Neutronics Experiments for ITER at JAERI/FNS

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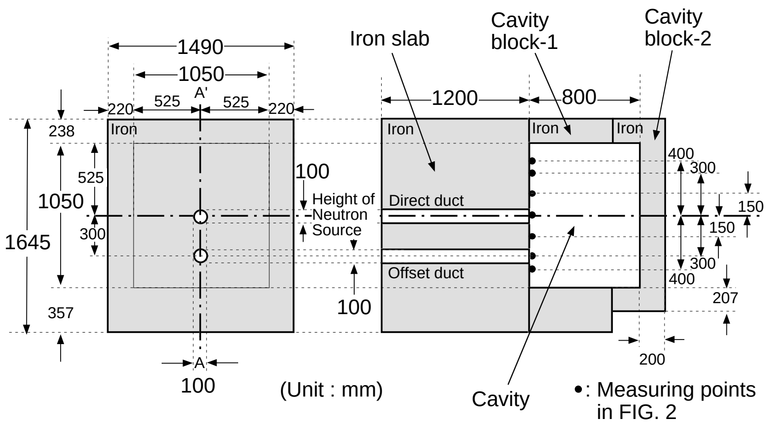
Abstract. A series of fusion neutronics experiments has been performed at the Fusion Neutronics Source (FNS) facility at JAERI as ITER/EDA R&D Tasks in order to deal with various nuclear problems originating from 14-MeV neutrons in ITER. Recently three experiments were carried out; 1) straight duct streaming experiments, 2) decay heat experiments and 3) development of a fusion power monitor utilizing activation of water. The straight duct streaming experiments suggest that the calculation accuracy for straight duct streaming analyses in ITER nuclear designs is ± 40 %. The decay heat experiments show that the accuracy of the decay heat calculation is within 10 % for copper and type 316 stainless steel, while it is ~ 30 % for tungsten. It is demonstrated that a fusion power monitor utilizing activation of water is applicable to ITER.

1. Introduction

A series of fusion neutronics experiments has been performed at the Fusion Neutronics Source (FNS) facility at JAERI as ITER/EDA R&D Tasks in order to deal with various nuclear problems originating from 14-MeV neutrons in ITER. Bulk shielding, gap streaming, nuclear heating and induced radioactivity experiments were presented at the previous IAEA Conference [1]. After these experiments, straight duct streaming experiments were carried out to demonstrate the design accuracy for more complicated shields in ITER. Also new decay heat experiments were started to study safety against loss of coolant accidents. Moreover a reliable fusion power monitor was developed using activation of water. This paper overviews these experimental activities newly conducted for ITER/EDA at JAERI/FNS.

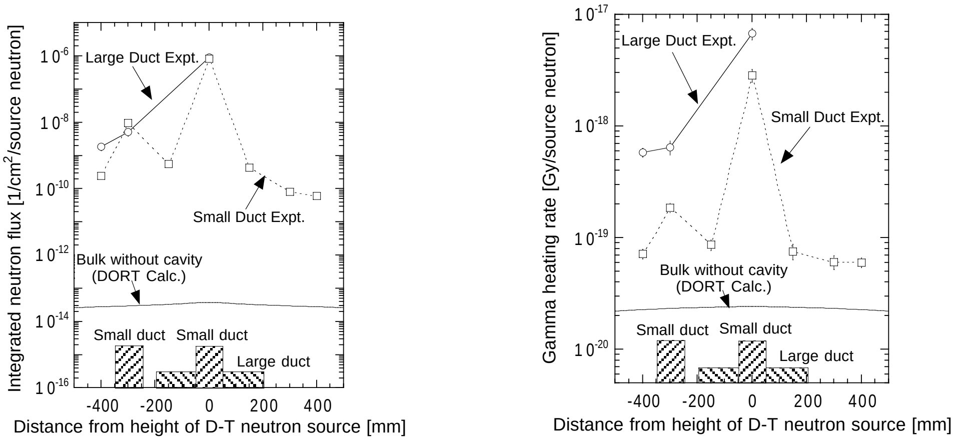
2. Straight Duct Streaming Experiments [2]

The aim of the experiments is at investigation of influences of radiation streaming through straight ducts, such as diagnostic ports and neutral beam injector (NBI) ports, and validation of the accuracy of design calculations. To simulate various duct dimensions, small duct and large duct streaming experiments which simulated the diagnostic ports and NBI ports, respectively, were



(a) Front view (b) Side view [Vertical cross section of A-A']

FIG. 1. The small straight duct streaming experimental assembly.



