ITER RELEVANT NEUTRONICS EXPERIMENTS AT FNS

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Abstract

Under the ITER R&D Task framework, a series of experimental measurements and analyses have been conducted at Fusion Neutronics Source (FNS) Facility at JAERI, on various neutronics issues addressed from the critical nuclear design of ITER. The experiments comprised items of (1) bulk shielding of the ITER shield blanket configuration including the superconducting magnet layer, (2) streaming effects simulating a gap in two adjacent blanket modules, and (3) nuclear heating and induced radioactivity. Overall validity of design calculation, consequently, is assured in most cases by the C/E values. This paper deals with an overview of neutronics work at FNS/JAERI.

1. INTRODUCTION

As ITER is assumed to be a D-T burning plasma machine, neutronics issues associated with 14 MeV neutrons are of key importance. To insure the ITER nuclear design credibility, experimental validation of all nuclear relevant calculations along with nuclear data is strongly requested. In accordance with the Task Agreement under the ITER R&D Task framework, we have conducted a series of neutronics experimental measurements and analyses at Fusion Neutronics Source (FNS) [1] at Japan Atomic Energy Research Institute (JAERI). The experimental task comprised various experiments of (1) bulk shielding with SS-316, SS-316/Water, SS-316/Void and SS-316/superconducting magnet (SCM), (2) neutron streaming through a gap in two adjacent blanket modules, and (3) nuclear heating and induced radioactivity. The objectives of tasks are to execute experiments with close simulation of components of ITER structure, to perform experimental analyses with currently available nuclear data and code systems to be applied in ITER nuclear design, and to confirm the credibility in those ITER nuclear design calculation.

2. OVERVIEW OF NEUTRONICS EXPERIMENTS

2.1 NEUTRON SOURCE

The experiment was performed at FNS [1] of JAERI. D-T neutrons were produced by bombarding a water-cooled ^{3}T-Ti target with a 350 keV d^+ beam for bulk shielding experiments and a rotating ^{3}T-Ti target for streaming experiments. D-T source intensities were around 2 x 10^{11}/s and 3 x 10^{12}/s, respectively.

2.2 BULK SHIELDING EXPERIMENTS

As the ITER shield blanket and the vacuum vessel both consisted of SS-316 as the main structural materials, the experiments with the SS-316 assembly was identified as the primary base for a series of experiments. The base experimental assembly structure was made of SS-316. The dimension of the experimental assembly was 1.2 m in diameter and 1.22 m in length cylindrical shape as the test zone as shown in FIG. 1. A source reflector which had 200 mm thick hollow cylindrical shape was attached to the test zone and the D-T neutron source was positioned at a 300 mm distance from the front surface of the test zone and located in the source reflector as shown in FIG. 1. The structure and dimensions were the same for the other experimental assembly configurations with SS-316/Water, Void, and SCM. In the configuration of SS-316/Water, several layers of water were assembled with SS-316 layers. Spaces of cylinder or annulus shapes were placed at the central location in the SS-316 assembly for the Void experiment. For the SCM experiments, a region in which the materials simulating the SCM composite was attached after the 661 mm SS/Water shield zone. The detailed descriptions for those assemblies are given in Refs. [2, 3]. Various experimental data were measured in the simulated bulk shielding material regions. The measurement items and methods are as follows; i) neutron spectra by an NE-213 scintillator, a small proton recoil gas proportional counter and the slowing down time method, ii) dosimetry reaction rates, iii) reaction rate of $^{10}B(n,\alpha)$ with a BF_3 gas proportional counter, vi) fission rates of ^{235}U and ^{238}U with micro fission chambers, v) γ-ray spectra with a BC537 liquid organic scintillation counter and vii) γ-ray heating rates with TLD.

