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EXPERIMENT AND ANALYSIS OF NEUTRON SPECTRA IN A CONCRETE ASSEMBLY BOMBARDED BY 14MEV NEUTRONS

Koji Oishi and Kazuyuki Tomioka

Institute of Technology, Shimizu Corporation
4-17, 3-chome, Etchujima, Koto-ku, Tokyo, Japan 135

Yujiro Ikeda and Tomoo Nakamura

Japan Atomic Energy Reserch Institute
Tokai-mura, Ibaraki-ken, Japan 319-11

Abstract: Neutron spectrum in concrete bombarded by 14 MeV neutrons was measured using a miniature NE213 spectrometer and multi-foil activation method. A good agreement between those two experimental methods was obtained within experimental errors. The measured spectrum was compared with calculated ones using two-dimensional transport code DOT3.5 with 125 group structure cross section libraries based on ENDF/B-IV, JENDL-2, and JENDL-3T(the testing version of JENDL-3.) In the D-T neutron peak region, measured and calculated neutron spectra agreed well with each other for those libraries. However, disagreements of about -10% to +50% and -30% to +40% were obtained in the MeV region and still lower neutron energy range, respectively. As a result, it was concluded that those discrepancies were caused by the overestimation of secondary neutrons emitted by inelastic scattering from O, Si, and/or Ca which were the main components of concrete.

(D-T neutron, spectrum, concrete, ENDF/B-IV, JENDL-3PR1, JENDL-3T)

Introduction

Recently various kinds of shielding design for fusion reactors have been proposed and investigated[1-3]. To make fusion programs surely successful, it is unavoidable to strive for the reduction in the cost for the future design of proto-type fusion power reactors. Concrete is one of the most practical shielding materials that conform to the purpose.

In D-T fusion facilities, the high energy neutrons resulting from 14MeV source greatly increase the importance of reactions such as (n, n'), (n, p), (n, np), and (n, 2n) reactions having threshold in MeV range. The measurement of neutron spectra in concrete by integral experiment became an important issue; not only for the estimation of shielding ability and induced activities in itself, but for the verification of nuclear data libraries, because concrete formed a complex with several chemical elements.

The objectives of this study are to obtain proper experimental data of neutron flux distribution in the concrete assembly and to verify the available nuclear data libraries which will be used for the future design of fusion power facilities.

Experiment

Experiments have been performed at the Fusion Neutronics Source (FNS) Facility[4] of Japan Atomic Energy Reserch Institute (JAERI). Deuterium-Tritium neutrons were produced by bombarding the water-cooled 7 Ci Tritium-Titanium target with the 330 KeV deuteron beam. The neutron yield was monitored by the associated alpha particle counting method.[5]

The assembly was made of mortal which composition was equal to the ordinary concrete in order to prevent heterogeneous distribution of composition by the concrete aggregates. The composition of the assembly was obtained by a chemical analysis. The assembly was the cylindrical one, 600mm in diameter and 600mm thick, which was a proper geometry for the calculation by the two dimensional transport code DOT3.5. The assembly was set at 200mm from the target in the direction of the deuteron beam line. Experimental layout is shown in Fig. 1.

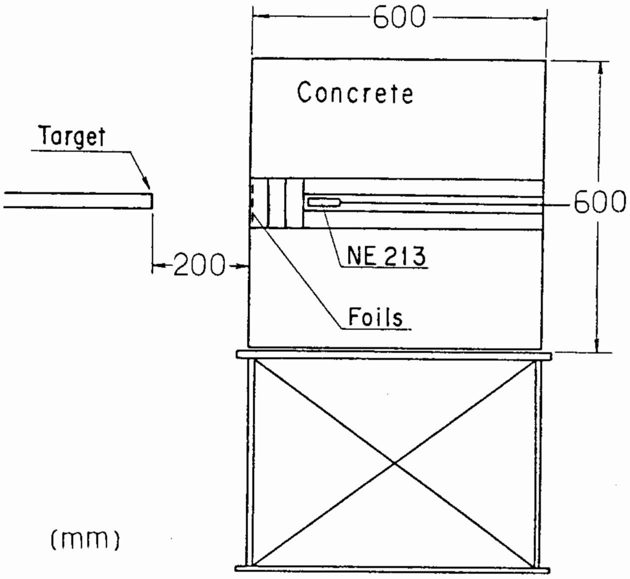

Fig. 1 Experimental layout

Table 1 Used reactions and their nuclear data

Reaction	Half life	Q-value(MeV)
^{56}Fe(n, p) ^{56}Mn	2.579h	− 2.9129
^{115}In(n, n')^{115m}In	4.3 h	− 0.3360
^{27}Al(n, α) ^{24}Na	15.02 h	− 3.1303
^{58}Ni(n, 2n) ^{57}Ni	1.50 d	− 12.1967
^{197}Au(n, γ)^{198m}Au	2.696d	6.5124
^{90}Zr(n, 2n) ^{89}Zr	3.27 d	− 11.9765
^{93}Nb(n, 2n) ^{92m}Nb	10.15 d	− 8.9674
^{58}Ni(n, p) ^{58}Co	70.8 d	0.4021