(a) neutron flux above 10 MeV *(b) gamma-ray heating rate of iron*

FIG. 2. Distributions of measured (a) neutron flux above 10 MeV and (b) gamma-ray heating rate of iron along the vertical line on the rear surface of the iron slab in the streaming experiments.

performed.

For the small duct streaming experiment, as shown in FIG. 1, two ducts of 100 mm in diameter were embedded in an iron slab of 1200 mm in thickness. One of the ducts (direct duct) viewed a point D-T neutron source located on the duct axis 1700 mm from the iron slab, and the other (offset duct) was located at a position shifted lower by 300 mm from the direct duct. One straight duct of 400 mm x 400 mm cross section was made at the same height as the D-T neutron source in the iron slab for the large duct streaming experiment. A cavity region was located behind the iron slab for simulation of a measurement or an ion source room at the end of a port.

Neutron spectra, various reaction-rates, gamma-ray spectra and gamma-ray heating rates of iron were measured along the duct and inside the cavity. Distributions of measured neutron flux above 10 MeV and gamma-ray heating rate of iron along the vertical line on the rear surface of the iron slab are presented in FIG. 2. Rough DORT calculations [4] with FENDL/MG-1.1 [5] for the bulk experimental assembly without ducts and cavity are also plotted for comparison in this figure. This figure shows that the ducts and cavity increase the neutron flux above 10 MeV by 10^7 and 10^6 times at the ends of the direct small and large ducts, and the offset small duct, respectively. It is noted that the neutron flux above 10 MeV increases by 10^3 times even at the shaded positions located 400 mm from the direct small duct. However, the increase of gamma-ray heating rate of iron due to the ducts and cavity is less than a few hundred times.

Experimental analyses were carried out by the Monte Carlo transport code MCNP4A with several transport cross section data; FENDL/E-1.1 [6], FENDL/E-2.0 [7] and JENDL Fusion File [8]. As a result, calculations predicted mostly within $\pm$ 40 % the complicated neutron and gamma-ray flux distributions in the ducts and cavity observed in the experiments for both small and large duct assemblies. Accordingly, it was demonstrated that the calculation accuracy for straight duct streaming analyses in ITER nuclear designs with MCNP4A and these data was $\pm$ 40 %, which is sufficiently accurate for radiation shielding calculations.

3. Decay Heat Experiments

Although accurate estimation of decay heat values is essential for safety analyses of fusion reactors against loss of coolant accidents, no experimental work has been devoted to validating the estimation. To ameliorate this situation, a decay heat measurement experiment was performed as a task of the ITER/EDA [9].

A new detector shown in FIG. 3, the Whole Energy Absorption Spectrometer (WEAS), was developed [10] for accurate and efficient measurements of decay heat. In the first phase experiments [11], decay heat produced in the thirty-two sample materials which were irradiated by 14-MeV neutrons was measured with WEAS for a long range of cooling time periods from 1 min to 400 days. Accordingly, the first experimental data for decay heat were obtained in the field of fusion neutronics. The validity of decay heat calculation codes such as ACT4 [12], activation cross section libraries of FENDL/A-2.0 [13] and JENDL Activation File [14], and decay data was investigated through analyses of the experiment. As a result, several points that should be modified were found in the codes and data. After solving the problems, it was demonstrated that decay heat values calculated for most of the samples were in good agreement, typically within 20 %, with the experimental data. For the two most important materials in ITER, copper and type 316 stainless steel (FIG. 4), it was found that the decay heat could be accurately predicted within 10 % with codes and nuclear data which were used in safety analyses. The data were further utilized for validating decay heat calculation codes and data used in other institutes through an international benchmark study [15].

