extensive developments in recent years, some limited work indicates there are opportunities for improvements in those negative areas through various metallurgical treatments, for example, nanoparticle dispersion (for example, ODS), introduction of internal impurity getters, and so on. Experiments with various Mo variations (ODS and TiC- containing alloys) at temperatures to ~1050°C and to 12 dpa have shown substantial decreases in radiation hardening [62]. Given the need to withstand very high neutron exposures in the GFR, there is a definitive need for irradiation effects information at substantially higher doses.

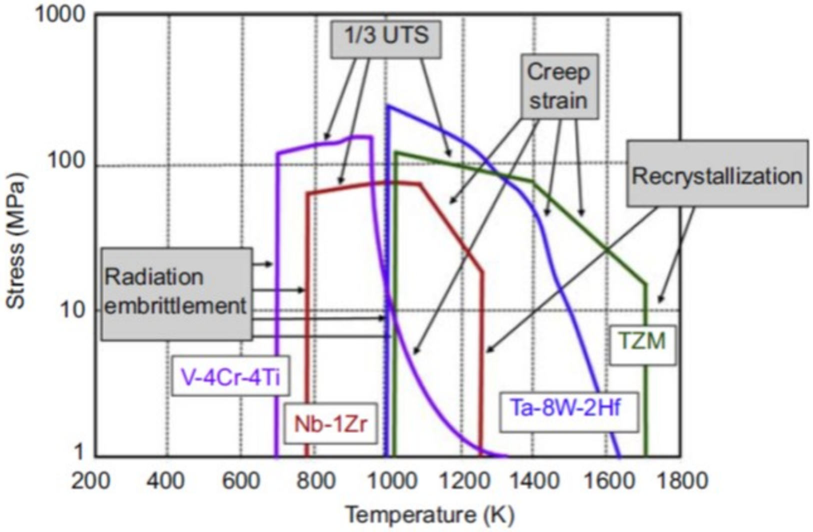


Figure 3-12
Design stress domains for indicated refractory alloys [62]

3.4 Candidate Materials for Reactor Pressure Vessel and Cross-Duct Vessel

For the RPV of the GIF reference design, Vasile [12] lists the need for very long term (60 year) creep and creep-rupture properties operating in the 400-550°C range and up to 100 dpa because of estimates of high temperature and high radiation exposures to the RPV. However, as noted in Section 3.1, the RPV and the cross-duct vessel for the EM^2 reactor will be constructed from SA-533, Grade B steel, as used for LWR RPVs, and will be internally insulated, implying that the RPV and cross duct vessels will be cooled to enable operation around 300-325°C. Given that information, the reader is referred to Section 2.4 of this report that discusses the RPV material issues for the VHTR. Because the internal irradiation levels for the GFR are so much higher than for the VHTR, it is necessary to know the anticipated fast neutron fluence for the GFR RPV to enable evaluation of the irradiation effects.

The RPV is 4.7 m diam. x 10.6 m high, and of a size that can be manufactured by many vendors and can be shipped by truck to the site [65]. Furthermore, it is internally insulated with silica/alumina fibrous insulation and retained with C-C cover plates, while the upper plenum contains a C-C heat shield that protects the top head elements from the hot helium [48]. Also, the bottom of the reactor vessel is ceramic lined so that it would take at least 12 h for vessel failure [49].

Since the cross vessel is also to be fabricated from the same SA-533 materials as the RPV, the materials acceptability issues are the same as those for similar components in the VHTR, those being emissivity and thermal aging as discussed in Section 2.5.

3.5 Candidate Materials for Power Conversion Unit Components

As discussed earlier in Section 3.1, all components in the PCU are in contact with the primary helium that enters at 850°C. The components of the PCU are not exposed to high radiation as are those in the reactor system, so mechanical properties at temperature and helium compatibility are the primary issues. GA has specified that the recuperator (helium-to-helium heat exchanger) will be constructed of Inconel 617 (also designated Alloy 617), while the precooler (helium-to-water heat exchanger) will be constructed from 2.25-1Mo steel in a diffusion bonded configuration [47]. Since the two heat exchangers will also operate at 850°C, Alloy 617 is a candidate material for those components as well. Alloy 617 is a well-known high-nickel (and relatively high chrome) material with substantial experience in high temperature applications. As shown in Figure 3-10, Alloy 617 maintains good tensile yield strength and creep strength at 850°C. Code Cases N-872 and N-898 were recently approved for use of Alloy 617 (UNS N06617) for both low temperature (to 425°C) and high temperature (954°C) service construction, respectively, for application within Section III, Division 5 of the ASME BPV Code [83, 29].

Corwin, et al. [14] provide discussion of Alloy 617 and point out that the high nickel and chromium contents provide the alloy with high resistance to a variety of reducing and oxidizing environments. Noting that this is particularly due to the high aluminum content which is in conjunction with chromium, offers oxidation resistance at high temperatures. In addition, the aluminum also forms intermetallic compound γ' over a range of temperatures, which results in precipitation strengthening in addition to the solid solution strengthening imparted by the cobalt and molybdenum alloying additions. Alloy 617 has good cyclic oxidation and carburization resistance, good weldability, lower thermal expansion than most austenitic stainless steels, and high thermal conductivity. Although it retains good toughness after long-term exposure at elevated temperatures, there are data showing that the room-temperature Charpy-V notch CVN energy dropped significantly with thermal exposure temperature and time at 800ºC and 900ºC for 10,000 h.

Another possible material for the recuperator may be Hastelloy XR. Hastelloy X is a well-known high nickel Cr-Mo-Fe superalloy often considered as a candidate for high temperature applications as it has mechanical properties similar to those of Alloy 617. The XR version was developed in Japan [67] because the X version does not have sufficient compatibility with the primary helium coolant. In fact, for the heat exchanger for the HTTR, the outer and inner shells are constructed of 2.25Cr-1Mo, while the heat transfer tube, hot header and center pipe are constructed of Hastelloy XR for operation in hot helium at ~950°C. The advantage of Alloy 617 is its current inclusion in the ASME Code, including an impending increase in maximum allowed temperature. In the JAEA report [67], the work accomplished was funded at the behest of ASME Standards Technology for their project related to the former Nuclear Code Case NH, now incorporated in Section III, Division 5 (Hastelloy is not currently included). Additional information regarding helium compatibility for Hastelloy XR is discussed in Section 4.

The choice of 2.25Cr-1Mo for the precooler indicates that component will operate at a much lower temperature than the recuperator, although it was not defined. The maximum temperature for 2.25Cr-1Mo in Section III, Division 5 is 650°C. Additional information regarding operating conditions for that component and other components in the PCU are needed for a comprehensive assessment. It is notable that the PCU has been designed with a turbo-compressor cartridge that can be removed for repairs or even replaced. They further note that the simpler direct-drive system also means reduced construction costs, since elements such as the steam generators, the main steam system and the feedwater-condensate system are eliminated.

Given that the design of the RPV is for an insulated component to operate at about 300°C as for LWR RPVs with SA-533 Grade B steel, the same choice of material for the PCU vessel is a likely scenario. If the PCU will not be insulated as such, then higher temperature materials would be required such as F-M steels (for example, Grade 91, and so on), dependent on the operating temperature.

The GIF Roadmap (2014) discussed that internal core structures and cooling systems, such as intermediate heat exchangers, hot gas ducts, process components and isolation valves, that are in contact with hot helium, can, as mentioned earlier for the VHTR, use current metallic materials up to a core outlet temperature of about 700 to 800°C. They further note that efforts in the United States have focused on developing the data needed to extend Alloy 800H allowance for use up to 850°C and Alloy 617 for use up to 950°C. Within the next four years, all of the data needed to codify these materials will be provided to the ASME. Thus, portions of the GFR PCU that are not exposed to neutron radiation but may operate at those higher temperatures may be able to take advantage of such materials with ASME Code approval.

3.6 Candidate Materials for Graphite Components

Because graphite is an excellent moderator, the fast-spectrum concept of the GFR limits the use of graphite and the only graphite material outside of the fuel is in the small blocks of graphite associated with the reflector. The size of the blocks is not known, but their presence will certainly soften the neutron spectrum at the RPV and reduce the fast neutron fluence. Because the issues associated with graphite for the VHTR were also originally noted for a VHTR outlet of 950C, they are essentially the same for the GFR, with the important exception that the neutron fluence for the small portion of graphite in the GFR reflector is much higher. The reader is referred to Section 2.6 of this report.

3.7 Summary of GFR Material Properties Literature Review

Choi, et al. [48] specifically noted that the EM^2 design relies on non-conventional nuclear materials and GA is performing high-risk R&D programs that include characterization of structural materials under high neutron irradiation, with some irradiation experiments already performed in various research reactors. Schleicher and Bertch [47] stated that: "because there is no precedent for the EM^2 core and PCU designs, GA believes that a one-unit prototype plant is required to reduce the technical and licensing risk to an acceptable level before embarking upon a commercial plant."

