MIDTERM SUMMARY OF JAPAN-US FUSION COOPERATION PROGRAM TITAN

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Japan-US cooperation program TITAN (Tritium, Irradiation and Thermofluid for America and Nippon) started in April 2007 as 6-year project. This is the summary report at the midterm of the project. Historical overview of the Japan-US cooperation programs and direction of the TITAN project in its second half are presented in addition to the technical highlights.

I. PROGRESS IN JAPAN-US COLLABORATION

I.A. RTNS-II, FFTF/MOTA and JUPITER Projects

The Japan-US collaboration was started in FY(Japanese Fiscal Year starting April) 1981, and the first phase was the RTNS-II (Rotating Target Neutron Source-II) project[1], followed by the second phase known as the FFTF/MOTA (Fast Flux Test Facility/Materials Open Test Assembly) project[1] from FY 1987. In the RTNS-II program, fusion neutron irradiation experiments using the D-T neutron source of the U.S.A. were carried out. Though the maximum irradiation dose was limited to about 0.1 dpa, Japanese advanced technologies on microstructure observation by electron microscopy and material strength evaluation by testing miniaturized specimens were utilized to obtain numerous highly precise data on the generation and accumulation processes of defects induced by the D-T neutrons as well as on the microstructure-strength correlation. Irradiation of breeding materials, superconducting materials and device materials were also carried out in the project.

The main theme in the FFTF/MOTA project was to explore the effects of high-dose neutron irradiation up to 100 dpa, which was the expected life time for fusion reactor materials. A more fundamental approach, compared to conventional irradiation tests, was promoted and a thorough understanding of the entire microstructural evolution process under irradiation covering low dose to high dose irradiation regimes was obtained.

The irradiation experiments in the above projects were performed under relatively simple and steady irradiation conditions. The JUPITER project[1,2], started in FY 1995, focused on dynamic radiation effects detectable only during irradiation and effects of variable and combined irradiation conditions.[2] This included radiation-induced conductivity of ceramic insulators, varying temperature irradiation effects, and combined transmutation effects. These experiments extracted valuable information on materials response in fusion relevant environments.

I.B. JUPITER-II Project

The projects before JUPITER-II[3] focused on materials performance under irradiation and materials development based on the knowledge gained from the irradiation experiments. In fusion blankets, materials will be used as joined or coated materials (materials systems). In addition, the use of materials in blankets involves new issues associated with the interaction with breeders and coolants. Thus basic technological studies and their integration are essential. The issues are better specified with the progress in the blanket design.

JUPITER-II[3] project aimed at (1) developing key technology for fabrication and operation of “self-cooled liquid blankets” and “high temperature gas-cooled blankets” which are combinations of low activation structural materials and breeder/coolants which are capable of high tritium breeding ratio and high coolant exit temperature, and (2) evaluation of the irradiation performance of the materials system which is a key feasibility issue for blanket development. Overall evaluation of blankets and materials systems based on modeling, and orientation of the development towards the commercialization of the system were also carried out in this project.[3]

II. OUTLINE OF TITAN PROJECT

II.A. Objectives

Blankets are component systems whose principal functions are extraction of heat and tritium. Thus it is crucial to clarify the potentiality for controlling heat and tritium flow throughout the first wall, blanket and out-of-vessel recovery systems. The TITAN project continues the JUPITER-II activity but extends its scope including the first wall and the recovery systems with the title of "Tritium and thermofluid control for magnetic and inertial confinement systems". The objective of the program is to clarify the mechanisms of tritium and heat transfer throughout the first-wall, the blanket and the heat/tritium recovery systems under specific conditions to fusion such as irradiation, high heat flux, circulation and high magnetic fields. Based on integrated models, the breeding, transfer, inventory of tritium and heat extraction properties will be evaluated for some representative liquid breeder blankets and the necessary database will be obtained for focused research in the future.

