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Development of advanced blanket performance under irradiation and system integration through JUPITER-II project

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ABSTRACT

The Japan–USA collaborative program, JUPITER-II, has made significant progress in a research program titled “The irradiation performance and system integration of advanced blanket” through a six-year plan for 2001–2006. The scientific concept of this program is to study the elemental technology in macroscopic system integration for advanced fusion blankets based on an understanding of the relevant mechanics at the microscopic level. The program has four main research emphases:

(1) Flibe molten salt system: Flibe handling, reduction–oxidation control by Be and Flibe tritium chemistry; thermofluid flow simulation experiment and numerical analysis.
(2) Vanadium /Li system: MHD ceramics coating of vanadium alloys and compatibility with Li; neutron irradiation experiment in Li capsule and radiation creep.
(3) SiC/He system: Fabrication of advanced composites and property evaluation; thermomechanics of SiC system with solid breeding materials; neutron irradiation experiment in He capsule at high temperatures.
(4) Blanket system modeling: Design-based integration modeling of Flibe system and V/Li system; multi-scale materials system modeling including He effects.

This paper describes the perspective of the program including the historical background, the organization and facilities, and the task objectives. Important recent results are reviewed.

1. Introduction

This report describes an outline of the activities of the JUPITER-II collaboration (Japan–USA Program of Irradiation/Integration Test for Fusion Research-II), which has been carried out through six years (2001–2006) under Phase 4 of the collaboration implemented by Amendment 4 of Annex I to the DOE (United States Department of Energy)-MEXT (Ministry of Education, Culture, Sports, Science and Technology) Cooperation. This program followed the RTNS-II Program (Phase 1: 1982–1986), the FFTF/MOTA Program (Phase 2: 1987–1994) and the JUPITER Program (Phase 3: 1995–2000) [1].

In the RTNS-II Program the fundamental mechanism of radiation damage was emphasized and “cascade effects” of neutron irradiation were clarified at low fluence levels by using a fusion relevant 14 MeV neutron source. In the FFTF/MOTA Program, on the other hand, the microstructural development and property changes

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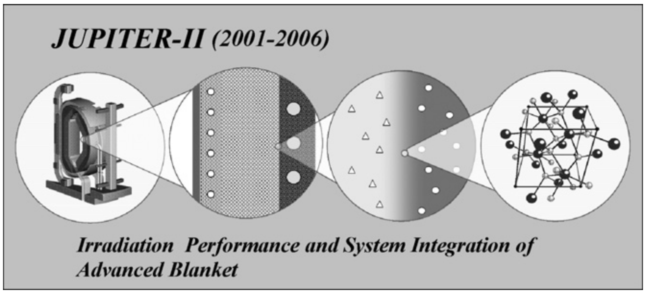

Fig. 1. Schematic concept of JUPITER-II Program.

caused by relatively high dose levels up to about 100 dpa were studied using a materials irradiation capsule in a fast reactor.

In the JUPITER Program, in order to study the dynamic behavior of fusion reactor materials under irradiation and their response to variable and complex irradiation conditions, a program of systematic irradiation experiments utilizing fission neutrons at HFIR and ATR reactors has been performed. The irradiation experiments in this program included low activation structural materials, functional ceramics and other innovative materials.

Throughout these programs, the irradiation resistance of various structural and functional materials was surveyed during and after neutron irradiation. These basic understandings of irradiation performance, as well as new developments in alloy technology, led to the prospect that ferritic steels, vanadium alloys and SiC/SiC composite became candidates for low activation structural materials and led naturally to a new research theme of materials system integration for advanced fusion blankets.

The JUPITER-II collaboration was established to provide the scientific foundations for understanding the integrated behavior of blanket material combinations operating under conditions characteristic of fusion reactors, including interactive neutron irradiation effects, high-temperature coolant flow phenomena, heat and mass transport in blanket materials, and coolant chemistry and its interactions with surrounding materials. The scientific concept of this program is to study the elemental technology in macroscopic system integration for advanced blanket based on an understanding of the relevant mechanics at the microscopic level, as schematically illustrated in Fig. 1 [2].

Outcomes from this program have been presented at international fusion oriented meetings, such as ICFRM and ISFNT, and published in related articles. Technical results of each research subjects in detail and full lists of publication will be summarized in a Special Report of the National Institute for Fusion Science [3]. The overall perspective of the program and typical references are described in this paper.