Choi and Schleicher [49] also noted that: "Compared with other advanced reactors, the technology readiness level of the GFR is relatively low because no experimental or commercial GFR has been built and the materials and core concepts are new. However, in order to meet the demanding economic, operational, and safety requirements of a twenty-first century power source, it is essential to implement advanced features in the new reactor design. Fortunately, the GFR can share the technologies from other advanced reactors such as the high-temperature power conversion of the very high-temperature reactor and the fast neutron physics of the liquid metal reactor, which would be the stepping stone to the demonstration and commercialization of the GFR." Additionally, they espoused that licensing and construction for the first-of-a-kind EM^2 should address the key technical challenges of the GFR and demonstrate the viability of the core physics, material performance, and safety. More specifically, in addition to research and development associated with the fuel and system control issues, they recommended specific follow-up studies for demonstration of critical components made of SiC-SiC and Zr_3Si_2 material under high temperature and high-irradiation environment.

This review has identified issues that need resolution for the GFR, either due to insufficient information to judge the issue, or issues that have been noted but for which there is insufficient materials data under the appropriate conditions of exposure. In many cases, the most prominent factor is the need for sufficient data for a 60-year design lifetime. Key specific gap findings are:

- The anticipated radiation levels are not known for the EM^2 reactor, but such estimates will be critical for the ultimate material selections for many of the components in the reactor system with current estimates to 100 dpa.
- Given the peak temperature of 1920°C for SiC-SiC core components, very high temperature strength needs to be confirmed for the specific SiC specified.
- The loading conditions on the various components are not known, thus, it is not clear if lower values of fracture toughness compared with the Handbook values present an issue to be resolved for the SiC composites.
- A potential issue needing further review regards diffusion of fission products through the clad or through micro-cracks; this has been noted in the literature with the recommendation to use an internal refractory metal liner (W-Re) for prevention.
- In spite of the encouraging observations from various experiments on SiC at relatively high temperatures and high irradiation doses, the extreme conditions likely to be imposed on some of the SiC-SiC components in the GFR far exceed the available results, and further experiments are needed to provide confidence for a 60-year design lifetime. This is not a unique observation, as similarly stated about “high temperature resistance silicon carbide cladding” in a report for GFR activities in Europe.
- Continuing experiments are needed to obtain sufficient irradiation effects data for the Zr_3Si_2 reflector material to enable reasonable prediction for a 60-year lifetime.
- Knowledge of the effects of impure helium on Zr_3Si_2 material are needed.
- Radiation effects on potential fibrous insulation as affecting degradation and thermal stability are not well known. There is very limited information available and, if proposed for application, those issues need to be reviewed.
- All of the advanced alloys that offer high temperature tensile properties, creep strength, and swelling resistance, must also be evaluated for fatigue and creep-fatigue behavior under the conditions anticipated. Creep-fatigue is a significant issue for Grade 91 and other F-M steels and is accounted for in Section III, Division 5 of the ASME Code. Recent changes in the Code have increased the creep-fatigue design window and decreased the stress relaxation conservatism for Grade 91 steel, but not for all potential alloys. Thus, knowledge of the anticipated loading conditions for each component is needed with respect to the specific material proposed.
- Additional information regarding operating conditions for the recuperator and other components in the PCU are needed for a comprehensive assessment.

- In addition to sensitivity to oxidation, refractory alloys are also subject to corrosion and degradation of mechanical properties from impurity contamination in the helium. If refractory alloys are considered, realistic impurity control of the helium must be achievable to possibly as low as 1 ppm, or experimental results demonstrating a lower sensitivity for the selected material, or demonstration of an effective anticorrosion coating.
- Also, if refractory alloys are considered, given the very high neutron exposures, there is a definitive need for irradiation effects information at substantially higher doses.
- Because the internal irradiation levels for the GFR are so much higher than for the VHTR, it is necessary to know the anticipated fast neutron fluence for the GFR RPV to enable evaluation of the irradiation effects on the RPV itself as well as on the graphite portion of the reflector.
- Since the RPV and the cross vessel are to be fabricated from SA-533 materials, the issues are the same as for the VHTR, those being emissivity and thermal aging. Also, insulation and cooling methods to keep the vessels below 371°C need to be more precisely defined and specified.

4

MATERIALS COMPATIBILITY ISSUES

In general, structural materials in a helium environment exhibit varying sensitivities to the gas, depending on the material composition, the exposure temperature, material changes due to radiation effects, exposure time, gas flow velocity, and the specific composition of the helium. For the reactor designs discussed in this chapter, the VHTR concepts will present somewhat less of a challenge due to lower coolant temperatures and significantly lower radiation exposures. Effects of the environment on some material systems have been mentioned in the previous sections and may be repeated in some manner here. As with the major environmental exposure issues of temperature and irradiation, knowledge of materials compatibility with the coolant is critical, especially for long term operation.

One item to note regarding previous experience is that the Dragon Reactor, the AVR, and the Peach Bottom Reactor were all operated successfully with a helium environment at temperatures from 734 to 950°C. As noted by Cabet and Rouillard [68], all those reactors had extensive gas cleanup systems intended to keep the total impurity levels in helium below a few hundredths of µbar (for example, 0.05µbar = 5E-9 MPa) under a total pressure of about 10 to 40 bar (1 to 4 MPa). Moreover, the cleanup system is designed to capture tritium and any fission products that may be released. For the VHTR with a massive amount of graphite, the graphite actually plays a dominant role in maintaining gas chemistry. The combination of factors such as the alloy surface composition, temperature, time of exposure and the helium relative impurity level determines the type of effects that may occur, for example, corrosive oxidation, carburization, and/or decarburization. The authors note that carburization and decarburization are particularly significant to degradation of mechanical properties, such as low-temperature embrittlement and reduced creep rupture strength, respectively. Of course, it was also noted that, although the experience is quite limited relative to a design lifetime of 60 years, there have been no problems with failure of components on the primary side of these reactors that could be associated with environmental effects. The authors also point out that, ideally, the most corrosion-resistant alloy would experience a continuous, adherent, self-healing, slow-growing passivating oxide layer. Of course, off-normal conditions create special problems due to extreme temperatures, in which the time of exposure is even more important.

Quadakkers [69] developed a “modified” chromium stability diagram that can be used to support assessments of the effects of the He environment on the surface stability of steels and nickel base alloys. The diagram graphically displays the expected instantaneous behavior of Cr-rich alloys in impure helium, with a dependence on temperature and gas chemistry. Because the gas phase is not in thermodynamic equilibrium due to high dilution of gaseous species, traditional thermodynamic potentials of oxygen and carbon cannot accurately predict the surface corrosion features. Thus, the knowledge gained about surface reactivity of Cr-rich alloys in impure helium provides the information needed to control the gas composition to avoid carburization in practice. Figure 4-1 shows the effects of carburization, decarburization, and oxidation on Alloy 617, demonstrating the more deleterious effects of carburization. Notable also is that the deleterious effects occur after only 500 hours of exposure.

It is therefore critical that coolant chemistry be controlled such that it favors passive oxidation over the more damaging corrosion processes of carburization and decarburization for the metallic materials selected for the VHTR system, especially those that will be subjected to the highest temperatures. Fortunately, given that the subject VHTRs with an outlet temperature of 750°C will most likely utilize relatively well-known structural materials, a substantial amount of experience exists with which to design and construct the systems needed to maintain impurity controls on the helium. Having said that, and as often stated in this report, design of such systems for 60 years is problematic from almost all aspects, so design that allows for component replacement where possible would be advantageous.

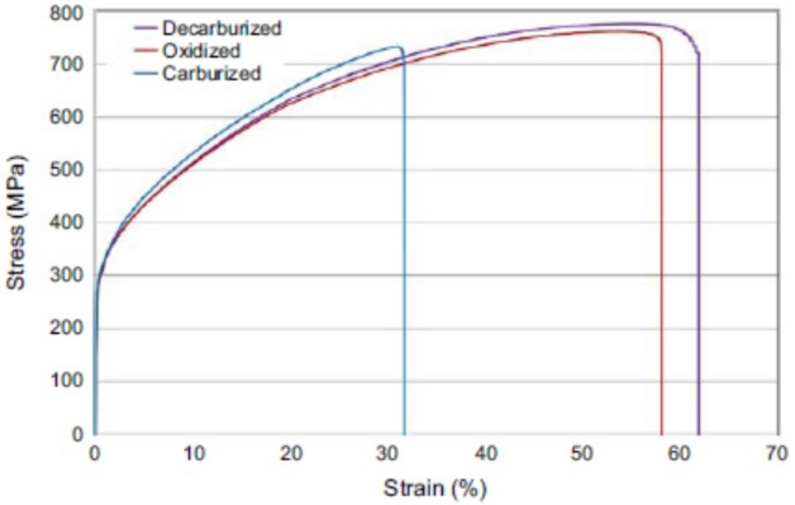


Figure 4-1
Tensile curves at room temperature for Alloy 617 exposed to different helium environments at 950°C-500h [68]

The graphite structures in the VHTR are very sensitive to oxidation; thus, maintaining low oxygen levels in the helium gas is also essential for that material. Of most interest is the effects of oxygen (O_2), carbon dioxide (CO_2), and water vapor (H_2O), as well as effects of the oxidation reactions, carbon monoxide (CO) and hydrogen (H_2). Graphite is known to oxidize in air at and above about 450°C [44]. At higher temperatures, especially above 900°C, mass transfer can occur, and the external geometry of the graphite will be most affected.

