The Challenge of Nuclear Reactor Structural Materials for Generation IV Nuclear Energy Systems

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1 ABSTRACT (NO MORE THAN 200 WORDS)

Generation IV Nuclear Energy Systems have been studied in the framework of the “Generation IV” International Forum, there can be little doubt that nuclear reactor structural materials technology, including: Selection, development and qualification, is one of the key issues to success of Gen Ⅳ Nuclear Energy Systems. By striving to meet this challenge, which is beyond the experience of the current nuclear power plants, the belt zone structural materials are required good resistance to irradiation damage, high thermal stress capacity, excellent resistance encompassing stress-corrosion cracking, and highly predictable responses to extreme levels, compatibility with Heat-Transfer media and other materials, very long-term stability enhanced in the system, adequate resources and easy fabrication as well as weld-ability, etc, to guarantee the security and integrity of the pressure boundary. This paper analyses and compares several materials under active consideration performance requirements, and also discusses candidate materials for use in different reactor components, which include various ferritic-martensitic steels, advanced austenitic stainless steels, nickel-base super-alloys, oxide-dispersion strengthened alloys, refractory alloys and etc. It is demonstrated that new materials, such as metals, carbides, nitrides, oxide, novel alloy, solid solutions or composites focuses on higher stability and better mechanical performances. In addition, the emergence of new international structural materials initiatives and their potential roles will be described.

2 INTRODUCTION

As the world’s population grows, so does its reliance on electricity, likewise, energy demands are soaring as new technologies and expanded development create additional energy needs, striving for a better quality of life. Security of energy supply and energy cost stability in the long term, plus the efforts to combat the fossil fuels dwindle and global climate change, argue in favour of a greater diversity in sources of energy supplies, as renewable energy sources are still in their infancy, being an unrealistic means to provide base-load generation. Thus, it's time to realize nuclear energy systems that are safe, plentiful, economical and environment friendly, having an essential role world-wide to play for the future.

Against this background, Generation IV Nuclear Energy Systems have been studied in the framework of the “Generation IV” International Forum (GIF), aiming to meet challenging technology in four areas: sustainability, economics, safety and reliability, proliferation resistance and physical protection. Six Generation IV systems have been selected over others for further research and development internationally by GIF countries on the basis of being clean, safe and cost-effective means of meeting increased energy demands on a sustainable basis, while being resistant to diversion of materials for weapons proliferation and secure from terrorist attacks.

There can be little doubt that nuclear reactor structural materials technology is one of key challenges to success of Gen IV Nuclear Energy Systems. Improved economics and reliability are prerequisites of each Gen IV system, and improved structural materials performance will allow higher operating temperatures and pressures, longer lifetimes and reduced down time. Therefore, irradiation resistance, high-temperature strength, and high-temperature design methodology are greatest materials challenges.

3 THE CHALLENGE OF MATERIALS FOR GEN IV

3.1 The common challenge

Although each system presents its particular challenges, several common themes emerge notably: resistance to irradiation damage, dimensional stability particularly with respect to high-temperature creep, corrosion resistance encompassing both uniform corrosion and localized effects such as stress-corrosion cracking, and highly predictable responses out to extreme levels of both temperature and neutron damage.

1) Excellent dimensional stability and high thermal stress capacity, responses to higher operation temperatures, harsher neutron irradiation flux, and extremely corrosive environments, which are beyond the experience of the current nuclear power plants. Favourable mechanical properties such as strength, ductility, creep rupture, fatigue, creep-fatigue interactions, and etc.
2) High degree of compatibility with Heat-Transfer media (especially, process-heat use for large-scale hydrogen generation) and other materials (cladding, reactants), in this regard, stress corrosion cracking (SCC), irradiation-assisted stress corrosion cracking (IASCC) and many other issues are important.
3) 60-year operating lives for Gen IV systems will require very long-term materials enhanced stability, high reliability, adequate resources and easy fabrication as well as weld-ability, environmental and aging effects, and etc. are other important aspects that need to be looked into during materials selection process.

3.2 The challenge of materials for VHTR

The Very High Temperature Reactor System (VHTR) is a next step in the evolutionary development of high-temperature gas-cooled reactors. This system will operate at higher temperatures than other gas-cooled reactors and will require different materials for important components.

