VHTR Materials R&D in KAERI

Ji Yeon Park*, Dong Jin Kim, Dae Whan Kim, Sung Ho Kim, Woo Gon Kim, Weon-Ju Kim, Eung Sun Kim, Hong Pyo Kim
Korea Atomic Energy Research Institute, P. O. Box 105, Yuseong, Daejeon 305-600, Korea
Corresponding author: jypark@kaeri.re.kr

1. Introduction

KAERI has established a plan to demonstrate a massive production of hydrogen using a very high temperature gas cooled reactor (VHTR) by the early 2020s [1]. The feasibility of nuclear hydrogen has been surveyed since 2002 and a government funded comprehensive project to develop very high temperature gas cooled reactor technology, the coated fuel technology, hydrogen production technology, and the relevant technologies started in 2004. From 2006, the Korean government started a project called "Development of Key Technologies for Nuclear Hydrogen." The main focus of the project is (1) to develop the computer codes and the databases to support the design activity of the very high temperature reactor; (2) to develop a manufacturing process and the qualification of the TRISO coated fuel particle; (3) to select and modify the materials for the very high temperatures and coupled thermo-chemical process; (4) to verify the performance of the materials in the small gas loop to be constructed; and (5) to develop the Sulfur –Iodine (SI) thermo-chemical process for a pressurized condition [2-3]. In parallel, multi-lateral GIF collaborations are being carried out. The GIF/VHTR materials R&D plan is composed of three work packages - graphite, metallic materials and composites & ceramics. These materials have been designated for the reactor sub-systems and nuclear energy system components as shown in Fig. 1 [4].

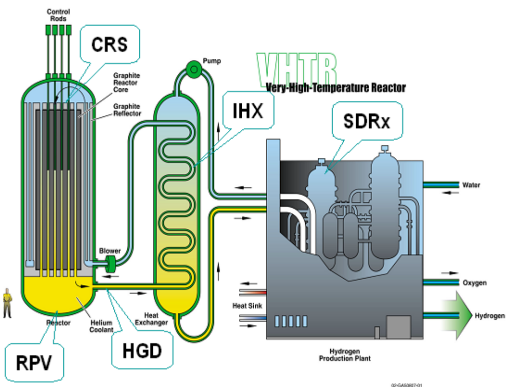

Fig. 1. The objective components of materials R&D.

There are some material issues in realizing a VHTR at 950^{o}C. Many of potential candidate materials are not currently listed in Section III of the ASME Code or have not been used in current generation nuclear plants. Most of mature metallic materials being considered have little or no margin at 900-1000^{o}C. Many of potential candidate materials lack high temperature mechanical, physical and irradiation data, and data related to high temperature helium exposure or helium contaminated with gaseous impurities. Composite materials have not been used previously for a key support system, the control rod or heat exchanger components in the nuclear plants. Therefore, selected materials for the reactor sub systems and components such as the reactor pressure vessel (RPV); the high temperature metallic core internals; the hot gas ducts and other pressure boundary components; the reactor/process intermediate heat exchanger (IHX); graphite used for the reflectors and support structures in the core region; high-temperature control rod cladding and/or the guide tube components; and ceramic materials for a thermal insulation and corrosion resistant coating; process heat exchanger materials for the SI process should be assessed and evaluated. Additionally, high-temperature design methodology improvements in the structural design methods, the materials testing and the databases, and the nuclear design codes and standards will be needed. Numerous areas of the code will have to be modified or expanded to address these new design conditions.

In this presentation, current activities on VHTR materials in KAERI will be briefly overviewed

2. Current Activities of VHTR materials R&D

Our scope for the materials R&D works is as follows; (1) material screening/selection and qualifying for the RPV, the IHX, the core structures, and process heat exchanger materials for the SI system based on the high temperature properties, the irradiation behaviors, the corrosion resistance and the manufacturability; (2) codifications of the high temperature structural design rules to extend the ASME and KS up to the very high temperature region and to support the licensing of the system design; (3) the material characterizations and database establishments; (4) the alloy modifications and developments; (5) the non-destructive evaluations (NDE); and (6) the test and optimization of the high temperature components.