The very low amount of graphite in the GFR will likely result in a somewhat different composition for the helium. As noted by Corwin, et al. [17], with 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. Also, for the GFR core structural materials, Cabet and Rouillard [68] state that SiC-SiC composites are the material of choice (also chosen by GA for the EM^2 reactor) and demonstrate outstanding resistance at high temperatures in oxidizing atmospheres due to the properties of silicon carbide which forms a silica protective layer (passive oxidation regime).

As discussed in Section 3.3 for GFR applications, although they offer significant advantages regarding mechanical properties at high temperatures, refractory alloys are particularly sensitive to oxidation. For example, V-Cr-Ti alloys exposed to reactor grade helium with oxygen and hydrogen impurities exhibited severe embrittlement. Similarly, the other refractory alloys mentioned in Section 3.3 are notable for sensitivity to carbon and nitrogen in addition to oxygen. As stated by Muroga, [62] O and N pick-up causes serious degradation of the mechanical properties, and evaporation of oxide resulting in loss of weight. Without some type of protection, Corwin, et al. [17] note that oxygen content in the system must be maintained well below ~1ppm. On the positive side, because it is likely that the helium can be treated to reduce oxygen and moisture, it is possible that the environment in the GFR can be controlled to the desired composition.

The high temperature parts of the heat exchanger for the HTTR in Japan are constructed with Hastelloy XR, as discussed in Section 3.5. In actuality, the material is specifically designated Hastelloy XR-II [67]. This development is informative here as an example of a successful material development for application in an impure helium environment. The Hastelloy X specification was modified because it did not have sufficient compatibility with the primary helium coolant at very high temperatures. The XR version was developed by optimizing the manganese and silicon contents that enabled formation of stable and adherent oxidation films of $MnCr_2O_4$ spinel and SiO_2 on the base metal. Moreover, internal oxidation and intergranular attack are suppressed by lowering the aluminum and titanium contents. They also lowered the cobalt content to decrease cobalt-containing corrosion products in the primary cooling system to negligible levels. Although addition of boron improves the creep strength for Hastelloy XR, it causes contamination of the core and degradation in weldability; thus, they optimized the boron content by lowering it and tightening the specification to 40-60 ppm. The impurity compositions of the helium gas used in their evaluations are provided in Table 4-1.

Table 4-1
Impurity levels of simulated HTGR helium called JAERI-type B helium (in Pa) [67]

H_2	H_2O	CO	CO_2	CH_4
20 to 21	0.08 to 0.12	10 to 11	0.2 to 0.3	0.5 to 0.6

Figure 4-2 compares measurements of depth of chromium depletion for the Hastelloy X and XR materials, showing the superiority of Hastelloy XR to Hastelloy X demonstrated as expected from the protective oxide film formed on Hastelloy XR. Relative to mechanical properties at high temperatures, Figure 4-3 compares creep rupture stress vs time for Hastelloy XR-II in both air and the JAERI-type B helium environment. The results show no degradation of creep rupture properties due to helium up to 1000°C.

Similarly, for Alloy 800H, Inconel 800HT was developed by Special Metals Corporation with restricted chemical composition that results in better creep properties at elevated temperatures and without significant loss of tensile elongation [70]. That alloy is not currently approved in the ASME Code for nuclear construction, but it is for non-nuclear construction. In another development, grain boundary engineering to tailor the microstructure with a series of thermomechanical treatments has been employed to improve Alloy 800H and resulted in strength enhancement and improved resistance to oxidation [71].

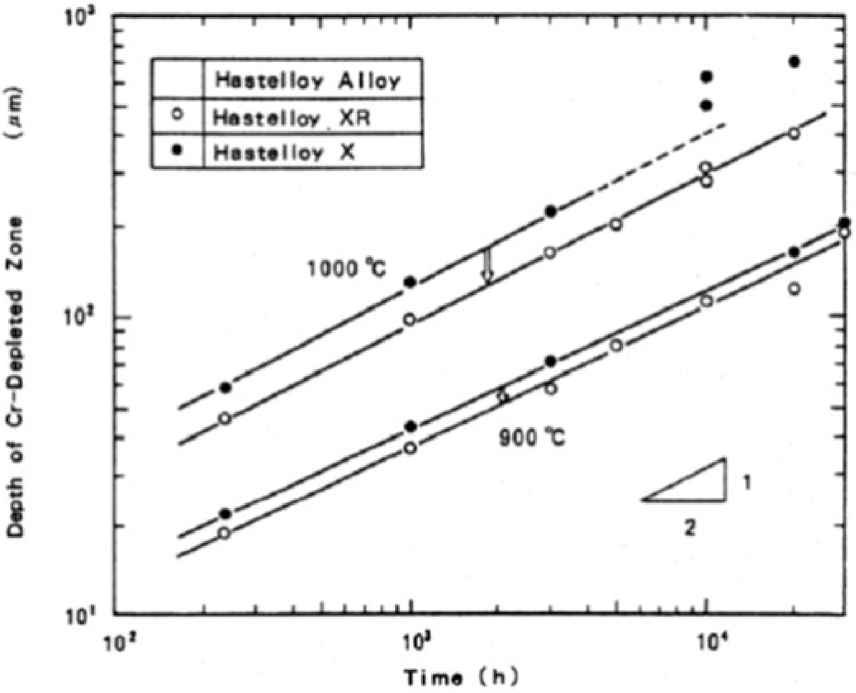


Figure 4-2
Depth of Cr-depletion zone vs time for Hastelloy X and Hastelloy Xr in JAERI-TYPE B helium [67]

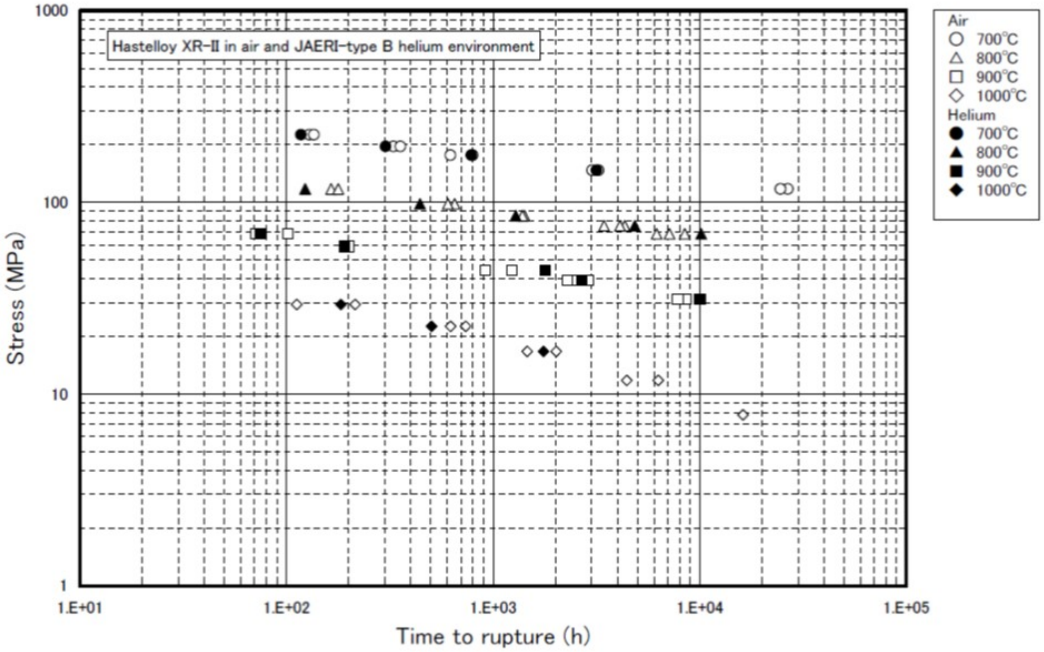


Figure 4-3
Creep rupture stress vs time for Hastelloy XR-II in air and helium showing no degradation of helium at temperatures to 1000°C [67]

5
ADVANCED MATERIALS

As noted in many places in this report, the issue of a 60-year design lifetime is a major issue due to lack of data under conditions of high temperature, high radiation, and degradation by the coolant. Such designs require substantial extrapolation of available shorter-term data which increases the uncertainties, often resulting in the need for imposition of lower design allowable stresses than may be necessary. Most of the difficulties associated with obtaining long-time data are improbable to overcome, but one possible option to mitigate those factors is to use currently available advanced materials that have improved properties but are not yet ASME Code qualified for nuclear construction. Examples are the Type 316FR stainless steel mentioned earlier with respect to the VHTR, and microstructural tailoring (for example, grain boundary engineering) of alloys such as 800H, 617, and the ferritic-martensitic (F-M) 9Cr-steels, Grades 91 and 92, with thermomechanical treatments. Similarly, developments with F-M ODS steels offer the possibility of significantly enhanced radiation resistance [75] as previously mentioned, especially for some GFR components subjected to very high radiation doses. Of course, another option is to develop new materials that are capable of operating at higher stresses and higher temperatures, with better radiation resistance and resistance to coolant degradation. Obviously, the path to develop such materials and, notably, to achieve approval from relevant standards and code committees, and the regulatory authority is a long and expensive process. However, as noted by Zinkle, et al. [59], advanced materials could directly reduce construction cost and enable improved safety margins with further cost reductions.