2. Objectives and tasks

2.1. Advanced blanket system and integration issues

Fusion reactor blankets must successfully generate and recover tritium, function as radiation shielding, and convert the kinetic energy of fast neutrons to usable heat. Advanced blankets are also required to satisfy requirements for thermal efficiency and low induced radioactivity. There are several advanced blanket concepts that combine various breeding/cooling materials and low activation structural materials [4,5]. Structural materials considered are ferritic steels, vanadium alloys and SiC/SiC composite instead of austenitic steels. Breeding materials studied are Li ceramics, liquid metals (Li, Li–Pb), and molten salt ($LiF–BeF_2$, Flibe). Coolant materials considered are water, He gas, liquid metal and molten salt. The advanced blanket systems studied in this program are (a) Flibe molten salt system [6], (b) Vanadium alloys with Li [7], and (c) SiC/SiC with He [8].

The Flibe system design avoids magneto-hydrodynamic (MHD) problems due to its low electrical conductivity and it has relatively low reactivity with oxygen and water. However, the low thermal conductivity of Flibe limits the power density, and its low tritium breeding ratio requires a neutron multiplier Be like solid breeder. Reduction–oxidation (Redox) control and corrosion problems involving the structural materials must also be solved.

The liquid lithium breeder concept has high thermal conductivity and a high tritium breeding ratio. If lithium is combined with vanadium alloys, relatively simple blanket design with high thermal efficiency and low induced radioactivity is possible. However, MHD problems must be solved, and liquid lithium and vanadium alloy technology must be further developed.

SiC/SiC composite as structural material has relatively lower induced radioactivity and can be used at very high temperatures with He gas cooling. Using solid breeding materials, a blanket design with high thermal efficiency and low induced radioactivity is proposed. Issues to be solved are irradiation performance for thermal conductivity degradation under high concentration of He of the composite and thermomechanics of breeding materials combination. And the manufacture technology for SiC/SiC composites must be further developed.

Table 1 summarizes characteristics and issues for these three types of blanket systems. Irradiation performance of materials system and related tritium issues are common to all systems.

2.2. Tasks and objectives

In the program, the research theme was categorized and research objectives are clarified below.

Task 1 Self-cooled liquid blanket

1-1 Flibe cooled.

1-1A Flibe handling/tritium chemistry.

Technical requirements necessary to apply Flibe to a self-cooled liquid blanket of a fusion reactor were surveyed. Maintaining Flibe under a reducing atmosphere is a key issue to transform TF to T_2 with a faster reaction rate compared with the residence time in blanket. The purpose of the task was to clarify whether or not the Redox control of Flibe can be achieved with Be through the reaction ($Be + 2TF \rightarrow BeF_2 + T_2$).

1-1B Flibe thermofluid flow simulation.

The thermal conductivity of Flibe is low, and its kinematic viscosity is high compared to other lithium-containing metal alloys. The high viscosity and low thermal conductivity put Flibe in the class of high Prandtl number fluids. In order to obtain sufficiently large heat transfer for such a fluid coolant, high turbulence is required under magnetic field. The effect of magnetic field to the turbulence is an important issue for heat transfer.

1-2 Lithium cooled with V-alloy structure.

1-2A Coatings for MHD reduction.

In order to reduce the pressure drop from the MHD force assisted with liquid lithium coolant flowing through magnetic

Table 1
Characteristics of advanced blanket systems studied in JUPITER-II Program [6–8].

System	Flibe system	Vanadium alloys/Li	SiC/SiC /He
Candidate structural material	Ferritic, ODS vanadium alloy	(Vanadium alloy) V–4Cr–4Ti	SiC/SiC composite
Coolant	Flibe (He)	Liquid Li	He
Breeding materials	Flibe	Li	Li_2O etc.
Typical blanket design	FFHR liquid blanket	ARIES-RS liquid blanket	DREAM gas cooled blanket
Activation	Medium low	Low	Very low
Inlet/outlet	450/550 (700) °C	330/610 °C	500/800 °C
Temperature and heat flux	Medium	High	Medium
Thermal efficiency	37%	45%	50%
Issues for power reactor	Flibe technology Redox control	MHD drop Li technology	Thermal conductivity hermeticity
Materials system issues	Corrosion	Ceramic coating fabrication	H, He production fabrication

fields, conductive structural materials must be coated with insulating ceramics. The purpose of the task is to find stable ceramics in high-temperature lithium, to coat the layer on to vanadium alloys and to evaluate its stability under irradiation.