TABLE I Task structure of TITAN project

Task	Subtask	Facility	Goal
Task 1 Transport phenomena	1-1 Tritium and mass transfer in first wall	TPE PISCES	Mass transfer and tritium inventory in first wall, and tritium transfer between first wall and blanket
	1-2 Tritium behavior in blanket systems	STAR	Tritium transfer in blanket elements
	1-3 Flow control and thermofluid modeling	MTOR	Thermofluid in high magnetic field and comparison between experiments and modeling
Task 2 Irradiation synergism	2-1 Irradiation-tritium synergism	HFIR STAR	Neutron irradiation effects on tritium transfer in first wall and structural materials
	2-2 Joining and coating integrity	HFIR	Radiation response of joint and coated materials, and synergistic tritium and helium effects
	2-3 Dynamic deformation	HFIR	Deformation of structural materials during irradiation, and effects of tritium and helium production
Common Task System integration modeling	MFE/IFE system integration modeling		Integrate modeling of mass and heat transfer in first wall, blanket and recovery system, and contribution to reactor system design

II.B. Task Structure, Research Subjects and Facilities

Table 1 summarizes the task structure of TITAN project. The following US facilities are used.

(a) STAR (Safety and Tritium Applied Research) [INL]

Established at the Idaho National Laboratory in 2001, with an allowable tritium inventory of 16,000 Ci, STAR includes the Tritium Plasma Experiment (TPE) and various facilities for testing tritium behavior in blanket conditions. Unique features of the facility include use of Be and neutron-irradiated materials.

(b) HFIR (High Flux Isotope Reactor) [ORNL]

HFIR is a 100MW mixed spectrum research reactor, currently operated at 85MW, which is planned to remain in operation until 2035. The HFIR is a unique irradiation facility with the potential for high environmental and temperature control during irradiation, and very low to high flux irradiation. Some of the HFIR-irradiated specimens were shipped to Oarai Center of Tohoku University for Post Irradiation Examinations and others were tested at ORNL radiological facilities.

(c) MTOR (Magneto-Thermofluid Omnibus Research Facility) [UCLA]

The facility includes a magnet of homogeneous field to 2T in an area 15cm wide and 1m long, which can be used for testing the fluid dynamics of liquid metals and MHD flow for high Prandtl number stimulant fluids using electrolytes.

(d) PISCES (Plasma Interactive Surface Component Experimental Station) [UCSD]

A linear plasma simulator which can produce high-density plasmas with H, D, He and Be. Pulsed lasers are equipped for synergistic plasma exposure and pulse heating studies. Plasma diagnostics and surface characterization systems are furnished.

The TITAN project supports MFE/IFE common technology research including the performance of ablation plumes of the first wall by pulse heat loading.[4]

III. RESEARCH HIGHLIGHTS

III.A. Tritium and Mass Transfer in First Wall (TASK 1-1)

III.A.1. Background and objectives

It is of great importance to understand tritium behavior in fusion reactors in terms of safety and fuel economics. In fusion reactors, edge plasma environment is very complicated with several ion species such as fuel ions (D, T), fusion reactant (He), wall materials (Be, W, C for ITER 1st set of divertor), and edge cooling gas (Ne, N, Ar). In addition, neutron irradiation affects material degradation due to radiation damage and transmutation.

These effects, especially synergistic ones, are still under intensive investigation and are not known very well. In addition, first walls of blankets will be coated with armor materials (probably tungsten), but it is not well known how this armor layer affects tritium retention and diffusion. In Task 1-1 of TITAN, the purpose is to clarify tritium and mass transfer in the first walls of blankets by using tritium plasma, Be and He seeded plasma, and pulsed laser system.

III.A.2. Research plan

In Task 1-1, the following subjects are studied.
(1) To investigate tritium retention and diffusion behavior in tungsten and tungsten coated reduced activation materials (F82H, Vanadium alloy)
(2) To investigate production processes of material mixing layers and D retention by Be/He/D mixed plasma irradiation to tungsten under high temperature (>1100 K) and low temperature (~600 K) conditions.
(3) To investigate erosion of Be/W mixed materials under simultaneous irradiation of high density plasma and pulsed laser.

III.A.3. Experimental facilities

TPE is used to expose specimens with T/D mixed plasma (T/D = 0.1~0.5%). Then specimens are cut in a dry system and 2D tritium distribution is measured by Imaging Plate (IP). Using PISCES, Be/He/D mixed high density plasma is exposed to tungsten. Then D retention mesurement by thermal desorption spectroscopy (TDS) and observation of microstructure (He bubbles, etc.) by transmission electron microscope (TEM) are made. Ablation tests of Be/W mixed layer by pulsed laser irradiation were also carried out.