One current example of this is the inclusion into the ASME Code of Alloy 617 which, as shown in Figure 3-10, has more than double the strength of Alloy 800H at 750°C, the current proposed outlet temperature the both VHTRs reviewed in this report. However, at the higher temperatures of 950-1000°C proposed for some future VHTRs, such benefits decrease rapidly. Also, as shown in that figure, Alloy 230 is under consideration as an alternative to Alloy 617 because it has equivalent or even superior creep properties and may suffer less from oxidation [68]. Experiments by Tsai, et al. [74] at temperatures to 950°C in impure helium showed Alloy 625 superior to Alloy 617 and, as shown in Figure 3-10, that alloy is about 50% stronger than Alloy 617. Designating Alloy 625 an advanced alloy is tenuous in that it was developed more than 50 years ago, as was Hastelloy 718 which was developed on the heels of Alloy 625. At that time, the development of new alloys was achieved by significant experimental variations with different compositions, taking considerably more time than is now afforded through the tools of computational thermodynamics.

In the past few decades, advancements have been made with austenitic stainless steels as well. Some of these alloys are appropriate for consideration in the VHTRs with a 750°C outlet especially, and for both the VHTRs and GFR for applications outside of the reactor. Advanced stainless steels, such as NF709, the ultra-fine precipitate strengthened (HT-UPS+5% CW) stainless steel, and CW 12-15/15-25 Ti + Nb are three prominent examples [75,76,77]. With possibilities for operation at higher temperature than previous standard stainless steels, such alloys may enable application in the place of Alloy 800H, a much higher nickel-containing alloy.

The advancements made with ferritic-martensitic (F-M) steels and oxide-dispersion-strengthened (ODS) steels have taken advantage of significant increases of understanding of radiation effects at the atomic scale and, more recently, through the use of computational thermodynamics. One example was shown in Figure 3-9 for the standard Grade 92 steel and two modified versions that exhibit increases of 50% in strength, no significant loss of ductility, and retained toughness. The ODS concept involves a process of mechanically alloying the steel with, usually, Y_2O_3 to achieve a fine distribution of the Y_2O_3 particles that significantly enhance high-temperature strength by pinning movable dislocations, but also act as sinks for radiation defects to enhance radiation resistance [75, 76, 77]. Incorporation of titanium with the Y_2O_3 was also developed with the 14YWT alloy [78]. The ODS concept was also investigated for development of improved austenitic alloys [79].

Although there are loading applications that require the use of metallic materials, there are some structural material applications that are appropriate for non-metals such as advanced ceramics, particularly ceramic composites. For example, as discussed in this report and as reflected in the choice for internal structures in the EM^2 GFR, SiC-SiC composites offer substantial advantages relative to high resistance to irradiation and coolant degradation.

Zinkle [80] provides a recent review of advanced materials that includes discussion regarding next-generation steels, multilayer metallic nanocomposites, ceramic composites, MAX phase ceramics that are thermodynamically stable nanolaminates, bulk metallic glasses, and high-entropy alloys. Many of the next-generation steels (both ferritic-martensitic, austenitic, and ODS variants as discussed earlier), ceramics, and high entropy alloys show varying degrees of promise for irradiation resistance and high temperature stability. With the exception of the next-generation steels, most of these advanced materials are relatively very early in development as potential structural materials for advanced reactors. The gas turbine industry is performing substantial research on oxide-oxide ceramic matrix composites (Ox-Ox CMCs) that have been shown to survive temperatures of 1150°C and even higher under some conditions [81, 82]. Such materials, as with SiC-SiC, may be applicable for many high temperature portions of the Generation IV gas-cooled reactors. Although there has been no substantial work on radiation effects with such materials.

6
SUMMARY OF KNOWLEDGE GAPS

6.1 VHTR

This review considered two proposed and developing VHTRs, the AREVA/NGNP Alliance Reactor and the X-energy Xe-100 Reactor. Both are helium-cooled reactors with a gas outlet temperature of 750°C. The AREVA system uses the prismatic (graphite blocks) design with TRISO fuel particles, while the X-energy system uses a pebble-bed design for the core and with TRISO fuel particles embedded in larger graphite particles. Three major components comprise their systems; 1) a helium cooled nuclear reactor, 2) a heat transport system (steam generator), and 3) a cross vessel for helium flow from the reactor to the heat transport system. Although many details of operating temperatures, irradiation doses, and so on, are not available for the many individual components of the systems, the information that is available for each of them as well as detailed information from previous studies of VHTRs have been utilized to evaluate the gaps in knowledge that need either more detailed data to make an assessment or need additional research and development for deployment.

The GIF Annual Report [27] stated that, because the outlet temperatures are 750°C, current state-of-the art materials and components are applicable. Relative to the subject of heat utilization systems materials for the VHTR, the GIF Roadmap (2014) [18] stated that internal core structures and cooling systems, such as intermediate heat exchangers, hot gas ducts, process components and isolation valves, that are in contact with hot helium, can use current metallic materials up to a core outlet temperature of about 700 to 800°C. That document further noted that efforts in the United States have focused on developing the data needed to extend Alloy 800H for use up to 850°C and Alloy 617 for use up to 950°C, and that within the next four years (that is, by 2018), all of the data needed to codify these materials will be provided to the ASME.

In many cases, the most prominent factor is the need for sufficient data/information for a 60-year design lifetime. A detailed list of the identified potential gaps is included in Section 2.7 for the VHTRs, while this summary table (Table 6-1) below focuses on those considered highest priority.

Table 6-1
Summary of Gaps and Actions for VHTRs

Status	Gap	Action
Type 316FR material has improved properties relative to type 316 for long lifetimes.	Type 316FR is not currently approved in the ASME Code, but a fairly large database of relevant properties has been established.	Develop code case and pursue to code approval.
Composite materials have potential for use as internals component.	Composite performance at realistic fluence conditions are not known.	(From Design) Better define expected life fluences for composite internals' components. Perform testing (Irradiation exposure and post exposure mechanical properties) to determine viability of C-C and SiC-SiC composites under those conditions.
C-C and SiC-SiC may be susceptible to long term degradation under low partial pressures of oxygen.	Extent of potential degradation of C-C and SiC-SiC composites under low partial pressures of oxygen (at high temperature) is not known.	Experimental evaluation of stability of C-C and/or SIC-SiC composites under conditions simulating low partial pressures of oxygen at high temperatures.
LAS and 2.25Cr-1Mo steel appear promising for RPV but Materials degradation acceptable limits under fast fluence are expected to be lower than the 1 x1019 n/cm^2 for which LAS is used in LWRs.	LAS and 2.25Cr-1Mo steel mechanical and fracture properties after VHTR and GFR fast neutron irradiation design conditions are not known.	Confirm expected vessel end of life fast neutron fluence. Generate materials' fracture and mechanical properties as response to expect level of fast neutron fluence.
Identified core barrel materials will be expected to display time dependent behaviors under internals operating conditions.	Code validated creep and other time dependent materials properties are needed to support internals design and design assurance for 60-year life.	Time dependent high temperature properties must be developed and incorporated into Sec III, Div 5 code.
Fibrous insulation offers significant potential for reducing the material temperatures of vessel and hot gas containing ducting.	The thermal stability (morphology and structural integrity) of fibrous insulations under long term high temperature flowing gas is not known.	Available pertinent data should be reviewed and if necessary testing performed to establish stability or otherwise of candidate fibrous insulations under expected gas conditions.