The analysis of these experiments was performed by MCNP-4A [4] with nuclear data libraries made of JENDL-3.2 [5], JENDL Fusion File [6] and FENDL/E-1.0 [7]. The DOT3.5 [8] calculations (P_5S_{16} approxima-

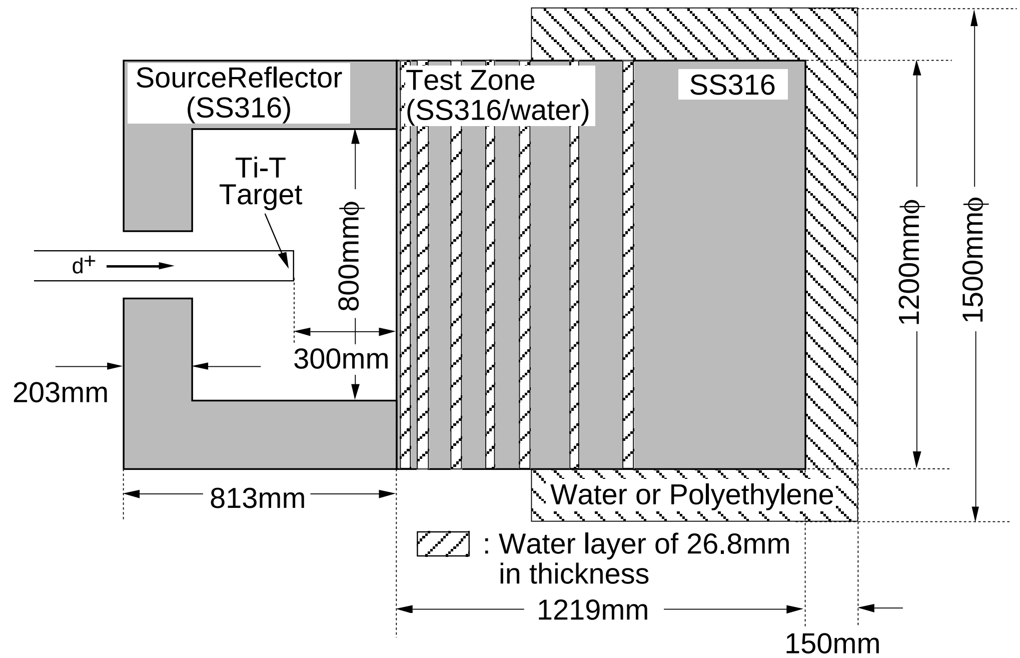

FIG. 1 Experimental assemblies for shield blanket simulation with SS-316 and SS-316/Water material configurations.

tion) with JENDL-3.2 base library were also executed to examine the adequacy of the discrete-ordinate code. In FIG. 2, C/E ranges are shown for the various neutron responses, as an example for the status of the calculation adequacy. As a result, the following conclusions were derived through the analyses.

(i) All the calculated results by MCNP, i.e., neutron fluxes from 14-MeV down to the thermal energy and γ-rays, agreed within 40 % with the measured data for nearly 1 m shield thickness with attenuation of 4 or 5 orders of magnitude.

(ii) Any types of voids give less attenuation of high energy neutron flux. For low energy neutron flux, however, there is no significant difference in the attenuation from the case of no void.

(iii) Agreements between calculations and experiments in the SCM region are not so good as those in the SS-316/water shield. The most possible reason is the uncertainty of the ^{10}B content in the insulator made of the epoxyglass.

(iv) Although contents of ^{181}Ta and ^{10}B are very small in SCM, it is found that these nuclei absorb more than a half of neutrons in the conductor and emit γ-rays which are sources of γ -ray heating in SCM.

