§1. Overall Examination of Tritium Transfer and Thermofluid Control in Fusion System

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The TITAN collaboration started April 2007 to provide the scientific foundations for tritium and thermofluid control and materials performance in the first wall and blanket under conditions characteristic of fusion reactors, including interactive neutron irradiation effects. There is a mutually agreeable and beneficial balance between Japanese and U.S. interests in studying tritium behavior, irradiation effects and thermofluid for advanced first wall and blanket concepts of common interest, in resolving feasibility issues of fusion energy systems using fission reactors and other experimental facilities, and in developing computational models and predictive design and analysis tools.

For the subtask of Tritium and Mass Transfer in First-Wall, there were several technical accomplishments. The work involved the design and beginning installation of laser safety systems for the PISCES-B Be enclosure. A similar laser safety system was designed and installed on the PISCES-A device. Both lasers are currently being used to irradiate samples in PISCES-A during plasma exposures.

In the subtask of Tritium Behavior in Blanket Systems, a number of accomplishments were made. Measurements were made of hydrogen solubility in LLE at temperatures in the range 300-650°C and charging pressures ranging between 1.0 - 1.E5 Pa.

In the subtask of Flow Control and Thermofluid Modeling, experiments on flow distribution in the manifold test section were carried out. For fusion relevant magnetic interaction parameter, the flow distribution becomes nearly uniform, differing only by a few percent, between the 3 channels in the test. 3D MHD modeling gives a similar result. At lower interaction parameter the center channel takes a much larger portion of the flow and the gap

In the subtask of Irradiation-Tritium Synergism, neutron irradiation was carried out in HFIR (High Flux Isotope Reactor), Oak Ridge National Laboratory (ORNL). Irradiated specimens were shipped to STAR (Safety and Tritium Applied Research) facility in Idaho National Laboratory (INL) to examine tritium behavior with TPE (Tritium Plasma Experiment) or TRIIX (Tritium Ions Irradiation Experiment). Tungsten, which is a potential candidate of plasma-facing material, was selected as the main material to be examined because of lack of data on effects of neutron irradiation on tritium behaviors. Nickel is also examined as a fundamental material.

In the subtask of Joining and Coating Integrity, capsule design and analysis for the first set of the specimens were carried out. Finite element method temperature analysis of a 300 ºC capsule indicated that temperature variation will be reasonably small as 22, 14, 17, 22 ºC for tensile, TEM, small and large T-transport experiment discs, respectively. In the current irradiation schedule, capsules to 1.2 and 9.6 dpa irradiations at 300 ºC will start in cycle 421 (May 2009), and 500 ºC and 650 ºC capsules follow to start in cycle 422.

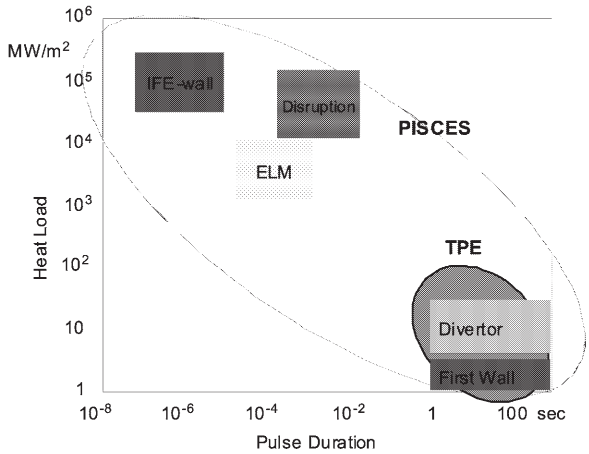

Figure 1 Pulse heat conditions for MFE and IFE.

In the subtask of Dynamic Deformation, the 18J capsule, completed irradiation of 6+ cycles, was transferred to the disassembly cell, and then was disassembled. Samples are being transferred from the hot cell facility to the LAMDA post-irradiation examination facility at ORNL. Post-irradiation examination including swelling, sonic modulus, four-point flexure, thermal diffusivity, and trans-thickness tensile tests has been initiated. A major part of the first order examination of the irradiated samples is anticipated to be tested before the end of JFY-2008.

In the Common Task: Integrated Systems Modeling, covering MFE and IFE fields, regarding to identification of modeling targets, a seminar series in Journal of Atomic Energy Society of Japan has been arranged mainly by this task members, and published for the total 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. This article included results of discussion in the past TITAN workshops.

Enhancement of common technology to MFE and IFE was discussed. Fig.1 is a pulse heat flux parameters covering MFE and IFE, which will be used as a guideline for designing the common technology research in the TITAN framework.

Based the present NIFS collaboration framework, extensive efforts were made to analyse the key feasibility issues throughout the first wall, the blanket and the recovery systems and decided the focal points of the collaboration activities in FY2008 and 2009. Also carried out was to review the domestic research activity relating with the subject of TITAN, and complementary roles of domestic and international collaboration researches were defined.