Table 6-1 (continued)
Summary of Gaps and Actions for VHTRs

Status	Gap	Action
For the Xe-100 RPV, X-energy specifies SA-508, Grade 3, Class 2 for some of the RPV components (vessel flange, top head CRDM housings); X-energy also specifies SB-637, Alloy 718 for top head fasteners, and SA-540, Grade B24, Class 1 for center and bottom manway fasteners.	These materials are not included in Section III, Division 5.	High temperature data must be generated for these materials and code cases for inclusion into Sec III, Div 5 must be developed.
Fatigue for the RPV is not expected to be an issue, but there still exists the potential for fatigue degradation of materials.	Potential for high temperature fatigue exists.	More detailed design work is needed to resolve the potential for damage and failure mode by fatigue for the RPV.
Use of SA-508/533 material at temperatures the RPV is expected to exceed 371°C in the RPV for a 60-y lifetime, will require consideration of creep effects.	Creep and creep-fatigue effects for a 60-y lifetime will need to be included when evaluating strain limits in accordance with the requirements of Section III Division 5.	Creep and creep-fatigue properties need to be measured at temperatures exceeding 371°C and incorporated in Section III Division 5 of the ASME code.
SA-508 Grade 3, Class 2 material which is specified for the vessel flange and top head CRDM housings in the X-energy Xe-100 RPV, and maximum anticipated temperatures for those components are not known.	Section III, Division 5 does not include SA-508 Grade 3 Class 2 material.	Maximum temperatures need to be resolved the mechanical properties data for SA-508 Grade 3 Class 2 material need to be developed and incorporated into Section III Division 5 of the ASME code.
Consideration based on the design anticipated temperatures indicates the potential for thermal aging of the low-alloy steels for the RPV, cross vessel, and steam generator.	In plant knowledge of the extent of thermal aging as life progresses.	Consideration of a surveillance program for thermal aging is recommended for the vessels and so on.
Since radiation is a key component of system cooling Emissivity needs to be available for RPV and cross vessel materials (and potential coatings) in and after exposure to helium environment.	Emissivity of exposed materials is not known.	Appropriate measurements of simulation service exposed materials should be made.

Table 6-1 (continued)
Summary of Gaps and Actions for VHTRs

Status	Gap	Action
Aspects associated with creep and fatigue, rotational stress, and so on, in the PCS are not currently accurately known.	More accurate values of creep and fatigue, rotational stress, and so on, in the PCS need to be known.	Design progression work is needed to resolve unknowns in values of creep and fatigue, rotational stress, and so on, in the PCS.
Missile generation and debris generation from accident in the turbomachinery could impact the reactor system (it should not).	Currently consequences of damage from missile debris generation from turbo machinery is not known.	Assessments of damage from missile debris generation from turbo machinery to remainder of plant should be included in design work.
There will be dissimilar metal joints between 800H and 2.25Cr-1Mo steel.	Concern for corrosion questions where alternating wet-dry conditions may exist.	Further corrosion testing required.
Dissimilar metals will be interfaced at supports and long-time thermal excursions may cause mutual damage.	Potential for fretting at supports.	Fretting testing of DMWs.
Heat rejection from RCCS during accident conditions may incorporate radiative cooling as a key component.	Whether RCCS panel emissivity is needed to maintain fuel temperature below the limits during accident conditions is not known.	More detailed assessments of heat rejection from the RCCS under accident conditions are required for specific designs.
Properties of graphite are known to be affected by irradiation.	Extent of damage under expected life conditions is not known.	More precise correlations between expected irradiations and known degradation of real graphite structures must be developed.
Numerous grades of graphite that have been identified as potential candidates for both the AREVA and X-energy VHTRs.	Selection of specific grades for the various components needs continued development of adequate databases.	Development of grade specific material properties for candidate graphites.
Graphite supply fluctuates over time – grades, availability and so on.	Consistent supply of graphite for fuels/structures will be required to support 60-year life of plant.	

Table 6-1 (continued)
Summary of Gaps and Actions for VHTRs

Status	Gap	Action
Design with graphite is not currently based on well-developed standards and codes.	Consensus design codes and materials testing standards are not in place.	Continue to development “graphite data” and drive to implement codes and standards for graphite.
Effect of neutron irradiation-induced displacement damage in graphite is not well known.	Need for improved understanding and models for displacement damage of graphite, and development of whole-core structural models.	Develop improved understanding and models for displacement damage of graphite, and development of whole-core structural models.
Predictive models of graphite behavior are required to support in service reliability assessments.	Continued development of graphite analytical models for oxidation, property changes, and dimensional changes and creep induced by irradiation is needed to develop enhance predictive reliability.	Measure degradation data and incorporate into predictive reliability models for irradiated graphite.
Plant monitoring via non-destructive evaluations will be required to monitor plant integrity during operations.	Advanced NDE methods are not yet developed to fully characterize damage in high temperature materials employed in VHTR.	Identify required NDE tools and develop technical bases for them.

6.2 GFR

Choi, et al. [48] specifically noted that the EM^2 design relies on non-conventional nuclear materials and GA is performing high-risk R&D programs that include characterization of structural materials under high neutron irradiation, with some irradiation experiments already performed in various research reactors. Schleicher and Bertch [47] stated that: "because there is no precedent for the EM^2 core and PCU designs, GA believes that a one-unit prototype plant is required to reduce the technical and licensing risk to an acceptable level before embarking upon a commercial plant". Choi and Schleicher [49] also noted that: "Compared with other advanced reactors, the technology readiness level of the GFR is relatively low because no experimental or commercial GFR has been built and the materials and core concepts are new. However, in order to meet the demanding economic, operational, and safety requirements of a twenty-first century power source, it is essential to implement advanced features in the new reactor design. Fortunately, the GFR can share the technologies from other advanced reactors such as the high-temperature power conversion of the very high– temperature reactor and the fast neutron physics of the liquid metal reactor, which would be the stepping stone to the demonstration and commercialization of the GFR." Additionally, they espoused that licensing and construction for the first-of-a-kind EM^2 should address the key technical challenges of the GFR and demonstrate the viability of the core physics, material performance, and safety. More specifically, in addition to research and development associated with the fuel and system control issues, they recommended specific follow-up studies for demonstration of critical components made of SiC-SiC and Zr_3Si_2 material under high temperature and high-irradiation environment.

This review has identified issues that need resolution for the GFR, either due to insufficient information to judge the issue, or issues that have been noted to not have sufficient knowledge. In many cases, the most prominent factor is the need for sufficient data/information for a 60-year design lifetime. A detailed list of the identified potential gaps is included in Section 3.7 for the GFR, while this summary table (Table 6-2) focuses on those considered highest priority.

Table 6-2
Summary of Gaps and Actions for GFRs

Status	Gap	Action
The anticipated radiation levels are not known for the EM^2 reactor (current estimates to 100 dpa).	Estimates of radiation levels will be critical for the ultimate material selections for many of the components in the reactor system.	Design progression to better identify expected irradiation levels.
Peak temperatures of 1920°C are anticipated for SiC-SiC core components. Properties of SiC-SiC grades depend on processing and so on.	Properties of specific grades of SiC-SiC for core components are not know for such high temperatures.	Very high temperature strength needs to be confirmed for the specific SiC-SiC grade.
Integrity of SiC-SiC against fracture under expected loading conditions is needed.	Loading conditions on the various components are not known. It is not clear if lower values of fracture toughness compared with the Handbook values will present an issue for the use of SiC composites.	Identify expected fracture toughnesses of component processed SiC-SiC composites and validate integrity vs expected loading conditions.
SiC-SiC composites can be porous to fission products (via atomic diffusion or molecular diffusion through microcracks).	Demonstration of sufficiently low porosity of liner systems.	Demonstration of improved hermeticity of SiC-SiC composites of development of liner (for example, W-Re) liner systems.
Extreme conditions likely to be imposed on some of the SiC-SiC components in the GFR will involve relatively high temperatures and high irradiation doses for a 60-y design lifetime.	The extreme conditions likely to be imposed on some of the SiC-SiC components in the GFR far exceed the available results.	Further experiments on response of SiC-SiC to high dose and high temperature are needed to provide confidence for a 60-y design lifetime.
Zr_3Si_2 has been selected as a reflector material.	Current irradiation effects data for the Zr_3Si_2 reflector material are insufficient to enable reasonable prediction for a 60-yr lifetime.	More irradiation effects data need to be measured for Zr_3Si_2 reflector material sufficient to enable reasonable prediction for a 60-y lifetime.
Zr_3Si_2 has been selected as a reflector material.	Insufficient knowledge of the effects of impure helium on Zr_3Si_2.	Generate data on the effects of impure helium on the structure and properties of Zr_3Si_2.