1-2B V-alloy capsule irradiation.

The candidate vanadium alloy is V–4Ti–4Cr, whose irradiation creep properties and compatibility with lithium are important for high-temperature applications. Related mechanical properties and their improvement by alloy modification and heat treatment are also important. The purpose of the task is to obtain information on property changes caused by neutron irradiation in the lithium capsule.

Task 2 High-temperature gas-cooled blanket.

2-1 SiC fundamental issues, fabrication, and materials supply.

2-2 SiC system thermomechanics.

2-3 SiC capsule irradiation.

The purpose of Task 2 is to demonstrate the feasibility of high efficiency gas cooling blanket systems using advanced SiC/SiC composites based on new material development strategy and thermomechanical properties and irradiation resistance of the materials system. Three subtasks are arranged systematically. In Task 2-1, the research subjects are fundamental issues, fabrication, and materials supply of SiC/SiC composites. The combination of mechanical integrity and thermal properties is focused. In Task 2-2, thermomechanics, compatibility and heat transfer performance of gas blanket system were studied. The thermomechanical interaction between the lithium-oxide pebble bed and SiC materials must be known. Radiation behavior of the advanced SiC/SiC composites and solid breeding system in high-temperature environment were studied in Task 2-3. The properties of composite and also the behavior of matrix, fiber and interface after high-temperature irradiation and effects of H and He are important.

Task 3 Blanket system modeling.

3-1 Design-based integration modeling.

The main purpose of this task is to develop the integrated engineering model of blanket systems and to make clear key issues in each task in this project from the point of view of blanket systems based on fusion reactor designs. For this purpose, it is essential to optimize the blanket system not as a single component but as an integrated system device under the boundary conditions of burning core plasmas and out-vessel environments. Therefore, collaboration to connecting each task is very important.

3-2 Materials system modeling.

In order to understand the microscopic behavior of materials under fusion relevant irradiation conditions, multiscale modeling of microstructural evolution including He effects and interface must be studied. Extension to macroscopic behavior will require combination with the finite element method (FEM) analysis.

3. Managing structure and facilities

3.1. Managing system

The management structure continued with a Steering Committee consisting of a Representative (Japan: K. Abe, US: S. Berk/G. Nardella) and two Program Coordinators (Japan: A. Kohyama, S. Tanaka, US: S.J. Zinkle, D.K. Sze) for each side, with five Task Coordinators for each side, and with nine Subtask Coordinators for each side. In addition, there were six Liaison Officers for the U.S. side and four for the Japanese side, their purpose being to facilitate communications and assist in the coordination of collaboration activities. Tasks and coordinators are shown in Table 2. The whole program plan and research schedule were discussed at annual Steering Committee meetings. Research assignments and appropriate workshops were arranged for each task. The Personnel Assignment Guidelines for the JUPITER-II Collaboration, developed by the Steering Committee in the second year, continued to contribute toward the fruitfulness and productivity of assignments from both sides.

3.2. Facilities

Fig. 2 shows the main facilities used and flows of test specimens in this program. Flibe experiments under the JUPITER-II Program were performed in the Safety and Tritium Applied Research Facility (STAR) at the Idaho National Laboratory. These experiments included production, purification, sampling, analysis and Redox control of Flibe. STAR is a unique facility for systematic experiments using Flibe and related tritium work.

Thermofluid simulation experiments of Flibe were done using the FLiHy Loop with and without magnetic field at UCLA. Simulation experiments with heat transfer salt were done using the TNT (Tohoku-NIFS-Thermofluid) loop at Tohoku University.