The second phase experiments were devoted to decay heat produced in ITER baffle plates made of tungsten. In ITER-like neutron environments, the main decay heat source in the baffle plates is ^{187}W produced by the $^{186}W(n,\gamma)^{187}W$ reaction with low-energy neutrons. Hence, a decay heat experiment featuring low-energy neutron induced ^{187}W production [16] was performed. Tungsten foil samples were irradiated in an ITER-like neutron field, and the decay heat produced in the samples was measured with WEAS with experimental errors of ~ 5 %. The analyses of the experiments were performed with the MCNP transport calculation code. It was found that the use of an effective activation cross section calculated by MCNP was essential for properly treating the self-shielding effect of cross sections having sharp resonance peaks. When a result calculated with the FENDL cross section library was compared with the experimental data, the calculated decay heat was 21 % smaller than the experimental data. In conclusion, the safety margin

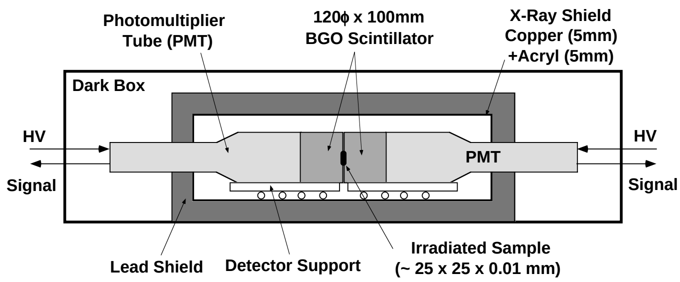


FIG. 3. Schematic diagram of the Whole Energy Absorption Spectrometer (WEAS).

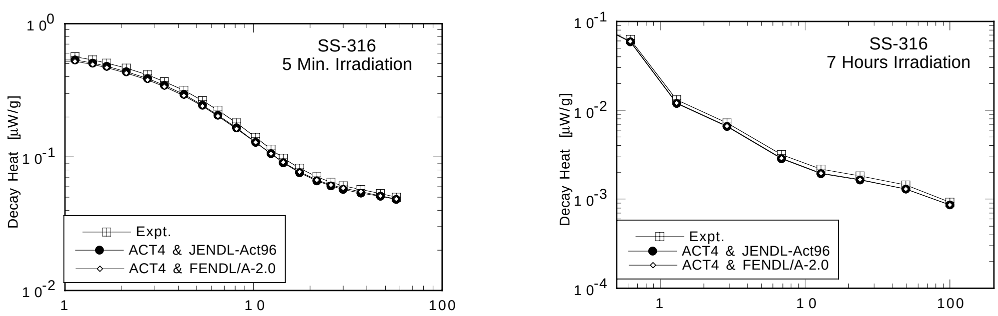


FIG. 4. Measured and calculated decay heat values for SS-316 samples irradiated for 5 minutes and 7 hours.

of 15 % used in the safety designs has to be increased to 25 % ~ 30 % for the ITER baffle plates.

4. Development of Fusion Power Monitor based on Activation of Water Flow

Neutron activation methods using solid metal samples are used for accurate measurements of the neutron yield in many fusion devices without temporal resolution. We have developed a neutron activation detector with water flow based on the $^{16}O(n,p)^{16}N$ reaction for accurate fusion power monitoring with reasonable time resolution for fusion experimental reactors such as ITER. When water is irradiated by 14 MeV neutrons, radioactive ^{16}N nuclei are produced by the $^{16}O(n,p)^{16}N$ reaction. A ^{16}N nucleus emits 6.13 (68.8%) or 7.12 (4.7%) MeV γ-rays accompanied by disintegration with half-life of 7.13 seconds. Since the $^{16}O(n, p)^{16}N$ reaction has a threshold energy at 10.4 MeV, ^{16}N nuclei are practically produced only by 14 MeV neutrons generated by D-T reactions in a fusion reactor.