Table 6-2 (continued)
Summary of Gaps and Actions for GFRs

Status	Gap	Action
Fibrous insulation offers significant potential for reducing the temperatures of metal containment components.	The radiation stability (morphology and structural integrity of fibrous insulations under long term high temperature flowing gas is not known.	Available pertinent data should be reviewed and if necessary testing performed to establish stability or otherwise of candidate fibrous insulations under expected gas and irradiation conditions.
Metallic materials will be operated at high temperatures – significantly in excess of those for which the materials are currently code allowed for nuclear construction.	High temperature tensile properties, creep strength, and swelling resistance, must also be evaluated for fatigue and creep-fatigue behavior under the conditions anticipated. Creep-fatigue is a significant issue for Grade 91 and other F-M steels.	High temperature behaviors of all materials need to be measured and incorporated into Section III, Division 5 of the ASME Code.
Currently operating conditions for the recuperator and other components in the PCU are not accurately known.	Additional information regarding operating conditions for the recuperator and other components in the PCU are needed for a comprehensive assessment.	Progress design of GFR so that operating conditions for the recuperator and other components in the PCU are better known.
Refractory alloys may be considered for service in He (low ppm 0_2) environment.	Refractory alloys are sensitive to impurity corrosion and degradation of mechanical properties from impurities (low ppm 0_2) in the He environment.	Demonstrate realistic impurity control of the helium to possibly as low as 1 ppm, or Demonstrating a lower sensitivity for the selected material, or Demonstration of an effective anticorrosion coating.
Refractory alloys may be considered for service in high radiation dose environment.	(quantitative) Effects of neutron irradiation at high dose on refractory alloys is not known.	Develop irradiation effects information at substantially higher doses than currently available.

7

OVERVIEW SUMMARY

This review considered two proposed and developing VHTRs, the AREVA/NGNP Alliance Reactor and the X-energy Xe-100 Reactor, as well as the General Atomic GFR, the Energy Multiplier Module (EM^2). Both VHTRs are helium-cooled reactors with a gas outlet temperature of 750°C. The AREVA system uses the prismatic (graphite blocks) design with TRISO fuel particles, producing 625 MWt, while the X-energy system uses a pebble-bed design for the core and with TRISO fuel particles embedded in larger graphite particles to produce 200 MWt with a 4-pack module plant. Both VHTRs include three major components: 1) a helium cooled nuclear reactor, 2) a heat transport system (steam generator), and 3) a cross vessel for helium flow from the reactor to the heat transport system. The GFR EM^2 is a high temperature direct Brayton cycle helium cooled fast reactor with an outlet temperature of 850°C to produce 500 MW thermal, designated an advanced small modular reactor (SMR) by GA. The system has a reactor connected with a cross vessel to a power conversion system that has a direct-drive closed cycle gas turbine. The system includes a separate residual heat removal system.

Although many details of operating temperatures, irradiation doses, and so on, are not available for the many individual components of the systems, especially the GFR, the information that is available for each of them as well as detailed information from previous studies of VHTRs and GFRs have been utilized to evaluate the gaps in knowledge that need either more detailed data to make an assessment or need additional research and development for ultimate deployment for the defined 60-year design lifetimes. The review considered metallic, non-metallic, and graphite materials, including brief discussions concerning material compatibility with impure helium and a brief discussion regarding advanced materials. One notable common item is that, in contrast with many previous designs for high temperature reactors, all three systems plan to construct their reactor pressure vessel and cross vessel from the low-alloy steels traditionally used for light-water reactors and to maintain the temperatures of those components within the limits prescribed by the ASME Boiler and Pressure Vessel Code with the use of insulation. However, information regarding the methods to be used for limiting the temperatures, specific insulation and cooling methods, were not described for the VHTRs nor the GFR.

Although the Generation IV International Forum has stated that the decrease of outlet temperature from the previous 950°C to 750°C will allow for current state-of-the art materials and components, this review has identified a variety of gaps/issues that need resolution for the VHTRs, some due to insufficient information with which to judge the issue and others that have been noted to not have sufficient knowledge. *In many cases, the most prominent factor is the need for sufficient data/information for a 60-year design lifetime.* For a number of components for which Type 316 stainless steel would be a likely candidate, Type 316FR material has improved properties relative to type 316 for long lifetimes; it is not currently approved in the ASME Code, Section III, Division 5, but the fairly large available database should make Code approval relatively easy. There are some RPV top head components that will experience high temperature, but the specified candidate materials are not included in Section III, Division 5.

Irradiation effects on graphite material properties (creep, expansion/contraction, thermal conductivity), consistency of graphite quality and performance over the service life, and availability of graphite supply (coke sources, graphite vendors) for long lifetime designs (for example, 60 years) are the primary issues associated with graphite for the current VHTR designs.

General Atomics (GA) specifically noted that the EM^2 design relies on non-conventional nuclear materials and GA is performing high-risk R&D programs that include characterization of structural materials under high neutron irradiation, with some irradiation experiments already performed in various research reactors. They have also noted that there is no precedent for the EM^2 core and power conversion unit designs, that no experimental or commercial GFR has ever been built, leading them to believe that a one-unit prototype plant is required to reduce the technical and licensing risk to an acceptable level before embarking upon a commercial plant. This prototype plant will be designed with the same inlet and outlet temperatures as the commercial system but will produce 120 MWt, about one-fifth the size of the larger version. For the GFR, the higher outlet temperature presents a somewhat greater challenge than that associated with the VHTRs, but the most significant challenge is the high temperature combined with very high radiation exposures that many of the materials and components will experience.

As for the VHTRs, this review has identified gaps/issues that need resolution for the GFR, either due to insufficient information to judge the issue, or issues that have been noted to have insufficient knowledge. In the case of the GFR, the most prominent factor is the need for data/information relative to irradiation effects for all in-reactor candidate materials, especially for a 60-year design lifetime. For the internal components for which SiC-SiC composites are the candidate material, the very high temperature of 1920°C under off-normal conditions compels the need for investigation of the material strength under the temperature-time conditions anticipated. Additionally, the stated requirements for fracture toughness and the very high irradiation doses need further clarification and experimental data to verify the material capability under the anticipated extreme conditions. For the Zr_3Si_2 reflector material, continuing experiments are needed to obtain sufficient irradiation effects data to enable reasonable prediction for a 60-year lifetime, as are the effects of impure helium. Because the radiation levels in the RPV are extremely high, it is necessary to know the anticipated fast neutron fluence for the GFR RPV to enable evaluation of the irradiation effects on the RPV itself as well as on the graphite portion of the reflector.

It is reasonably clear that the VHTRs are close to being ready for commercial deployment. Outlet temperatures of 750°C, relatively low radiation levels to structural components (other than the graphite), and the fact that there is experience with test reactors, VHTRs could soon see commercial deployment. The primary challenge for those systems is the specified design lifetime of 60 years, which compels designs for component replacement where practicable, as well as continuing research and development of advanced materials with increased potential for long term operation at high temperature. Design concepts for advanced VHTRs are similar to those for the GFR, with an outlet temperature of 850°C and very high radiation levels for many structural components. Under these conditions the operating and off-normal conditions imposed on components are much more challenging. The designers' stated plan to design and deploy a small demonstration reactor seems reasonable not only as a demonstration system, but also as a test bed for system operation and structural materials performance.

Lastly, materials gaps and actions to resolve those gaps were provided in Tables 6-1 (VHTRs) and 6-2 (GFRs). These gaps and actions will be addressed through various EPRI and industry projects over the next decade.

8
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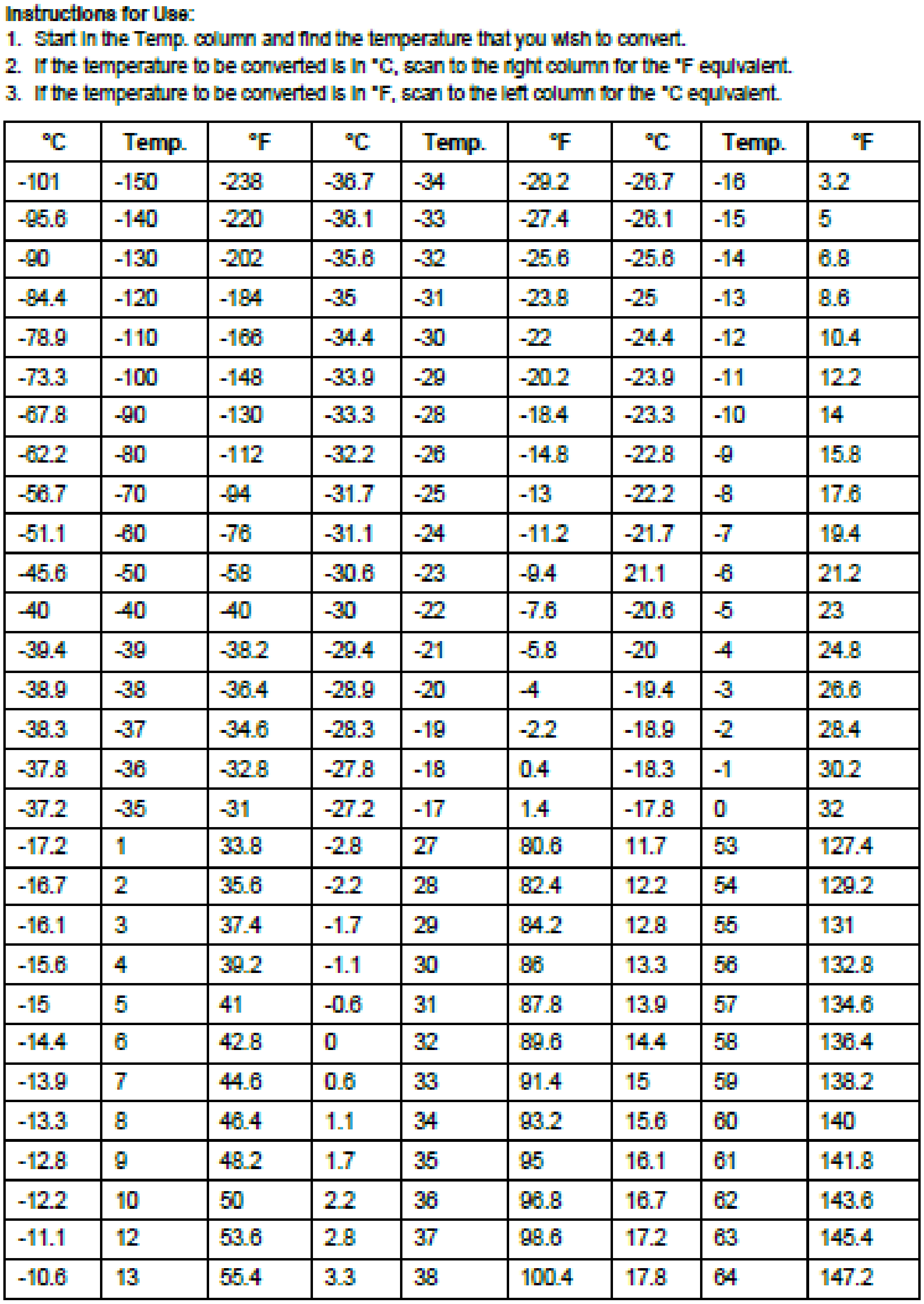
A