The key feature of VHTR system is the specification for an outlet temperature of 950 to 1,100°C, at these core-outlet temperatures, the reactor pressure vessel temperature will exceed 450°C. To realize this goal, further development of Ni-Cr-W super-alloys and other promising new metallic alloys will be required. Graphite can be used for core internals, and further improvements in graphite properties such as oxidation resistance and structural strength will be necessary. Ceramic materials might be required for other core internals and cooling system components, such as the intermediate heat exchanger, hot gas ducts, and isolation valve sheets. Carbon-carbon composites are being considered for control sheaths for VHTR that employ prismatic graphite blocks for core structures. Other promising ceramics that might be considered include fiber-reinforced ceramics, sintered aloha SiC, oxide composite ceramics, and compound materials.

Table 1. Candidate high temperature materials.

Material	Temperature Limit	Limiting Factor
9Cr steel (T91)	*Operating: 550 / Max: 650*	*Long term Creep Strength* *ODS* version-Fusion development tested in MACROSTAR*
Austenitic Stainless Steel	*Operating: 550 / Max: 750*	*Long term Creep Strength*
Alloy 800	*Operating: 700 / Max: 950*	*Long term Creep Strength*
Alloy 800H		*Corrosion*
Alloy 617	*Operating: 800 / Max: 950*	*Long term Creep Strength*
Alloy 230		*<10MPa@100,000h*
Alloy XR		*Corrosion; Cr2O3 evaporation*

ODS-Alloys	*Operating: 550 / Max: 750*	*Manufacturing development required* *Investigated in EXTREMAT & MACROSTAR* *Difficult to from & fabricate*

*ODS: Oxide dispersion-strengthened steels (typical ferritic-martensitic)

The R&D activities proposed for VHTR materials are shown as follows:

1) Acquisition of mechanical and thermal properties data, fracture behaviour information, and oxidation data,
2) Irradiation tests,
3) Post-irradiation heat-up test of fuel to evaluate accident-related transient behaviour,
4) Development of materials behaviour models and a stress analysis code that considers anisotropy,
5) High-temperature alloys which are capable of extended operation at desirable temperatures will need to be identified.

3.3 The challenge of materials for MSR

The Molten Salt Reactor (MSR) is unique among selected Generation IV concepts in which it uses a fluid fuel rather than a solid fuel. However, the higher outlet temperatures of the system imply similar materials property, as other Generation IV systems concerned.

The materials proposed for MSR application are metal alloys and graphite. The metal alloys include INOR 8, Hastelloy B and Hastelloy N, Ni-based alloys, and niobium-titanium alloys, while a modified Hastelloy N alloy be preferred. An addition of about 2 wt% titanium appears to reduce irradiation embrittlement due to helium bubble formation and resistance enhancement to intergranular tellurium attack. Graphite is proposed as an in-core structural material and as a moderator. The challenges for graphite use in this application are radiation damage that leads to dimensional changes in the graphite, salt penetration into the graphite, and xenon absorption into the graphite, which can increase fission product poisoning effects and reduce the breeding/conversion ratio.

The proposed performance R&D phase calls for additional testing of candidate structural materials in molten fuel salt test loops. In particular, such testing with materials as Hastelloy N would include in-pile testing in loops. The tests would allow assessment of thermal gradient-enhanced corrosion effects in molten salt, including measurements of dissolution parameters, diffusion coefficients and kinetic coefficients toward temperature. Development of redox control techniques would be particularly important for materials compatibility in the secondary loop. The extent of tellurium embrittlement and the effect of radiation damage on mechanical properties would also be assessed. Test materials should include nickel based alloys with demonstrated performance in MSR test programs of the 1950s and 1960s such as INOR-8, Hastelloy B and N, and Inconel, as well as other promising materials such as niobium-titanium alloys, for which lifetime performances have not yet been demonstrated.

The nickel based alloys have been proven as suitable MSR structural materials. For example, INOR-8 is strong, stable, corrosion-resistant, and has good welding and forming characteristics. It is fully compatible with graphite, with non-sodium salts up to 815°C and with sodium salts up to 700°C. Modified Hastelloy N, developed for use with fluoride salt at high temperature, up to 800°C, has proven to be corrosion resistant but requires longer-term testing. For non-graphite core concepts, it must be noted that nickel based alloys are sensitive to He-induced embrittlement under irradiation, which results in a reduction of the creep ductility of the alloy. Tests show that titanium addition, up to 2wt%, solves the embrittlement problem and increases resistance to tellurium attack.