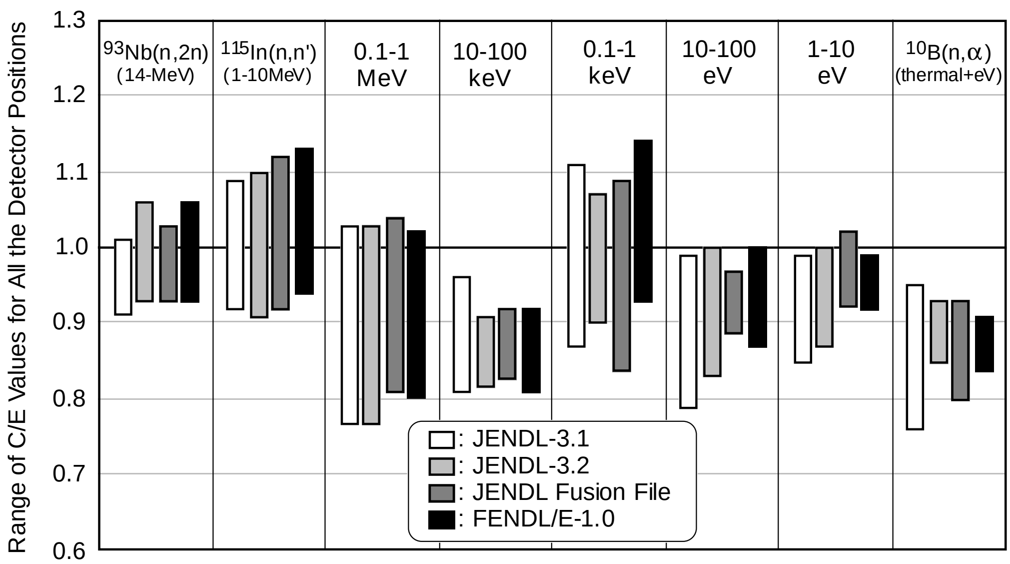

FIG. 2 Range of C/E values corresponding to various neutron responses for the bulk shielding experiments.

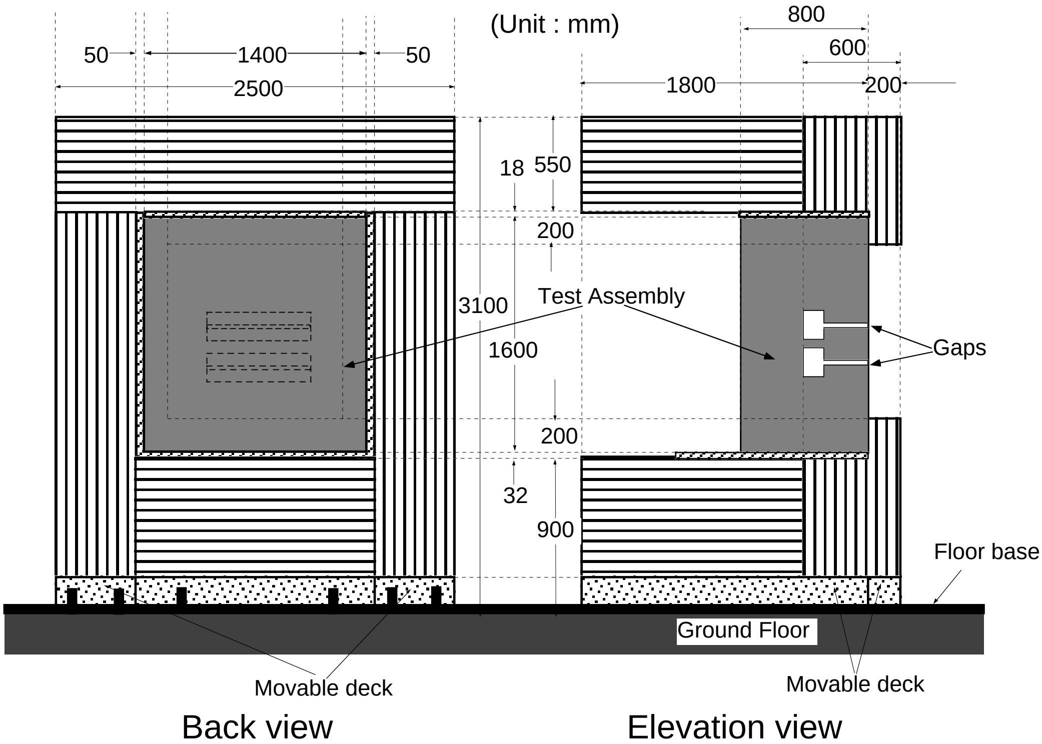

FIG. 3 Drawing of the experimental assembly for the streaming effect.

2.3 STREAMING SIMULATION EXPERIMENT

The experimental assembly made of iron as shown in FIG. 3, was installed in a large opening of an experimental room wall. The gap width was 22 mm and penetrated up to 300 mm depth in the experimental assembly. Cavities were placed behind the gaps. Firstly experiments of the source neutron and γ-ray field on the front surface of the experimental assembly was conducted. Then, neutron and γ-ray flux distributions were measured in the gaps and cavities, and on the rear surface of the experimental assembly at 800 mm. The measurement items were the same as those in the bulk shielding experiments. The same code and nuclear data used in the bulk shielding experimental analysis were used in the experimental analysis. The experimental conditions and geometrical configurations were modeled precisely.