Neutron spectrum in concrete was obtained by two different methods; using a miniature NE213 spectrometer and multi-foil activation (MFA) method. The unfolding code was FORIST[6] and SANDII[7] for NE213 and MFA method, respectively. The miniature 14mm-diam spherical NE213 spectrometer[8] was adopted to measure neutron spectra in the assembly. The detector was placed at 16, 41, 66, 116, 216, 416, and 616mm from the surface of the assembly. For the MFA method, seven kinds of foils were selected according to the different sensitivity energy region of the reactions. Used reactions are listed in Table 1 with their nuclear data[9]. In the concrete assembly it was suggested that the neutron spectrum became softer at the deep detector position, ^{197}Au(n, γ) reaction, which had a sensitivity in the lower energy region, was also applied. The positions of the foils were 0, 25, 50, 100, 200, 400, and 550mm from the front surface. After irradiation for 10 hours, reaction rates were deduced from the gamma-ray counts using Germanium detectors with necessary corrections.

Using the results of reaction rate measurement, adjustment of neutron spectrum calculated with ENDF/B-IV was performed by the MFA method. The response for each reaction was based on ENDF/B-V and IRDF-82[10] dosimetry file. For some high threshold reactions, ^{58}Ni(n,2n), ^{93}Nb(n,2n), and ^{90}Zr(n,2n) reactions, newly measured and estimated cross sections at FNS[11] were applied for the data processing.

Comparison between the neutron spectra obtained by the MFA method and those obtained by means of NE213 spectrometer was performed. Since the positions of foil detectors were different from those of NE213 spectrometer, the obtained integral flux En≧11.5 MeV by MFA method were normalized to that obtained by NE213 spectrometer. To compare the shape of the spectrum obtained by those two experimental methods directly, the spectra of MFA method were smoothed with the window width of the unfolded spectrum by FORIST. The results of comparison between those two experimental methods at the detector position deeper than 116mm are shown in Fig.2. As a result, it is concluded that the consistency between those two experimental methods was confirmed.

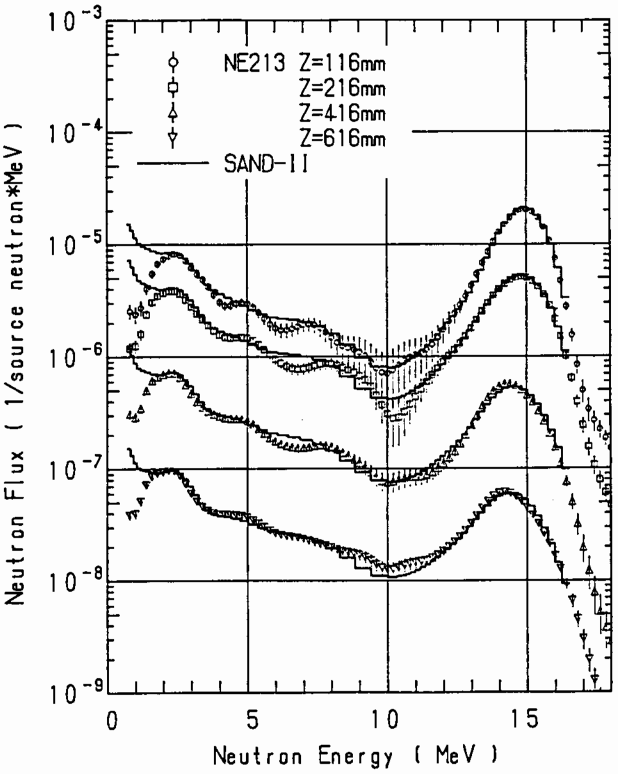

Fig.2 Neutron spectra obtained using NE213 and multi-foil activation method

Calculation

Calculation was performed using two dimentional transport code DOT3.5 with P5-S16 approximation. Nuclear data libraries used were independent cross section ones with 125 group neutron energies namely ENFKAS[12], JACKEX[12], and FSXJ3T[12] processed using ENDF/B-IV, JENDL-2, and JENDL-3T, respectively. The evaluated nuclear data files of each library are listed in Table 2 with every material. Some elements, which were not included in the libraries, were replaced by the main component of the assembly.