Neutron irradiation experiments were done using the High Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory (ORNL). The HFIR provides both a high flux of fast neutrons to produce displacement damage and a high flux of thermal neutrons to produce helium and hydrogen through (n,a) and (n,p) reaction. Complementary irradiation tests were done using the fast experiment reactor

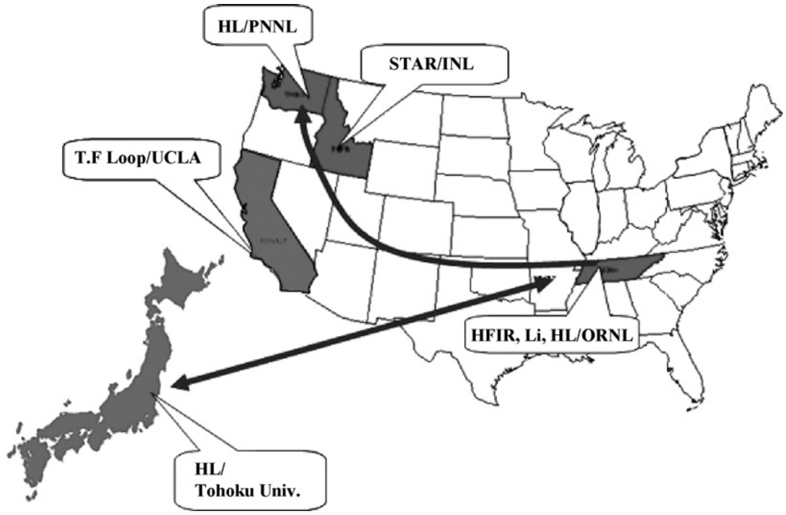

Fig. 2. Facilities used in JUPITER-II Program.

(JOYO) and Japan Material Test Reactor (JMTR) at Japan Atomic Energy Agency (JAEA). Post-irradiation experiments (PIE) were performed at the hot laboratories (HL) at ORNL, and also at Pacific Northwest National Laboratory (PNNL) in the early stages of the program. Subsidiary tests were performed at hot cells of Oarai Branch of IMR, Tohoku University. Lithium experiments for ceramic coating of vanadium were performed at ORNL.

Fabrication and testing of SiC/SiC composite were done using the High Temperature Materials Laboratory (HTML) at ORNL and at the Institute of Advanced Energy of Kyoto University. Thermomechanics experiments of SiC and breeding materials were done at UCLA.

Other facilities used for experiments in Japan were the DUET facility at Kyoto University, the Cyclotron and Dynamitron accelerators at Tohoku University, the creep apparatus at NIFS, the molten salt apparatus at the University of Tokyo, etc.

4. Highlights of accomplishments

4.1. Flibe system

4.1.1. Flibe chemistry [9–12]

The Redox control of Flibe by Be was successfully proved experimentally, and the prompt transformation of HF to H_2 was completed in the Flibe crucible. Flibe handling and purification method was tested and an experimental apparatus to investigate the Be Redox control of Flibe was prepared. Recent results are summarized below.

For the subtask on molten salt handling and tritium/chemistry control, hydro-fluorination was demonstrated for reducing impurities in Flibe to acceptable levels with the resulting fluorides either precipitating or remaining stable in the salt. Many practical lessons were learned while operating systems using Flibe, such as the need for in-line filtration to remove large carbon particles, splatter shields to prevent vapor species transport which tends to plug gas purge exhaust lines, and coating of all system components with Ni to minimize corrosion from salt or salt vapor contact.

Mobilization studies were performed to quantify salt vapor species at temperatures lower than those of previous studies, producing results in the appropriate temperature range of a fusion blanket system. The predominant vapor species was verified to be BeF_2, however the quantity observed was 3–5 times less than values expected from extrapolated high-temperature data or model predictions. Concentrations of Li components also increased with temperature, reflecting a possible change in the vaporizing gas species.

Permeation studies performed with deuterium or tritium in Flibe provided data on mass transport parameters of solubility and diffusivity—key parameters required for the selection and design of blanket tritium extraction techniques. Two different experiment systems were developed, varying the contact surface of the salt and container with the permeating species. Diffusivity of deuterium in liquid Flibe was measured to be marginally larger than that for solid Flibe, whereas the activation energy of diffusion was similar in value to F+ diffusion.

A major objective of this subtask was to demonstrate Redox control with excess Be introduced into the Flibe, allowing the energetically preferred formation of BeF_2 and recombination of molecular tritium. Experiments were performed allowing introduction of HF into salt exposed to metallic Be samples, followed by quantifying the rate of HF conversion. Rapid and effective reduction of HF occurred at HF concentrations much higher compared to those expected in a blanket system; without neutrons to generate T+, low F-concentrations are practically unachievable. However, modeling of the chemical system assisted in calculating rate coefficients that can be used to predict system behavior at the very low concentrations applicable to the blanket. An unexpected result obtained from the experiment series concerns the mechanism of Be transport into the salt. Rather than dissolution of Be metal, which is known to have very low solubility in Flibe, the transport appears driven by galvanic action. This discovery can potentially influence the design of the chemical control system for a blanket.