TEMPERATURE CONVERSION TABLES

Table A-1
Temperature conversion chart

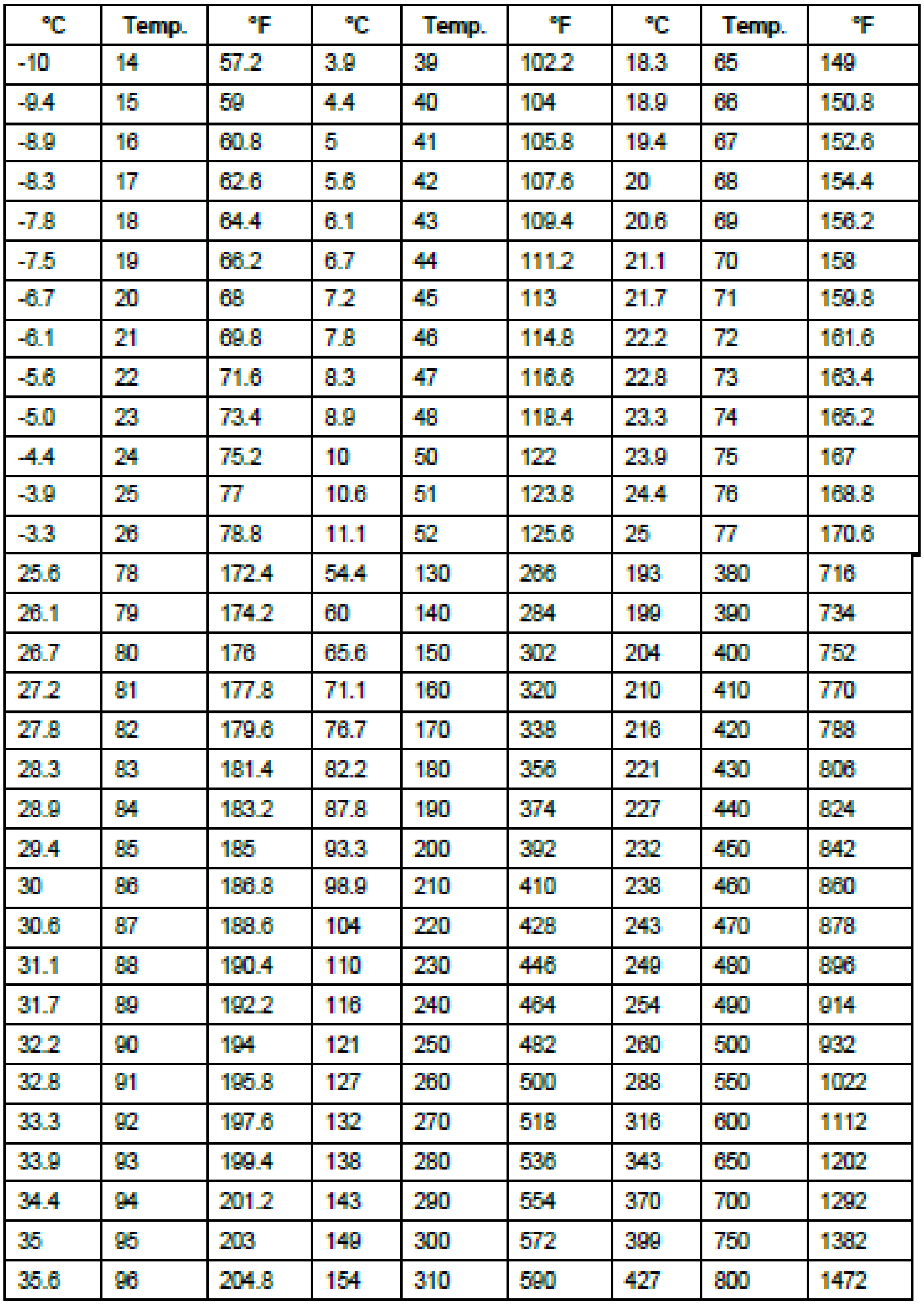


Instructions for Use:

1. Start in the Temp. column and find the temperature that you wish to convert.
2. If the temperature to be converted is in °C, scan to the right column for the °F equivalent.
3. If the temperature to be converted is in °F, scan to the left column for the °C equivalent.

°C	Temp.	°F	°C	Temp.	°F	°C	Temp.	°F
-101	-150	-238	-36.7	-34	-29.2	-26.7	-16	3.2
-95.6	-140	-220	-36.1	-33	-27.4	-26.1	-15	5
-90	-130	-202	-35.6	-32	-25.6	-25.6	-14	6.8
-84.4	-120	-184	-35	-31	-23.8	-25	-13	8.6
-78.9	-110	-166	-34.4	-30	-22	-24.4	-12	10.4
-73.3	-100	-148	-33.9	-29	-20.2	-23.9	-11	12.2
-67.8	-90	-130	-33.3	-28	-18.4	-23.3	-10	14
-62.2	-80	-112	-32.2	-26	-14.8	-22.8	-9	15.8
-56.7	-70	-94	-31.7	-25	-13	-22.2	-8	17.6
-51.1	-60	-76	-31.1	-24	-11.2	-21.7	-7	19.4
-45.6	-50	-58	-30.6	-23	-9.4	21.1	-6	21.2
-40	-40	-40	-30	-22	-7.6	-20.6	-5	23
-39.4	-39	-38.2	-29.4	-21	-5.8	-20	-4	24.8
-38.9	-38	-36.4	-28.9	-20	-4	-19.4	-3	26.6
-38.3	-37	-34.6	-28.3	-19	-2.2	-18.9	-2	28.4
-37.8	-36	-32.8	-27.8	-18	0.4	-18.3	-1	30.2
-37.2	-35	-31	-27.2	-17	1.4	-17.8	0	32
-17.2	1	33.8	-2.8	27	80.6	11.7	53	127.4
-16.7	2	35.6	-2.2	28	82.4	12.2	54	129.2
-16.1	3	37.4	-1.7	29	84.2	12.8	55	131
-15.6	4	39.2	-1.1	30	86	13.3	56	132.8
-15	5	41	-0.6	31	87.8	13.9	57	134.6
-14.4	6	42.8	0	32	89.6	14.4	58	136.4
-13.9	7	44.6	0.6	33	91.4	15	59	138.2
-13.3	8	46.4	1.1	34	93.2	15.6	60	140
-12.8	9	48.2	1.7	35	95	16.1	61	141.8
-12.2	10	50	2.2	36	96.8	16.7	62	143.6
-11.1	12	53.6	2.8	37	98.6	17.2	63	145.4
-10.6	13	55.4	3.3	38	100.4	17.8	64	147.2