The gap effect is defined as the ratio of a nuclear response with the gap to that without the gap. On the back plate surface, the gap effects of ^{235}U for low energy neutrons are at most 1.2, for both the direct and offset gap configurations, while those of ^{238}U for high energy neutrons are about 20 for the direct gap and ~ 5 for the offset gap configuration, respectively. The result indicates that the helium production rate, which is sensitive to high energy neutrons above a few MeV, increases significantly by high energy neutron contributions. Since the rewelding points on the connection legs are located apart from the gap center line, it is found that the helium production in SS-316 is reduced to be ~ 1/30 for the direct gap and ~1/3 for the offset gap configuration, respectively. Also, it is suggested that the large peaking factor is mainly due to the cavity.

As a whole, agreements between the experiment and the calculations are good. The all measured nuclear responses are predicted within 30 % by the calculations with the precise simulation.

2.4 NUCLEAR HEATING AND INDUCED RADIOACTIVITY EXPERIMENTS

2.4.1 Nuclear Heating

Nuclear heating rates of several structural materials in assemblies made of SS-316, copper and graphite which were irradiated with 14 MeV neutrons were measured with a calorimetric method. From C/E values for the total heating rates, following findings are identified: For carbon, chromium, copper, nickel, niobium and molybdenum, both JENDL and FENDL gave good agreements with measurements. Concerning iron, JENDL overestimates the experiment by 15 % and FENDL underestimated it by 8 %. Considering experimental error of 8 %, FENDL seems adequate. On the major structural material of SS-316, both calculations agree with experiment. However, JENDL is higher by 10 % than FENDL. It may be due to the difference in the KERMA data. The results are consistent with those in the copper assembly. For aluminum, JENDL seems adequate.

FENDL, however, overstimulates the experiment by 25 %. On vanadium, JENDL underestimates by 18 % and FENDL overestimates by 28 %. Both KERMA data should be checked carefully. Concerning W, both JENDL and FENDL give an overestimation by around 10 %. The overestimation is consistent with the previous results. Both JENDL and FENDL underestimates by 20 % for Zr. The KERMA should be checked. For Si it seems adequate though JENDL slightly underestimates by 8 %. While, FENDL largely overestimates by 50 %. The FENDL KERMA data should be revised after a careful investigation. JENDL underestimates Ti result by 20 %. FENDL agrees with the measurement.

2.4.2 Induced Radioactivity

Induced radioactivities in structural materials were investigated. The parameters were cooling time, neutron spectrum, cross section libraries, which should be factored into the consideration in the experimental analysis. The samples of Al, Mg, Ti, V, Mn, Fe, Ni, SS-316LN, Cu, Zn, Nb, Mo, Ag, In, Sn, Dy, Hf, Ta, W and Pb were irradiated with D-T neutrons at tow positions in the copper and graphite assemblies. After irradiation, induced radioactivity decay rates were measured by γ-ray spectroscopy with Ge detectors at cooling time from several minutes to about one year. Experimental data were derived as the radioactivity intensity per unit volume (Bq/cm^3). The ACT4 of the THIDA code system [9] was used as the radioactivity inventory codes with activation cross section libraries of JENDL-ACT96 [10], FENDL/A-1.0 [11] and FENDL/A-2.0 [12]. In general, FENDL/A-2.0 shows better agreement with the experiment than JENDL-ACT96 and FENDL/A-1.0. This fact is the direct demonstration of the quality of FENDL/A-2.0. In particular, for the major structural materials, the calculations agree within 20 % with the measurements.

3. CONCLUDING REMARKS

Summarizing the outcomes through those experimental analyses, we can conclude as follows:
(1) Nuclear responses in SCM region, e.g., nuclear heating, fast neutron flux, can be predicted within 30 % with the use of the state of art nuclear data, FENDL-1, JENDL-FF. (2) Capability of the calculation code and nuclear data for the local gap streaming between two shield blanket modules are assured within 30 %. (3) Nuclear heating relevant nuclear data for most of major structural materials are validated to be in an accuracy within 20 %. (4) Better accuracy of activation cross sections of FENDL/A-2.0 is proved to be within 30 % for almost all elements of major structural materials.

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