Corrosion tests were also performed with and without Redox control for samples of the advanced ferritic steel JLF-1. Exposures of up to 500 h in the non-flowing molten salt indicate a corrosion rate of 0.1 μm/h, with a Cr-rich oxide layer forming at the sample interface. Corrosion was effectively halted during periods of Redox control using Be.

4.1.2. Flibe thermofluid [13,14]

In the first half of the program, thermofluid flow experiment without magnetic field was performed. Selection of stimulant fluid (30–40% KOH aqueous solution) and reconstruction of FLiHy loop with enough length test section were completed. Development of particle image velocimetry (PIV) for non-MHD pipe flow was performed. Direct numerical simulation (DNS) code was improved and MHD turbulence model was developed.

Table 2
Research tasks and coordinators in JUPITER-II Program.

Task			Japan		US	
			Task Coordinators	Deputy	Task Coordinators	Deputy
Task 1: self-cooled liquid blanket	1-1: Flibe cooled	1-1-A: Flibe handling/tritium chemistry/safety	T. Terai	K. Okuno/M. Nishikawa	D.A. Petti	R.A. Anderl
		1-1-B: Flibe termofluid flow simulation	T. Kunugi	T. Kunugi	M.A. Abdou	N. Morley
	1-2: Li cooled with V-alloy structure	1-2-A: coatings for MHD reduction	T. Muroga	T. Muroga	R.J. Kurtz	B.A. Pint
		1-2-B: V-alloy capsule irradiation		H. Matsui		G.R. Odette
Task 2: high-temperature gas-cooled blanket	2-1: SiC fundamental issues, fabrication, and material supply	A. Hasegawa	T. Hinoki	L.L. Snead	Y. Katoh	
	2-2: SiC system thermomechanics	A. Shimizu		A. Ying		
	2-3: SiC capsule irradiation	A. Hasegawa		L.L. Snead		
Task 3: blanket system modeling	3-1: design-based integration modeling	A. Sagara	H. Hashizume	N.M. Ghoniem	D. Sze	
	3-2: material system modeling	N. Sekimura		R.E. Stoller		

In the second half of the program, the thermofluid flow experiment was extended to the condition with magnetic field. Velocity profile and turbulence quantities are measured under magnetic field. Heat transfer and temperature profile measurement was performed by developing thermocouple tower. Heat transfer tests at high parameter range were performed successfully to establish a database for MHD flow and heat transfer and to validate the MHD turbulence model.

As for flow characteristics, DNS results and PIV measurement regarding the mean velocity and the turbulent statistics of turbulent pipe flows showed very good agreement, and the laminarization of turbulent flow due to the Lorentz force was clearly observed in both DNS and PIV experiments. Therefore, a large database for turbulent flow with and without magnetic field was established and can be used for developing the non-MHD and MHD turbulence models.

As for heat transfer characteristics, temperature profiles with conducting wall were measured for lower Re number flow, and the effect of a B-field on temperature fluctuation was clearly observed. Effect of thermal stratification on heat transfer performance was examined and temperature profile was generated by interaction between laminarization due to B effect and thermal stratification. It was found that the degradation of heat transfer performance due to B-field is larger than the conventional prediction regarding low Pr number fluid.

4.2. Vanadium/Li system

4.2.1. MHD coating [15,16]

For the subtask on coatings for vanadium alloys to reduce the pressure drop from the magneto-hydrodynamic force associated with liquid lithium coolant flowing through magnetic fields, four important results were obtained. (1) Identification of leading candidate ceramics: Exploration of MHD coating candidates was carried out by Li exposure tests of bulk ceramics. Ceramics of Er_2O_3 and Y_2O_3 showed good stability in Li. Later, reaction layer was observed in Y_2O_3. Er_2O_3 is shown to be the promising new candidates. (2) Development of coating with sufficient stability in static Li: Various coating technologies were applied for fabrication of Er_2O_3 coating on V–4Cr–4Ti substrate. High crystalline Er_2O_3 coating showed good stability in Li to 973 K at 1000 h. (3) Development of two-layer coatings with sufficient stability in static Li: In situ resistivity measurements in Li for V/Er_2O_3/V-alloy substrate systems showed satisfactory resistivity in molten Li to 873 K. This two-layer coating showed good performance in static Li. (The two-layer coating is to coat the V-alloy with an insulator and then have another thin V-alloy coating outside the insulator to protect the insulating coating from Li corrosion.) (4) Compatibility in flowing Li: Using a stainless steel loop at 873 K for 1000 h, no significant damage was observed for Er_2O_3 bulk specimens and weight loss was small. Compatibility of the two-layer coating in flowing Li at 973 K was tested in ORNL using a loop made of vanadium alloy.