III.A.4. Main achievements

Results from TPE
(1) A new method to study tritium distribution in materials exposed to T plasmas with a dry cutting and IP technique. Tritium distribution in tungsten, F82H (Ferritic/martensitic low activation steel), and tungsten coated F82H (VPS coating) exposed to TPE T/D plasmas were measured by these methods. For tungsten samples, tritium distribution is well described by simple diffusion of T and D, while for F82H T distribution was not simply understood by a diffusion equation with a single diffusion coefficient and a plasma ion source term. For tungsten coated F82H, it was clearly shown that the W coating layer greatly reduced T diffusion into the F82H substrate material.[5]

Results from PISCES
(1) D/He (He:20%) mixed plasma exposure to pure tungsten was made at 573 K. By He addition, D retention was decreased by two orders of magnitude and blistering seen for pure D exposure was suppressed.[6] According to TEM observation, thickness of He nano bubble layers was 20-30 nm. This layer could have some functions to reduce D influx and increase D release, leading to significant reduction of D retention.
(2) Pure He plasma exposure to various W (SCR, stress relieved, W-Re alloy, fine grained W) at ~1100 K always showed nano-structure (often called "fuzz").[7]
(3) Simultaneous plasma and pulsed laser irradiations to solid Be and Be coated W were performed. Structure of ablation plume formed by Be atoms (clusters) and BeD molecules was observed.

III.A.5. Future directions

Task 1-1 activity came to halt after first half of 6 years period of TITAN because of the budget cut. But collaboration with UCSD and INL will continue based on TITAN achievements. In addition, Japanese domestic research activity will be activated more by the impact of TITAN collaboration.

III.B. Tritium Behavior in Blanket Systems (TASK 1-2)

III.B.1. Background and objectives

In a fusion energy system, tritium is generated by neutron capture in the blanket's breeder material, extracted and re-circulated as fuel in the plasma. When the breeder is designed to operate in the liquid phase, the interaction of tritium with the lithium bearing material is one of the most important physical processes in determining the feasibility and the attractiveness of the system because it is fundamentally linked with all aspects of plant operation, from fueling (tritium breeding ratio, tritium availability, etc) to power extraction (heat transfer capability, heat cycle efficiency, etc) to safety (tritium inventory, tritium release, etc). Since the blanket operates in steady-state conditions, the accurate determination of the tritium solubility, which links the concentration of dissolved tritium with its corresponding partial pressure at equilibrium over the liquid surface, is a fundamental design data need for all systems (magnetic or inertial confinement, hybrid systems, etc) based on liquid breeders. The other main transport properties, the diffusion constant and the mass transport coefficient at liquid and structures interfaces, play a major role in the selection and optimization of the tritium extraction and coolant purification system (in dual function blankets the two systems are acting on different fluids).

Another aspect related to tritium transport in blanket systems considered in this task is the evaluation of tritium permeation barriers, which may play a fundamental role in maintaining the tritium emission to the environment within the regulatory limits in a fusion energy system. Although the fabrication of such materials is not included in the Task scope, samples may become available from recent developments both in Japan and the US and could be tested with tritium within TITAN activities.

III.B.2. Research plan

1. Measure hydrogen isotopes solubility in lead-lithium eutectic at blanket conditions
2. Tritium permeation in structural materials and permeation barriers at very low concentrations
3. Investigate tritium extraction from lead-lithium eutectic (LLE) at blanket conditions
4. Modeling and system design for tritium extraction from fusion blankets

III.B.3. Main achievements

The initial focus of the experiments was the measurement of hydrogen solubility in LLE in a static configuration. The first results were obtained with hydrogen at input pressure above the liquid surface ranging from 10 Pa to 100 kPa and LLE temperature ranging between 300ºC and 650ºC. However, the initial technical approach was based on a set of implicit assumptions which have been undermined by the first 3 years experimental results[8], resulting in a more complex matrix of parametric evaluation of materials, test procedures and system configurations required to accurately define the solubility. As a result system optimization for tritium testing is ongoing.

Pre-conceptual design analyses of a forced convection experiment to investigate tritium extraction from lead-lithium eutectic based on the vacuum permeator concept have also been carried out.[9] They resulted in the definition of the main components based on a single leg loop, although further design development and construction is not currently foreseen.

III.B.4. Future directions

Tritium solubility tests will start in FY2010 in parallel with continued evaluation with hydrogen in a second bench-top experimental system that will later be modified to investigate other transport properties (diffusivity, mass transport coefficient). Test of tritium permeation barriers is foreseen in the last 2 years of the collaboration.