Table 2 Atomic density of compositions and evaluated nuclear data of each library

Component	Atomic density	FSXJ3T	JACKEX	ENFKAS
Si	1.1370+22 *	JENDL3T	JENDL2	ENDF/B-IV
Al	2.4650+21	JENDL3T	JENDL2	ENDF/B-IV
Fe	7.0330+20	JENDL3T	JENDL3PR1	ENDF/B-IV
Ca	5.2209+21	JENDL2	JENDL2	ENDF/B-IV
Mg	5.3073+20	ENDF/B-IV	ENDF/B-IV	ENDF/B-IV
S	1.1689+20	→ Si	→ Si	→ Si
Na	6.5574+20	JENDL3T	JENDL2	ENDF/B-IV
K	4.3714+20	JENDL3T	ENDF/B-IV	ENDF/B-IV
Ti	7.0278+19	→ Fe	→ Fe	→ Fe
P	2.4489+19	→ Si	→ Si	→ Si
Mn	3.0154+19	JENDL3T	JENDL2	ENDF/B-IV
Ba	1.0463+19	→ Si	→ Si	→ Si
V	5.249 +18	→ Fe	→ Fe	→ Fe
Co	2.27 +17	→ Fe	→ Fe	→ Fe
Zn	1.43 +18	→ Fe	→ Fe	→ Fe
Cu	6.52 +17	→ Fe	→ Fe	→ Fe
Ni	2.73 +17	→ Fe	→ Fe	→ Fe
C	2.1264+20	JENDL3T	JENDL3PR1	ENDF/B-IV
H	9.3507+21	JENDL3T	JENDL2	ENDF/B-IV
O	3.9484+22	JENDL3T	JENDL3PR1	ENDF/B-IV

* read as 1.1370 $\times 10^{22}$

It could be considered that there was no effect of that operation on the transport calculation, because the fraction of those rare elements was less than 0.3wt%. Source neutron spectrum at the target calculated by the Monte Carlo code simulating the structure of the target assembly was used as the D-T neutron source spectrum.

Comparison between Experiment and Calculations

Comparison between experimental spectra obtained using NE213 spectrometer and calculated ones at several positions were performed and the results were shown in Fig.3. The calculated spectra were smoothed using the window width applied in the unfolding process of NE213 spectrometer. Good agreement between experiment and calculations were obtained in the higher neutron energy range $En \geq 11.5$ MeV. Underestimation of JENDL-2 to the experiment was observed around the neutron energy En=10 MeV. In the lower energy region $1.5 \leq En < 8$ MeV, overestimation of all calculated results to experimental one in all detector position was obtained. It was also pointed out that there observed much difference between experimental and calculated spectrum by JENDL-2 in the neutron energy range $4 \leq En < 8$ MeV.

For the further comparison in detail, calculational to experimental (C/E) values of integrated neutron flux were examined. Experimental data and calculated ones were obtained using MFA method and ENDF/B-IV, respectively. C/E values for all detector positions were shown in Fig.4.

1) $En \geq 11.5$ MeV

This energy region corresponds to the D-T neutron peak. Good agreement between experiment and calculation was obtained within ±10%. The C/E value showed a tendency to increase due to the effect of accumulation cross section uncertainty by multiple scattering on calculated results.

2) $7.3 \leq En < 11.5$ MeV

In this energy region the ratio of the neutron population to that of other energy regions was very low, especially near the front surface. Good agreement between experiment and calculation was obtained except at the front surface.

3) $1.05 \leq En < 7.3$ MeV

The neutrons of this energy region was mainly produced by the inelastic scattering on concrete by the 14 MeV neutrons. Near the front surface, good agreement between experiment and calculation was obtained. However, the overestimation of calculated integral flux to experimental one was inclined to