The electrical resistance of the coated specimens in contact with Li was acceptable up to 600 °C and was not degraded by repeated temperature cycling between RT and 600 °C in static Li. Bulk specimens of Er_2O_3, Y_2O_3 and AlN irradiated at 450 °C in Li environment in HFIR-RB-17J were examined and no signification corrosion was observed.

4.2.2. Li capsule irradiation [17–19]

Performance of V-alloy components in Li and irradiation environments up to 3–5 dpa at 450 and 600 °C was clarified as follows. (1) Regarding the effect of Li environment on thermal creep performance of V–4Cr–4Ti, creep rate was lower in Li than in vacuum. (2) Regarding irradiation creep in Li, comparison with Na capsule irradiation in JOYO reactor showed insignificant difference in irradiation creep between Li and Na environments. (3) Regarding improvement of irradiation properties of V–4Cr–4Ti by doping with Si, Al, and Y, higher ductility after irradiation was confirmed for specimens in irradiated in HFIR-17J at 450 °C to 5 dpa. Microstructures showed retarded dislocation evolution by the addition of Si, Al, and Y.

Recent results are as follows: Measurement of diametral strains of pressurized creep tubes irradiated at 425 °C was completed. The results showed moderate irradiation creep strain with only small differences between specimens prepared from the US-Heat and NIFS-Heat of V–4Cr–4Ti. The results are also consistent with those obtained for specimens irradiated in a Na environment in the JOYO reactor. Measurement and analysis of the creep tubes irradiated at 600 °C has almost been completed.

The effect of Y, Al, and Si additions on microstructure and tensile properties was investigated. Suppression of irradiation-induced hardening and loss of ductility by addition of Y, Al, and Si was confirmed. Charpy impact tests are planned for an identical set of specimens. Shipping of the specimens irradiated at 425 °C to Japan was nearly completed.

Research activity continued to study enhancement of technology for testing small fracture and deformation specimens at variable loading rates for the application to HFIR-RB-17J irradiated specimens. The PIE of HFIR-RB-17J specimens includes the effect of trace elements on the microstructure and properties of vanadium alloys, characterization of laser weld joints and advanced alloys, measurement of radiation-induced fracture transition temperature shifts using disc compact tension and pre-cracked bend bar specimens, and determination of stress-state effects on flow localization in V–4Cr–4Ti.

4.3. SiC/He system

4.3.1. SiC fabrication [20–22]

Extensive R&D efforts for development and characterization of advanced SiC/SiC composites for fusion have been successfully carried out in Task 2-1. Advanced small specimen test technologies and procedures for mechanical properties of SiC/SiC were developed. Joining and hermetic sealing using NITE SiC/SiC and CVI/NITE hybrid process has been explored. The robust joining technique using NITE joint was successfully developed and evaluated. The optimized composite samples were produced and supplied to irradiation experiments. Guidance was provided for selection of appropriate materials for the pebble bed thermomechanics experiments.

Recently, progress has been achieved in the areas of fracture toughness evaluation and interfacial shear properties characterization for advanced SiC/SiC composites, and chemical compatibility of SiC in dual-cooled blankets. The fracture behavior of advanced SiC fiber composites with chemically vapor-infiltrated (CVI) and nano-infiltration and transient eutectic-phase (NITE) SiC matrices were successfully evaluated by bend testing of single edge notched beam (SENB) specimens. Based on that, J-integral analysis of SENB fracture was selected as a standard test method for post-irradiation examination of 18J specimens. For the interfacial characterization, methods to explicitly determine debond and frictional shear stresses at fiber/matrix interfaces by single fiber push-out experiments were developed. Additionally, static compatibility testing of high purity monolithic SiC and advanced SiC/SiC composite materials in liquid lead-lithium was performed in support of dual-cooled blanket R&D, providing another promise of those material systems.