III.C. Flow Control and Thermofluid Modeling (TASK 1-3)

Liquid metal (LM) flow in a fusion blanket interacts with the plasma-confining magnetic field, and eventually the flow structure is strongly affected by a MHD (magneto-hydrodynamics) effect. Especially, LM flows under the strong magnetic field including the flow instabilities and the flow distribution in a coolant channel such as a manifold may determine the heat removal, the hot spots, the corrosion and the tritium transport. Therefore, the understanding of MHD themofluid behavior and the flow control are key issues for fusion blanket research and development. This task has been studying characteristics of the thermofluid under the strong magnetic field and modeling the MHD phenomena. This task conducts some MHD experiments using the MTOR, which is able to make a uniform magnetic field of maximum 2T in a rectangular gap of 15 cm in width and 1m in length.

III.C.1. LM flow distribution under strong magnetic field

Flow distribution of liquid metal in the coolant manifold under strong magnetic field was studied, which consists of three parallel channels with the electrically insulating walls. The working fluid was mercury, and electrical potential probes were used to measure the flow rates in each channel. As results, a uniform distribution was achieved in the case of the ratio of a magnetic force (Ha) and an inertia force (Re): $N = Ha^2/Re > 90$.[10]

III.C.2. Development of LLE flow diagnostic and interfacial effects

LLE is chosen as the working fluid in this task because it is one of the candidate LM coolants for fusion blanket designs. In order to grasp the LLE flow field, we have been developing a High-Temperature Ultrasonic Doppler Velocimetry (HT-UDV). To do this, LLE acoustic properties necessary for the HT-UDV[11] were obtained and a tracer particle for LLE flow was investigated. On the other hand, for the object to be measured, the interfacial phenomenon related to the LLE wettability may affect the MHD flows. This task has made the quantitative evaluations of the interfacial effects by measuring the contact angles of a LLE droplet on a silicon carbide (SiC) surface, and also measuring an electrical contact resistivity at a SiC-LLE interface.

III.C.3. MHD modeling and numerical simulation

In order to establish the MHD flow control behavior, some MHD models and numerical simulations have been performed and the MHD turbulent effects on the heat transfer have been studied. A mixed convection which consists of both a forced convection driven by pump and a natural convection due to the buoyancy under the strong magnetic field is one of the key issues for the blanket designs. This task has developed the turbulent model for a MHD quasi two-dimensional (MHD-Q2D) flow, and also numerically simulated the MHD-Q2D flows based on this model, so that the detailed understandings on the complex thermofluid where strong magnetic field and the buoyancy force interact have been obtained[12].

This task focuses on LLE flows under strong magnetic field in later half of TITAN project. At present a

LLE loop at UCLA is being constructed, and the non-MHD/MHD flow measurements with the HT-UDV system installed on the LLE loop will be conducted soon. Moreover in the latter half of TITAN project, this task plans to carry out an experimental study on the main techniques mitigating the MHD pressure drop, e.g. a flow channel insert (FCI) and an electrically insulation coating techniques.

The goal of this task is to establish MHD thermofluid models based on both experiments and numerical simulation and couple it with the tritium transfer model for the integrated designing of fusion LM blanket.

III.D. Irradiation-Tritium Synergism (TASK 2-1)

Understanding of tritium behavior (diffusion, trapping, desorption, etc.) in neutron-irradiated materials is indispensable for evaluation of tritium balance in fusion reactors. Tungsten (W) is currently recognized as a candidate plasma facing material because of its low tritium solubility and other favorable attributes such as low sputtering. Recent ion irradiation experiments, however, showed that the radiation defects in W such as vacancies and voids act as strong trapping sites for hydrogen isotopes.[13] Thus, the tritium retention in W may significantly increase with neutron irradiation. The objective of this work is to understand the effects of neutron irradiation on tritium behavior in W. Disk-type samples of pure W are irradiated in HFIR and retention and desorption of hydrogen isotopes including tritium are examined in the STAR facility. Nickel was also examined as reference material. Accomplishments in the first three years are described here.