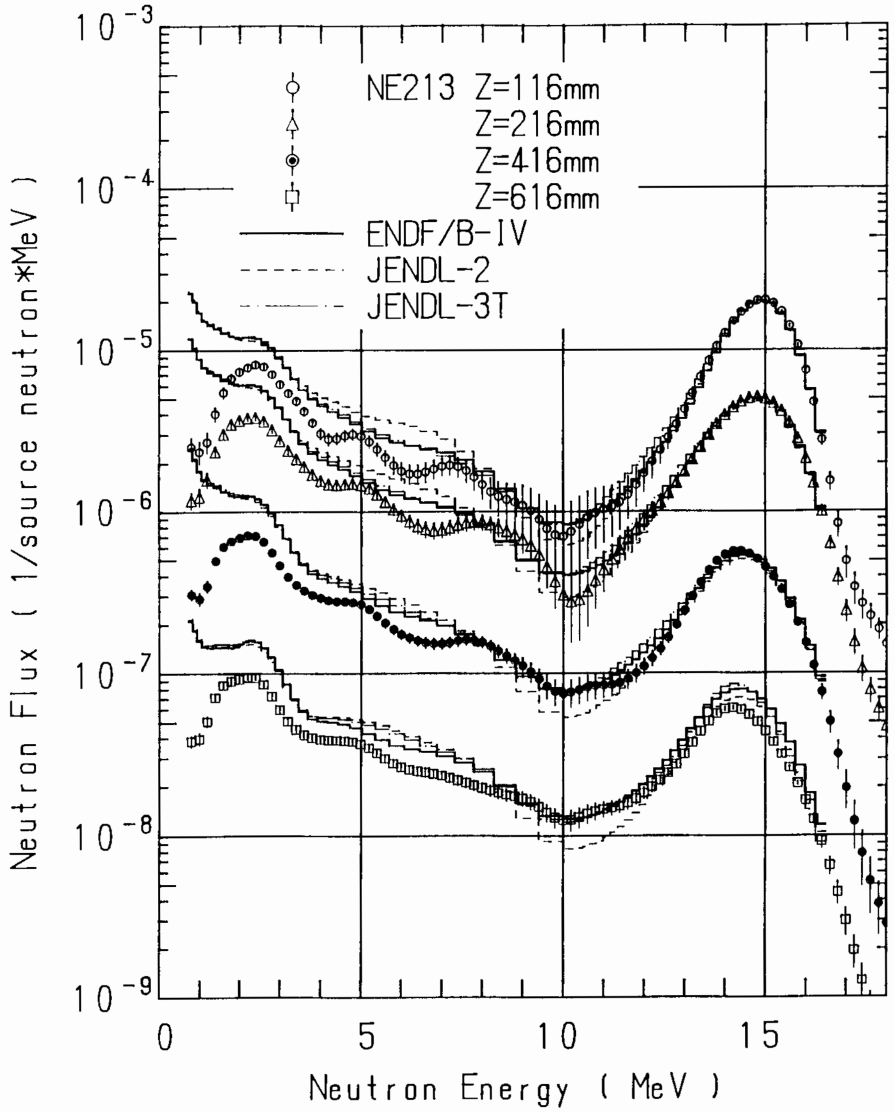

Fig. 3 Comparison between calculated neutron spectra and experimental one

increase as the depth of detector position increased. The C/E value increased up to 1.5 at the detector position Z=550mm. This inclination may be explained by the overestimation of secondary neutrons emitted from concrete materials by the 14 MeV incident neutrons.

4)En<1.05 MeV

Very large disagreement between experiment and calculation was observed, although good agreement of calculated integral flux among used libraries was obtained. The discrepancy at the front surface was explained by the lack of calculation on low energy reflected neutrons from the target room wall, because the wall was not included in the calculational code.

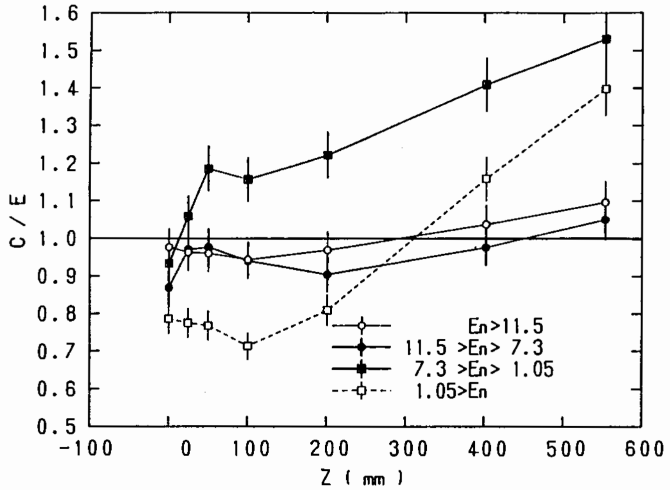

Fig. 4 The C/E values of integrated neutron flux for several neutron energy range

SUMMARY

The measurement of neutron spectra was performed using two experimental methods, and compared with calculations. Good agreement between those two experimental methods was obtained within experimental errors. The 125 group cross section libraries based on ENDF/B-IV, JENDL-2, and JENDL-3T were used for the analysis. From the comparison between experiment and calculations, it was concluded that further investigation on the secondary neutrons emitted from O, Si, and/or Ca by inelastic scattering for 14 MeV incident neutrons should be needed.

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