Further development of graphite is deferred until the Performance R&D phase. The development of sealing techniques to stop xenon penetration will be undertaken. High-quality, fine-grained graphite is specified to reduce salt penetration. However, dimensional changes induced by radiation damage would be considered, as those dimensional changes require somewhat frequent replacement of in-core graphite structure, e.g., every 4 years for the Molten Salt Breeder Reactor.

3.4 The challenge of materials for GFR

The Gas-Cooled Fast Reactor System (GFR) is proposed to combine the advantages of high-temperature gas-cooled reactors with the sustainability advantages that are possible with a fast-spectrum reactor, and to employ a direct helium Brayton cycle with core outlet temperatures of about 850°C. The GFR system presents many technical challenges, such as high-power density up to 100MW/m3, and high burn-up up to 15%, which is largely associated with the identification of structural materials capable of withstanding extremes of fast neutron flux and high temperatures, especially under fault conditions, when adequate heat removal from the core is likely to be the issue which determines the feasibility of the system.

Some Technical Working Groups have proposed basic research studies to complement an analysis of candidate materials for GFR application, selection of specific materials for in-core and out-of-core applications will be based on the best compromise of the following attributes:

1) Fabric-ability and welding capability (especially on-site joining technologies),
2) Physical, thermal and mechanical properties, initial characteristics and degradation of properties under neutron flux and dose which is high for in-core, material and low or moderate for out-of-core materials,
3) Microstructure and phase stability under irradiation,
4) Irradiation creep, in-pile creep and swelling properties,
5) Out-of-pile and in-pile compatibility with helium coolant, including impurities and actinide compounds (for inert matrix materials).

The main challenges are in-vessel structural materials, both in-core and out-of-core, which will have to withstand fast-neutron exposures and high temperatures, up to 1600°C in some accident scenarios. Ceramic materials are therefore the proposed option for in-core use, and composite ceramic or inter-metallic compounds will be suggested as alternatives. The most promising ceramic materials for core structures are carbides (preferred options are SiC, ZrC, TiC, NbC), nitrides (Zr N, TiN), and oxides (MgO, Zr(Y)O2). Inter-metallic compounds like Zr3Si2 are promising candidates as fast-neutron reflector materials. For other internal core structures, mainly the upper and lower structures, shielding, the core barrel and grid plate, the gas duct shell, and the hot gas duct, the candidate materials are coated or uncoated ferritic-martensitic steels (or austenitic as backup solution), other Fe-Ni-Cr-base alloys (such as Inconel 800), and Ni-base alloys. The main candidate materials for pressure vessels (reactor, energy conversion system) are 21/4 Cr steels and 9-12 Cr ferritic-martensitic steels. SiC-SiC and C-C are best candidates for non-metallic control rods structural applications.

3.5 The challenge of materials for SCWR

The Supercritical-Water-Cooled-Reactor System (SCWR) is one of the more promising Gen IV systems due to enhanced thermal efficiencies and relative compactness when compared to currently proven light water reactor (LWR) technology and supercritical fossil plant technology, with an operating pressure of 25 MPa and outlet temperature of 510 , possibly ranging up to 550. Even though many advantages of SCWR, these characteristics of increased temperature and pressure, radiation and supercritical water coolant bring into a more aggressive environment to candidate materials, structural materials research for SCWR system is concentrated in the areas of corrosion and stress corrosion cracking, radiolysis and water chemistry, dimensional and microstructural stability, strength, embrittlement and creep resistance and thermal-hydraulic.

The candidate materials for SCWR comprise austenitic stainless steel, martensite-ferritic steel and nickel-base alloy, while 9Cr steels are one of recommended candidate materials. Specially, 9Cr-1Mo-1V, which has several unique characteristics, shall be greatly given notice to.

Candidate alloys include austenitic iron (3xx series stainless steels) and nickel-base of solid solution, alloys 600, 690, 800 and precipitation-strengthened alloys625, 718, ferritic-martensitic alloys (HT-9, T-91), and oxide dispersion strengthened alloys of either ferritic or austenitic structure. Critical R&D needs will focus on the temperature range of 280-620℃ and irradiation damage dose ranges of 10-30 displacements per atom (dpa) (thermal spectrum) and 100-150 dpa (fast spectrum) and are categorized as follows).