4.3.2. Thermomechanics [23]

For the subtask on helium-cooled SiC/SiC composite and solid breeder pebble bed blanket system thermomechanics, experiments were performed for the Li_2TiO_3 pebble bed using a batch of Li_2TiO_3 pebbles not previously investigated. Specifically, CVD-SiC clad deformation-time histories were recorded using a laser position sensor. The thermomechanical interaction between this batch of pebbles and CVD-SiC clad appeared less significant as compared to what happened to the previous beds of Li_4SiO_4 pebbles, where hundreds of micrometers deformations have been recorded. Here the deformation dropped below twenty microns because of a lower thermal expansion coefficient of metatitanate pebbles, and therefore a lower differential thermal stress at the pebble/SiC interface. A lower differential thermal stress is preferable for blanket operations, and thus Li_2TiO_3 pebbles are more compatible for use in high-temperature SiC/SiC blankets. Stress relaxation of the bed was found to be fast even when initially applied stresses were low such as around 0.2 MPa.

To understand and develop predictive capability of thermomechanics concerning a structure/ceramic breeder pebble bed material system, experimental data from Task 2-2 confirmed that a different set of consecutive equations other than those obtained from the uni-axial experimental results was needed to better describe ceramic breeder thermomechanical properties under prototypical loading conditions. The design based on the DREAM concept reduces the above concern, where heat generated inside the breeder bed is removed by the forced convection (rather than conduction as in the ARIES-I case.) However, a cost-effective tritium extraction technology from the high-temperature helium stream is needed in order to make this concept attractive.

4.3.3. SiC capsule irradiation [24,25]

In Task 2-3, various PIE and analysis of HFIR-14J experiment were carried out to establish radiation resistant design strategy of SiC/SiC composites. Screening irradiation in HFIR rabbit was carried out on the new developed SiC/SiC composited to confirm the material selection. Supporting experiments in JMTR and JOYO, and accelerator irradiation were conducted to ensure He/H effect on microstructure development, mechanical properties of the advanced composites and monolithic SiC. Results of the rabbit and supporting irradiation experiments showed the advanced SiC/SiC composite was expected to have enough radiation resistance at high-temperature region. The HFIR-18J irradiation rig for high-temperature irradiation experiments was fabricated and irradiation experiment has to be finished in early 2008.

Other accomplishments include the completion of post-irradiation examination of tensile specimens from the screening rabbit irradiation campaign. Results confirmed the strength retention of advanced SiC fiber, CVI SiC matrix composites after irradiation at below 1000 °C, and also implied potential degradation in ultimate tensile strength after irradiation above about 1100 °C.

4.4. Design and modeling

4.4.1. Design-based integration modeling [26–28]

The main works performed in this task are as follows:

(1) Flibe blanket system (Task 1-1-A):
- Design and requirements on tritium recovery system.
- Proposal and key issues on tritium permeation.
- Design and evaluation of Flibe/V blanket.

(2) Thermofluid of Flibe (Task 1-1-B):
- Requirements on the first wall condition for FFHR.
- Modeling of MHD effects on heat-transfer.
- Enhancement of heat-transfer efficiency.

(3) Li/V blanket (Task 1-2-A):
- Evaluation and requirements on MHD coating.
- Design and evaluation of self-cooled Be-free Li/V blanket.

(4) All tasks:
- Improvement of neutronics calculation system for helical structure.

Recent results for the subtask on design-based integration modeling are as follows. The peaking factor of neutron wall loading in helical reactor design FFHR has been evaluated using the 3D Monte-Carlo neutron transport code, MCNP-4C, which has been newly improved for a non-axisymmetric helical system in order to enable frequent modification of blanket designs and to quickly check neutronics performance. The peaking factor evaluated is fairly low as 1.2 on a simplified model with a helical neutron source with typical profiles of plasma density and temperature. A new concept for a Flibe cooled vanadium alloy blanket has been proposed to increase the outlet operation temperature and ΔT (temperature difference between structure and Flibe). Thermodynamics analysis has revealed that, although the conventional Redox control with Be (TF to T2) results in an unacceptably high T inventory in V-alloy, another Redox control (T2 to TF) by MoF6 or WF6 doping works well with a corrosion protective Mo or W coating on V-alloy. However, tritium recovery in the form of TF is the key issue. As for the conventional Be in Flibe, the thermodynamics analysis proves that the Redox control can be achieved even under the TF concentration of 10 Pa expected in a steady-state fusion power of 1 GW in such as FFHR2. Regarding T recovery using Flibe-He counter-current extraction tower of 5 m height, the T decontamination factor estimated is sufficiently high around 10^6 at the outlet. T permeation through tube walls in a heat exchanger is the key issue.