III.D.1. Determination of Irradiation Matrix

From the viewpoint of neutron-induced activation, small samples are preferable. On the other hand, accuracy of retention and desorption measurements could be poor if samples are too small. In order to optimize the sample size, radioactivity after neutron irradiation was evaluated by FISPACT-2001 code in NIFS, and the accuracy of measurements was assessed with expected hydrogen isotope retention. Finally, the sample size was determined to be ϕ 6 mm x 0.2 mm. Small samples (ϕ 3 mm x 0.2 mm) were also prepared for microstructure examination.

Conditions of neutron irradiation were determined by taking account of expected wall temperature and neutron dose in fusion reactors, development of microstructure and project schedule. Irradiation temperature was decided to be 60ºC (quick start of irradiation due to simple capsule design), 300ºC (close to wall temperature of existing fusion machines and ITER), and 650ºC (close to expected wall temperature in DEMO).

Irradiation at 60 and 300ºC for about 1 day and 1 month has been completed. The extent of activation after about 1 day irradiation at 60ºC agreed well with the above-mentioned evaluation.

III.D.2. Measurement of Thermal Desorption Spectrum of Deuterium from Low-Dose Neutron-Irradiated W

The W sample irradiated at 60ºC for 0.025 dpa was shipped to INL and exposed to deuterium plasma in the linear plasma machine TPE.[14] The samples used in the present study are significantly small and thin in comparison with the standard size for TPE (1 inch diameter and about 1mm thickness). Hence, a new sample holder specially designed for the small samples with thermocouple mounted on precise linear motion feedthrough was fabricated. The surfaces of neutron-irradiated W samples were covered by black contaminant, although the samples were kept in sealed Mo envelopes during irradiation. One of W samples was exposed to deuterium plasma by TPE at ca. 200ºC and flux of 5.0×10^{21} D $m^{-2}s^{-1}$ for 2.0 h (fluence 3.6×10^{25} D m^{-2}). After exposure to deuterium plasma, the black contaminant completely disappeared from the exposed surface, indicating this black matter is a layer of W oxide. The cause of oxide formation was considered to be a leak in Mo envelopes resulting in reaction between samples and coolant water. Although the contaminant remained on side and backside, preliminary thermal desorption spectrum was measured.

Characteristic points of spectrum from neutron-irradiated samples are: (1) burst-like desorption observed in the low temperature region (<100ºC), and (2) continuation of desorption in the temperature range above 500ºC. The burst-like desorption was ascribed to deuterium release from oxide layers on the side and backside. The desorption in the high temperature region was attributed to the trapping effects by defects created by neutron irradiation. It should be emphasized that such clear effects of neutron irradiation was observed at irradiation dose as low as 0.025 dpa. Measurements of deuterium depth profiles with nuclear reaction analysis are currently under preparation. Behavior of hydrogen isotopes including tritium in high dose samples will also be examined in the near future.

III.E. Joining and Coating Integrity (TASK 2-2)

III.E.1. Background and objectives

There are many issues to be solved for practical construction of DEMO blanket, although the basic blanket fabrication technologies for ITER test blanket module made of reduced activation ferritic steels have been well progressing so far. Among the technologies, joining/coating technology is inevitable for assembling the blanket in which the weld line is expected to be 500 m per single body of blanket module. Therefore, it is considered that the lifetime of the blanket will be controlled by performance of the bonded joints.

The objective of this sub-task is to develop the joining and coating technology of advanced structural materials for DEMO blanket and to evaluate the performance of the joint and coating under irradiation.

III.E.2. Joining technology

The materials used were nano-sized oxide dispersion strengthened (ODS) steels, vanadium alloys and SiC composites. Since the ODS steels can be simultaneously used with reduced activation ferritic steels (RAFS) without any significant design modifications, ODSS/RAFS joint as well as ODSS/ODSS joint fabrication technology development were targeted. The maximum usage temperature of the ODSS is considered to be 150°C higher than RAFS, which can contribute to improving the thermal efficiency of the reactor. However, since melting during welding changes the morphology of the dispersion of nano-oxide particles and degrades material performance, a joining method other than welding is necessary for ODSS. In this sub-task, friction stir welding (FSW) and solid state diffusion bonding (SSDB) method was applied to the ODSS (16Cr-(0, 4)Al-2W-0.35Y_2O_3), and the joints were irradiated in HFIR to evaluate their performance under irradiation.