Table A-1 (continued)
Temperature conversion chart



°C	Temp.	°F	°C	Temp.	°F	°C	Temp.	°F
-10	14	57.2	3.9	39	102.2	18.3	65	149
-9.4	15	59	4.4	40	104	18.9	66	150.8
-8.9	16	60.8	5	41	105.8	19.4	67	152.6
-8.3	17	62.6	5.6	42	107.6	20	68	154.4
-7.8	18	64.4	6.1	43	109.4	20.6	69	156.2
-7.5	19	66.2	6.7	44	111.2	21.1	70	158
-6.7	20	68	7.2	45	113	21.7	71	159.8
-6.1	21	69.8	7.8	46	114.8	22.2	72	161.6
-5.6	22	71.6	8.3	47	116.6	22.8	73	163.4
-5.0	23	73.4	8.9	48	118.4	23.3	74	165.2
-4.4	24	75.2	10	50	122	23.9	75	167
-3.9	25	77	10.6	51	123.8	24.4	76	168.8
-3.3	26	78.8	11.1	52	125.6	25	77	170.6
25.6	78	172.4	54.4	130	266	193	380	716
26.1	79	174.2	60	140	284	199	390	734
26.7	80	176	65.6	150	302	204	400	752
27.2	81	177.8	71.1	160	320	210	410	770
27.8	82	179.6	76.7	170	338	216	420	788
28.3	83	181.4	82.2	180	356	221	430	806
28.9	84	183.2	87.8	190	374	227	440	824
29.4	85	185	93.3	200	392	232	450	842
30	86	186.8	98.9	210	410	238	460	860
30.6	87	188.6	104	220	428	243	470	878
31.1	88	190.4	110	230	446	249	480	896
31.7	89	192.2	116	240	464	254	490	914
32.2	90	194	121	250	482	260	500	932
32.8	91	195.8	127	260	500	288	550	1022
33.3	92	197.6	132	270	518	316	600	1112
33.9	93	199.4	138	280	536	343	650	1202
34.4	94	201.2	143	290	554	370	700	1292
35	95	203	149	300	572	399	750	1382
35.6	96	204.8	154	310	590	427	800	1472

Table A-1 (continued)
Temperature conversion chart

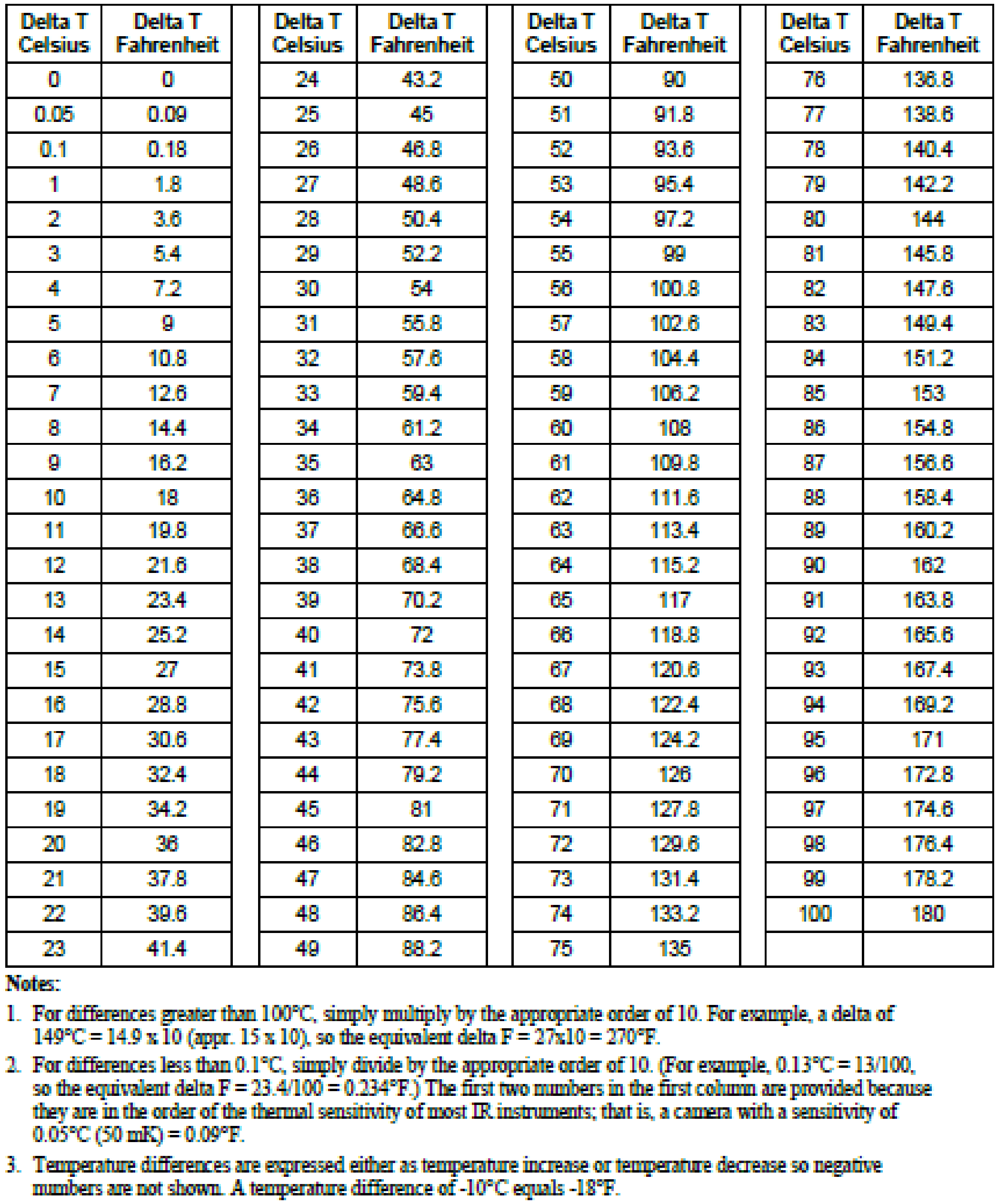
°C	Temp.	°F	°C	Temp.	°F	°C	Temp.	°F
36.1	97	206.6	160	320	608	454	850	1562
36.7	98	208.4	166	330	626	482	900	1652
37.2	99	210.2	171	340	644	510	950	1742
37.8	100	212	177	350	662	538	1000	1832
43.3	110	230	182	360	680	566	1050	1922
48.9	120	248	188	370	698	593	1110	2012
621	1150	2102	843	1550	2822	1066	1950	3542
649	1200	2192	871	1600	2912	1093	2000	3632
677	1250	2282	899	1650	3002	1149	2100	3812
704	1300	2372	927	1700	3092	1204	2200	3992
732	1350	2462	954	1750	3182	1260	2300	4172
760	1400	2552	982	1800	3272	1316	2400	4352
788	1450	2642	1010	1850	3362	1371	2500	4532
816	1500	2732	1038	1900	3452			

Conversion Factors

°C = (°F - 32) x 1.8 0 Kelvin = -273.16°C

°F = (°C x 1.8) + 32 0 Rankine = -459.69°F

Table A-2
Temperature difference conversion chart



Delta T Celsius	Delta T Fahrenheit	Delta T Celsius	Delta T Fahrenheit	Delta T Celsius	Delta T Fahrenheit	Delta T Celsius	Delta T Fahrenheit
0	0	24	43.2	50	90	76	136.8
0.05	0.09	25	45	51	91.8	77	138.6
0.1	0.18	26	46.8	52	93.6	78	140.4
1	1.8	27	48.6	53	95.4	79	142.2
2	3.6	28	50.4	54	97.2	80	144
3	5.4	29	52.2	55	99	81	145.8
4	7.2	30	54	56	100.8	82	147.6
5	9	31	55.8	57	102.6	83	149.4
6	10.8	32	57.6	58	104.4	84	151.2
7	12.6	33	59.4	59	106.2	85	153
8	14.4	34	61.2	60	108	86	154.8
9	16.2	35	63	61	109.8	87	156.6
10	18	36	64.8	62	111.6	88	158.4
11	19.8	37	66.6	63	113.4	89	160.2
12	21.6	38	68.4	64	115.2	90	162
13	23.4	39	70.2	65	117	91	163.8
14	25.2	40	72	66	118.8	92	165.6
15	27	41	73.8	67	120.6	93	167.4
16	28.8	42	75.6	68	122.4	94	169.2
17	30.6	43	77.4	69	124.2	95	171
18	32.4	44	79.2	70	126	96	172.8
19	34.2	45	81	71	127.8	97	174.6
20	36	46	82.8	72	129.6	98	176.4
21	37.8	47	84.6	73	131.4	99	178.2
22	39.6	48	86.4	74	133.2	100	180
23	41.4	49	88.2	75	135		

Notes:

1. For differences greater than 100°C, simply multiply by the appropriate order of 10. For example, a delta of 149°C = 14.9 x 10 (appr. 15 x 10), so the equivalent delta F = 27x10 = 270°F.
2. For differences less than 0.1°C, simply divide by the appropriate order of 10. (For example, 0.13°C = 13/100, so the equivalent delta F = 23.4/100 = 0.234°F.) The first two numbers in the first column are provided because they are in the order of the thermal sensitivity of most IR instruments; that is, a camera with a sensitivity of 0.05°C (50 mK) = 0.09°F.
3. Temperature differences are expressed either as temperature increase or temperature decrease so negative numbers are not shown. A temperature difference of -10°C equals -18°F.

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