Candidate materials of our researches were summarized in Table 1. Modified 9Cr-1Mo steels are considering as candidate materials for the RPV. We will select material based on the DB, codes, the fabrication capacity and cost. Alloys will be modified for a high strength and the fabrication process of the very thick forged vessel including the welding technique will be optimized. Additionally, the irradiation effects on the tensile properties and the fracture toughness will be assessed. Some mechanical properties such as a tensile/yield strength, a fracture toughness, a hardness, etc. of mod. 9Cr-1Mo were measured and the thermal aging effects at 600°C were analyzed. The creep and fatigue tests of the base metals and weldments are being performed.

Table 1. Summary of Candidate Materials for VHTR

Component	Standard Material	Type	T (℃)	Code Status
Reactor Vessel	SA533	Low alloy steel	380	ASME
	Mod.9Cr-1Mo	Ferr-Mart. steel	593	ASME 2004
IHX, Hot Duct	Hastelloy X (22Cr-18Fe-9Mo)	Ni-base alloy	900	Section II 2004
	IN 617 (22Cr-9Mo-12Co)	Ni-base alloy	982	ASME Draft Code Case
	Haynes 230 (22Cr-14W-5Co)	Ni-base alloy	900	Section II 2004
Control Rod component	X8CrNiMoNb 1616 (16Cr-16Ni-2Mo-1Nb)	Austenitic steel	650	Irradiation service
	C_f/C composite	Fiber reinforced composite	~1000	-
Reflector	**Graphite**	Nuclear grade	600	-
PHE for SI process	**Fe-Si, SiC, Alloy 800, Hastelloy, etc.**	-	500/ 800~900	-

Up to now, no super-alloy is qualified up to 950°C at a stressed condition for an expected lifetime. Moreover, many of materials (e.g., alloy 617) being considered for VHTR do not show any evidence of primary or secondary creep. For the IHX and the hot gas duct, the screening tests of Ni-based alloys and the evaluations of the high temperature properties are being performed. The creep tests of alloy 617 under 18~35 MPa at 900~975°C were finished and the He environment effects on a microstructural evolution and degradations in the mechanical properties are being estimated. A creep stress of alloy 617 was predicted by the Larson-Miller parameter method with the single n-value method and the multi n-values method at 950°C for 10^5 h. Using the multi n-values method, the higher reliable prediction data could be obtained.

Graphite will be used for the reflectors and support structures in the core region, and for the fuel elements. Experimental data for the mechanical and physical properties of selected major candidate graphites (IG-110, IG-430, NBG-18 and NBG-25) were and are being produced and the oxidation behaviors in the range of 600 to 900°C were estimated for a screening of nuclear graphites. The irradiation effects of selected graphites were evaluated using gamma-irradiation with 1.1×10^8, 1.5×10^9, and 7.0×10^9 Rad and C ions. The ASTM round-robin tests and an establishing program of the IAEA graphite DB are participating.

Ceramics and composites are being considered as candidate materials for the intermediate heat exchangers, the insulating structures, the control rods components, other internal structures, such as core restraints, belts, barrel and tie-rods. We are interested in an application of the control rods components. Evaluations of the high temperature properties such as thermal conductivities in the range of RT to 1200°C and a flexural strength of C_f/C composites (CX-270) were carried out. The oxidation behaviors and irradiation effects will be evaluated.

In order to select materials for sulfuric acid decomposition system, the corrosion behaviors and formability(ductility) of candidate materials such as Fe-Cr, Fe-Si, Fe base stainless steel, Ni base alloy, Ta, Zr, Ti, Noble metal (Pt, Au, Ir, Ru) and SiC were estimated. The corrosion tests of candidate materials in liquid sulfuric acid with a 50wt% H_2SO_4 at 125°C and a 98wt% H_2SO_4 at 320°C are carrying out by an immersion test and a polarization experiment. Additionally, corrosion tests in decomposed sulfuric acid at 850°C are also being performed.

As a member of the GIF VHTR materials project, KAERI is working on the characterization of high temperature materials for reactor vessel, IHX, PHE, hot gas duct, core graphite and carbon composite internals.

3. Conclusion

1. Experimental studies for high temperature materials are being performed on the creep and the fatigue behaviors of alloy 617 in both air and helium environments, the creep behavior of the 9Cr-1Mo base metals and weldments, the mechanical and thermal behaviors of carbon composites, the graphite oxidation and the ion accelerator irradiation tests and the corrosion behaviors of PHE materials.
2. In parallel, the GIF/VHTR materials project composed of three working groups - graphite, metallic materials and composites & ceramics is being carried out as a multi-lateral GIF collaboration.

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