FSW was performed for an ODS steel with high Cr concentration at a rotating speed of 800 rpm with a line-scanning speed of 50 mm/min. The FSW treatment resulted in a growth of the grains, and consequently, a remarkable reduction of the strength at RT. However, the reduction of strength at elevated temperatures was so small that the FSW is adequate for the application of ODSS to practical blanket fabrication.[15]

SSDB was carried out at 1200°C at 25MPa for 1 h with and without insert material. Since the melting temperature of the insert material was lower than 1200°C, insert material is melted and the method is often called as liquid state diffusion bonding (LSDB). Tensile strength of both the SSDB and LSDB was not degraded by the bonding treatment. The elongation of LSDB was reduced to about half of the unbounded material. However, the elongation of SSDB was not reduced at all, indicating the joining method is very suitable to ODS steels.

Also being carried out in this task is the development of joints of dissimilar materials such as RAFS/316SS and V-alloy/316SS with electron beam welding.

III.E.3 W coating technology

W-coating technology is inevitable for fabrication of the diverter and first wall towards realization of fusion energy. The greatest challenge is due to brittleness of W that was not plastically deformed at RT. The plasma facing material, which suffers a cyclic high heat loading, has been required to be refractory material and resistant to high thermal fatigue. Furthermore, the sustaining structure of the diverter and first wall should contact to the coating tightly.

In this work, SSDB of W-ODSS was done at 1240°C with and without insert material (Fe-3B-5Si). The strength of joint between W/ODSS was evaluated by means of torsion tests, and a rather high value of 300 MPa was attained for the SSDB joint. A vacuum plasma spray (VPS) method was also attempted for W-coating on ODSS, RAFM and V-alloys.[16] It was confirmed that the fabricated W-coating on ODSS withstood a cyclic heat load at 4.8 MW/m^2 to 100 cycles.

Some of the joint and coated specimens fabricated in this program are being irradiated in HFIR. Some post irradiation examinations have been started at ORNL.

III.F. Dynamic Deformation (TASK 2-3)

III.F.1. Background and objectives

Irradiation creep is an important irradiation effect phenomenon for materials for radiation service, because it is a major contributor to potential dimensional instability of materials under irradiation at such temperatures that thermal creep is not strongly anticipated. For silicon carbide (SiC)-based nuclear components, the irradiation creep may be important, in particular when a significant temperature gradient exists and the secondary stresses developed by differential swelling can be severe. Flow channel insert in liquid metal blankets of fusion energy systems is an example of such an application.

Studies on neutron irradiation creep of SiC have so far been extremely limited and insufficient. The experimental method of estimating the creep parameters based on the stress relaxation in elastically bent strip samples was later adopted to examine the thermal creep behavior of refractory ceramic fibers and was named bend stress relaxation (BSR) method. In a recent work, Katoh et al. applied it for studying the irradiation creep of high purity, stoichiometric CVD SiC, demonstrating that the BSR technique is effective to determine the irradiation creep parameters.[17,18]

Based on the previous demonstration of the experimental technique and the recognized importance of the irradiation creep, a more detailed study on the BSR irradiation creep of SiC ceramics and composites was planned as the Phase-I program of Task 2-3 on dynamic deformation. The objective of the Phase-I study is to gain understanding of the transient creep and stress relaxation behavior of SiC ceramics, model composites, and composite constituents during neutron irradiation. More specifically, the study aims at determining the dose, temperature, and stress dependent BSR behavior of those materials and the correlation between the point defect swelling and the transient creep deformation

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III.F.2. Experimental

The standard "SiC Bend Bar" type capsule housing configuration was employed. Inside the standard sleeve, which holds two ceramic bend bar samples in the standard configuration, a rectangular casing is accommodated. Four BSR assembly units are accommodated in each coffin. Each BSR assembly is of size approximately 48 mm x 5.1 mm x 1 mm. The BSR assembly units are stacked together with CVD SiC liners/separator plates at the top, bottom, and between the units.