3.6 The challenge of materials for SFR

The oxide-fueled fast Sodium-Cooled Reactor System (SFR) is proposed for Generation IV, with an outlet temperature of 550°C that is higher than previous experience with sodium-cooled reactors developed worldwide until the mid-1990s.

Therefore, an alloy with better high-temperature properties will be necessary, oxide dispersion-strengthened ferritic alloys have been proposed for this application. HT9 is an adequate cladding and duct material for the oxide-fueled (U-Pu-Zr fueled) sodium-cooled reactor with low-to-moderate core outlet temperature.

However, selection of a material with improved high-temperature strength and creep resistance will improve performance and extend the safety margin. The 12Cr ferritic steels, rather than austenitic steels, are viewed as promising structural materials for this demand.

Critical R&D needs in terms of 12Cr ferritic steels are as follows:

1) Accumulation of material strength database focusing on the creep-fatigue,
2) Improvement of toughness and ductility,
3) Welding procedure development,
4) Elevated temperature strength data of welded joints,
5) Manufacturing technology development for thick plate and thin-walled heat transfer tube.

3.7 The challenge of materials for LFR

A key challenge to the development of Lead-Cooled Fast Reactor System (LFR) is the selection of structural materials, especially in-core materials that are compatible with Pb or Pb-Bi at elevated outlet temperature operation. Some ceramic materials (such as SiC) and refractory metals have been proposed for this application, but little is known about such use of these materials.

3.8 Summary and review

Materials that meet the requirements of Generation IV systems must be identified, and databases sufficient to support design and licensing of Generation IV systems must be established. A summary of the materials options proposed for each of the concept sets is provided in Table 2.

Table 2. Structural Materials proposed for Generation IV systems.

System	Spectrum, Core outlet temperature	Structural Materials	
		In-core	Out-of-core
GFR	*Fast, 850*	*Refractory metals and alloys, Ceramics, ODS Vessel; F-M*	*Primary Cricuit;* *Ni-based Superalloys* *32Ni-25Cr-20* *Fe-12.5W-0.05C* *Ni-23Cr-18W-0.2* *CF-Mw/thermal* *Barriers Turbine:* *Ni-based alloys or ODS*

LFR	*Fast, 550 or* *Fast, 800*		*High-Si austenitics,* *Ceramics, or* *Refractory alloys*
MSR	*Thermal, 700-800*	*Ceramics, refractory metals, High-Mo Ni-base alloys(e.g.,INOR-8), Graphite, Hastelloy N*	*High-Mo Ni-base alloys(e.g.,INOR-8)*
SFR	*Fast, 550*	*F-M ducts 316SS grid plate*	*Ferritics, Austenitics*
SCWR	*Fast, 550*	*F-M(12Cr,9Cr,etc.)* *(Fe-35Ni-25Cr-0.3Ti)* *Incoloy 800,ODS* *Inconel 690,625&718*	*F-M*
VHTR	*Thermal, 1000*	*Graphites PyC, SiC, ZrC* *Vessel:F-M*	*Primary Circuit:* *Ni-based superalloys* *32Ni-25Cr-20Fe-12.5* *W-0.05CNi-23Cr-18* *W-0.2CF-M* *w/thermal barriers Turbine:* *Ni-based alloys or ODS*

F-M: Ferritic-Martensitic stainless steels (typically 9 to 12wt%Cr)

ODS: Oxide dispersion-strengthened steels (typical ferritic-martensitic)

Even for systems with different coolants, many applications have important similarities, such as temperature, stress, and neutron spectra. This suggests the opportunity to survey similar materials, or classes of materials, for use in Generation IV systems. Table 3 indicates classes of materials proposed for the system.

Table 3. Summary of classes of structural materials for Gen IV systems applications.

Syste / Item	GFR	SFR	MSR	LFR	SCWR	VHTR
Ferritic-Martensitic Stnlss Steel Alloys	*1P*	*P*	-	*P*	*P*	*S*
Austenitic Stnlss Steel Alloys	*P*	*P*	-	*P*	*P*	-
Oxide Dispersion Strengthened	*P*	*2S*	-	*P*	*S*	-

Ni-based Alloys	*P*	-	*P*	-	*S*	*P*
Graphite	-	-	*P*	-	-	*P*
Refractory Alloys	*P*	*S*	*S*	-	-	*S*
Ceramics	*P*	*S*	*S*	-	-	*P*

As the classes of materials proposed for Generation IV application are composed of a conglomerate of austenitic alloys, ferritic-martensitic alloys, ceramics, ODS materials and precipitation-hardenable Ni-base alloys, it shall be noted that a certain difficulty joins component parts during the fabrication of components. For some material classes, a suitable joining technique has not yet developed. For some of these materials, though of different compositions or form, will present similar challenges. So it is quite possible that the application of effective techniques to one material or class of materials can be extended to other materials. Therefore, an R&D task to develop broadly applicable joining techniques for Generation IV application is proposed.