The ITER requirements for this fusion power monitor are a temporal resolution of 100 ms and 10 % uncertainty. In the previous work [17], the temporal resolution was confirmed to be less than 10 s. We carried out experiments to investigate the temporal resolution more precisely at FNS. The experimental setup is shown in FIG.5. Part of a water pipe loop was placed in front of a neutron generation target with the center axis of the water pipe 1.5 cm from the target. The distance between the target and the γ-ray detector was chosen to be 10 m, which is the shortest practical distance from ITER-FEAT. As the γ-ray detector, a large BGO ($Bi_4Ge_3O_{12}$) scintillation detector was adopted, because BGO has very high effective atomic numbers suitable for counting high-energy γ-rays of 6.1 and 7.1 MeV from ^{16}N. The FNS accelerator was operated in a pulse mode for the time response measurement, with a pulse width of 10 ms. The flow velocity range was 2-11 m/s, which corresponds to a Reynold's number range of 2×10^4 to 1.2×10^5. We obtained a temporal resolution of 50 ms at a flow velocity of 10.7 m/s. It was found that the measured temporal resolution was described well by a turbulent dispersion model in this range of flow velocities [18].

In a real fusion experimental reactor, the irradiation end of the water loop will be installed in a blanket module. We investigated the effect of the surrounding material of the irradiation end to evaluate the accuracy of this fusion power monitor. A shielding blanket mock-up assembly consisting of multi-layers of type 316 stainless steel (SS316) and water with a source reflector (see FIG. 6) was utilized for simulating the neutron field at the shielding blanket region of ITER. The irradiation end was embedded in the first SS316 layer. The FNS accelerator was operated in a DC mode. The true neutron yield was monitored with an associated alpha detector mounted at the target assembly. Otherwise the setup was the same as that of the temporal resolution measurement. The neutron spectrum at the irradiation end and the detection efficiency of the γ-ray detector were calculated by the MCNP4B [19] code with the FENDL/E-2.0 library. The cross section data of the $^{16}O(n,p)^{16}N$ reaction was taken from the FENDL/A-2.0 file. The neutron yield from the ^{16}N activity was about 10% smaller than that from the associated alpha detector. The statistical error of the gamma-ray measurements was less than 1%. The uncertainty of the neutron yield from the associated alpha detector was 3%. The evaluated values of the $^{16}O(n,p)^{16}N$ cross-section have 5% difference among

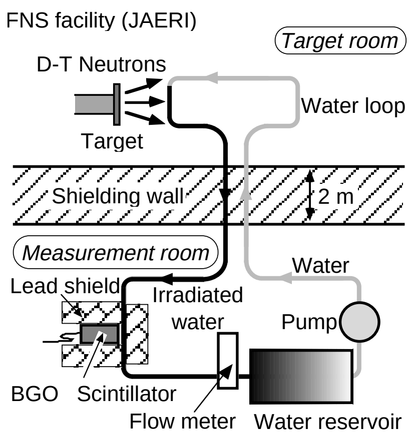


FIG. 5. Experimental set-up of the neutron monitor using activation of water flow for the temporal resolution.

nuclear data libraries. We need further investigations on the cross-section to improve the smaller neutron yield from the ^{16}N activity.

Finally, we can conclude that the fusion power monitor using water activation can measure the D-T neutron flux within 10% uncertainty with a temporal resolution smaller than 100 ms, and that the water-flow activation detector is applicable to ITER as a fusion power monitor.

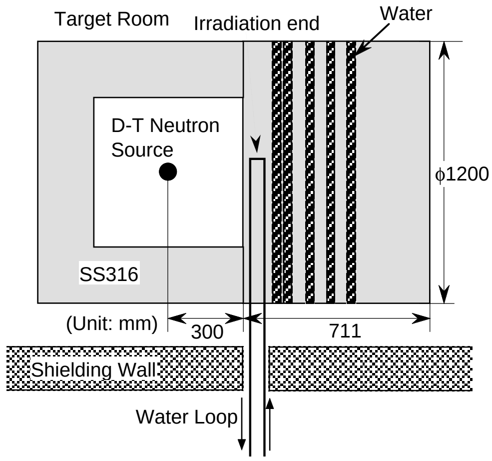


FIG. 6. Shielding blanket mock-up assembly to simulate the ITER neutron field.

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