Materials to be studied and specimen loading to each rabbit are CVD processed SiC, polycrystalline (R& H and Coorstek: beta SiC) and single crystal (Cree : 6H SiC) and NITE processed SiC matrix. The dimensions of the BSR creep specimens are 40 mm (l) x 1 mm (w) x varied thickness. The thickness was varied to allow determination of the effect of stress magnitude. Typical initial flexural stresses for the specimen thicknesses 0.05, 0.1, and 0.15 mm are ~100, ~200, and ~300 MPa, respectively. The neutron irradiation experiments were performed using hydraulic and fixed rabbit facilities of the High Flux Isotope Reactor (ORNL). Target irradiation temperatures and fluence in the Phase-I were 300, 500, 800, and 1200ºC in the irradiation level from 0.01 to 1dpa.[17]

III.F.3. Main achievements

The relationship between the stress relaxation ratio and the initial flexural stress of R&H SiC after various irradiation conditions were examined. The result show that the BSR ratios are about the same at 300°C and 500°C in spite of swelling difference and it does not have initial stress dependence. Detail analysis of the irradiation behavior compared with unirradiated data[19] and micro structural development is ongoing. The phase-I result indicated that the BSR method in a conventional pre-straining configuration may be used to determine the steady state creep compliance.

Radiation creep test capsules with steady stress loading using bellows are under design for application to SiC/SiC and metallic materials

III.G. MFE/IFE System Integration Modeling (COMMON TASK)

III.G.1. Background and objectives

Aiming at the formation of the fusion engineering basis which is necessary for the development of the MFE and IFE reactors, the integrated engineering model of reactor system is constructed with executing the tritium and thermofuid system designs under the boundary conditions of burning core plasmas and out-vessel environments. Those activities cooperate with science and engineering experiment researches on estimating and controlling the time and spatial behaviors of tritium and energy flow formed in nuclear reactions. The uniqueness of this task is (1) to perform modeling of the integrated engineering system for design optimization (2) by cross connecting each task in this project (3) with improving consistency of elemental device researches and (4) making feed-back from each task.

*III.G.2. Task pla*n

The initial stage of this task consists of
(1) Identification of key issues and key parameters,
(2) Identification of baseline designs and operating conditions in such as FFHR, ARIES, KOYO,
(3) Execution of analysis of key issues on tritium and heat systems.

The later half consists of
(1) Integration of results from TITAN tasks into design process,
(2) Integration of blanket modeling,
(3) Providing researchers with information needs.

III.G.3. Main achievements

(1) Experimental models in each task were reviewed and listed 25 major parameters for model integration with selecting models and physics to be included.
(2) To identify modeling targets, a seminar series in Journal of Atomic Energy Society of Japan was arranged mainly by this task members, and published in 12 issues with the main title of "The Fusion Reactor Wall is Getting Hot ! -- A Challenge towards the Future for Numerical Modeling", covering MFE and IFE reactor designs, edge plasma, PWI, materials damage, neutronics, tritium breeding, heat removal.[20] This article included results of discussion in the past TITAN workshops.
(3) The first review was performed for the recent stellarator/heliotron designs of FFHR, HSR, ARIES-CS.[21] Design optimization and engineering feasibility are highlighted including maintainability and fabricability with prospect towards DEMO. Development of long-life blanket and unscheduled blanket replacement are common engineering key issues.
(4) Evaluation of tritium permeation loss rate was performed on the Flibe blanket and LiPb blanket systems, and it was found that rate limiting processes depending on tritium concentration profiles are key issues in modeling.[9]

III.G.4. Future directions

By combining model components, which are expected to be improved in each task on microscopic behaviors of tritium and heat transfer in blanket systems, development of an integrated system model and methodology are expected regarding macroscopic behaviors between each element in the recycling loop for blanket systems. At the same time, by making feedback of key issues to each task, it is expected to enhance task activities on developing tritium and thermofluid system with high consistency between each task.

The integrated system model with high consistency is used for optimization of mass and heat transfer system in MFE and IFE reactor designs. Furthermore, creation of breakthrough on reactor designs and proposal of new research issues are expected. As a result, the formation of the fusion engineering basis is expected to be pushed forward.

IV. EMPHASES IN THE SECOND HALF OF THE PROJECT

The following are planned as the emphases for the second half of the TITAN projects.

(1) Unique results are being obtained for irradiation-tritium synergism studies. These activities will be enhanced.
(2) The task on thermofluid of LLE in magnetic field will enhance the analysis of the impact on tritium transfer including modeling studies.
(3) Neutron irradiation and post-irradiation examination will be accelerated including timely shipping of the irradiated specimens to Japan.
(4) Interaction between each task and the Common Task will be reinforced for enhanced contribution to the integration modeling and the reactor design.

V. CONCLUSION

The Japan-USA Fusion Cooperation Program TITAN successfully completed its first half period with progress toward its objectives and with significant scientific accomplishments that address key issues for several attractive first wall and blanket systems.

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