The possible of materials, such as ODS alloys and ceramics, raises the requirement to establish suitable techniques for non-destructive examination. In industrial practice, semi-finished parts and joints will need to be examined. Therefore, an R&D task to develop broadly applicable non-destructive examination techniques for Generation IV application is proposed.

Because Generation IV concepts will require deployment of materials and components operating under unprecedented conditions, new codes and standards must be established to govern their use. Materials composition and property data that are collected during the development of Generation IV technologies must be obtained in accordance with quality assurance standards so that it can provide the necessary bases for codes and standards and for system licensing. Each participating country must provide experts to collectively guide the data collection and maintenance processes and to draft appropriate codes and standards.

4 INTERNATIONAL INITIATIVES

While the current level of structural materials research and development is adequate to meet the needs of currently operating nuclear power reactors, the generation IV nuclear systems have major challenges. These are sufficiently substantial in terms of the science, technology, and required resources that no one country or small group of countries will likely be able to maintain the necessary research momentum over an extended period. So there is an agreed need for an international mobilisation of resources and internationally coordinated research efforts, probably over many decades.

On March 18-21, 2002, the Department of Energy, Office of Nuclear Energy, Science, and Technology (NE) and the Office of Basic Energy Sciences (BES) sponsored a workshop to identify needs and opportunities for materials research aiming performance improvements of structural materials in higher temperature reactors. This workshop reviewed potential reactor designs and operating regimes, potential materials for application in high-temperature reactor environments, anticipated degradation mechanisms, and research necessary to understand and develop reactor materials capable of satisfactory performance while subject to irradiation damage at high temperature. The workshop gathered experts from the reactor materials and fundamental materials science communities to identify research and development needs and opportunities in providing optimum high temperature nuclear energy system structural materials.

On January 2003, U.S. Department of Energy International Nuclear Energy Research Initiative DOE/ROK, launch the Investigation Developing and Evaluating Candidate Materials for Generation IV Supercritical Water. The goal of this project is to establish candidate materials for supercritical water reactor (SCWR) designs and to initiate the evaluation of the mechanical properties, dimensional stability, and corrosion resistance. For all GEN IV designs, significant materials property data must be obtained to license future reactor designs. The program will focus on the selection and qualification of advanced materials for SCWR applications. Three materials classes have been investigated, one is ferritic and ferritic-martensitic steels, another is austenitic alloys, and the other developmental alloys such as oxide dispersion strengthened (ODS) alloys, nano-crystalline alloys, and grain boundary engineered alloys. Although the project may not perform a complete qualification of materials for SCWR application, it will generate data to support a recommendation of prime candidate materials for further evaluation and in-core testing.

Development of a web-accessible materials database for Generation IV Reactor Programs has been ongoing for about three years. This handbook provides a single authoritative source for qualified materials data applicable to all Generation IV reactor concepts. A beta version of this Gen IV Materials Handbook has been completed and is presently under evaluation. Many material types. e.g. metals, ceramics, graphite and composites and etc., are involved in Gen IV reactor development, and activities such as material selection, component design, and stress analysis will be conducted.

5 DISCUSSION AND CONCLUSION

Nuclear power technology can be ranked in stages, or generation. Generation IV systems project in-service and off-normal levels that are beyond current nuclear industry experience, as well as most previous experience with developmental systems. All of them require relatively long service lifetimes for materials and relatively high burn-up for fuels. Therefore, such essential activities as candidate selection, fabrication development, properties assessment and qualification, irradiation testing & safety demonstration might be taken into action together with system establishment of new codes, standards and regulations.

In some cases, design techniques have been mastered by scientists in laboratories but the techniques of manufacture industries still exit a big gap from the laboratories. So, manufacture capacity of candidate materials for Gen IV is needed to strengthen and acceleratory.

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