4.4.2. Material system modeling [29,30]

The topics studied were as follows: (1) code development for evaluating atomic displacements due to neutron irradiations, (2) multiscale modeling of helium effects in irradiated materials, (3) interface damage modeling in irradiated alloys and compounds, (4) modeling of microstructural evolution of vanadium alloys under irradiation, (5) modeling of radiation damage in advanced SiC/SiC composites, (6) modeling of mechanical deformation through multiscale simulations, (7) the development of MD-FEM combination methodology, (8) calculation of ideal interfacial strength between vanadium and oxide ceramics, and (9) information technology for modeling and integration of data and model.

Recent results for the subtask on material system modeling were as follows. Multiscale modeling of microstructural evolution in reduced activation ferritic–martensitic steels during irradiation was done using molecular dynamics (MD) and kinetic Monte Carlo (KMC) simulation techniques. The nucleation path of He bubbles in metals under wide irradiation conditions was clarified. This effort can clearly explain a difference in formation mechanisms between He bubbles in fusion first wall materials where He is produced by (n, α) nuclear transmutation reactions and those in fusion divertor materials where He is directly implanted. Energetics of lattice defects in β-SiC was investigated using an MD technique with empirical interatomic potentials, which will be used to evaluate the formation kinetics of defect clusters in ceramics during irradiation. A new modeling method linking different size-scales with MD and the finite element method was developed to investigate dislocation movements and crack initiation and propagation in materials. This effort can be applied to clarify the mechanism of radiation-induced mechanical property changes of irradiated materials. A detailed Kinetic Rate Theory based Helium bubble evolution code (HEROS) was developed to describe bubble growth and gas release from helium implanted tungsten. A knowledge-based fusion materials database, Fusion NET, was developed for automated archiving of materials property data and rendering it in a web-based interactive form that is convenient for structural designers and analysts. For example, data and properties for F82H have been assembled and stored in the database. This is being applied to the design and analysis of test blanket modules for ITER.

5. Summary and conclusion

The Japan–USA collaborative program, JUPITER-II, has made significant progress in a six-year research program titled “The irradiation performance and system integration of advanced blanket” through a six-year plan for 2001–2006. The scientific concept of this program is to study the elemental technology in macroscopic system integration for advanced fusion blankets based on an understanding of the relevant mechanics at the microscopic level. The program completed its six-year plan with important progress toward its objectives and with significant scientific accomplishments that address key issues for several attractive blanket systems of common international interest The types of systematic experimental and theoretical studies of blanket materials conducted under JUPITER-II are important and essential elements toward realizing attractive fusion energy options.

The program had four main research emphases, as follows:

(1) Flibe system: It was shown that Redox (reduction–oxidation) control of Flibe by Be is feasible. Thermofluid flow simulation and experiment and numerical analysis demonstrated the MHD effects on turbulence and heat transfer. This will provide database for thermal analysis for a Flibe based fusion blanket.
(2) Vanadium/Li system: MHD ceramics coating of vanadium alloys and compatibility with Li was verified successfully. Neutron irradiation experiments in Li capsule and radiation creep experiments showed the radiation resistance of the candidate alloys.
(3) SiC/He system: Fabrication of advanced composites with high thermal conductivity was performed successfully and used for neutron irradiation experiment in He capsule at high temperatures to show good radiation resistance. Thermomechanics of SiC system with solid breeding materials were measured.
(4) Blanket system modeling: Design-based integration modeling of Flibe system and V/Li system improved substantially. Multiscale materials system modeling including He effect progressed.

In conclusion, for the three blanket systems studied, technical feasibility was shown from the viewpoint of their original issues. Remaining issues are tritium behavior and related irradiation performance of materials system in advanced blankets. Some of these issues are being studied now in Phase 5 of the Japan–USA Collaboration, called the TITAN Program.

Through this program, many young scientists including doctoral students contributed to the success of the task research work. It is expected that fusion research and developments for the future will be strengthened by these researchers.

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