DESIGN AND TECHNIQUES FOR FUSION BLANKET NEUTRONICS EXPERIMENTS USING AN ACCELERATOR-BASED DEUTERIUM-TRITIUM NEUTRON SOURCE

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The experiments performed in the Japan Atomic Energy Research Institute/U.S. Department of Energy collaborative program on fusion blanket neutronics are designed with consideration of geometrical and material configurations. The general guide that is used to design the engineering-oriented neutronics experiment, which uses an accelerator-based 14-MeV neutron source, is discussed and compared with neutronics characteristics of the reactor models. Preparation of the experimental assembly, blanket materials, and the neutron source is described. A variety of techniques for measuring the nuclear parameters such as the tritium production rate are developed or introduced through the collaboration as a basis of the neutronics experiments. The features of these techniques are discussed with the experimental error and compared with each other.

INTRODUCTION

Many neutronics experiments have used deuterium-tritium (D-T) accelerator-based neutron sources for D-T-fueled fusion reactor research. Most of them tested nuclear data or neutron transport calculation code deficiency, so the experiments used simple geometry, i.e., sphere or slab. However, a realistic fusion reactor has a very complicated structure and geometry. For example, an experimental reactor design such as the International Thermonuclear Experimental Reactor (ITER) shows a module blanket, a cooling system, a tritium extraction loop, a shield, a removal structure, etc. The nuclear design calculations that use neutron transport codes, i.e., deterministic and Monte Carlo codes, usually approximate those complicated structures by homogenizing or simplifying. Also, the calculations are limited to a small part of the system because of the
insufficient resources of current computers. However, these modelings may introduce uncertainties different from those due to nuclear data and/or transport codes. Examination of reactor design-related problems requires an engineering benchmark (mock-up) experiment. An experimental system should be designed to resemble the reactor as closely as possible. Also, experimental examination requires a number of test blanket configurations contained within design variations to predict uncertainties in reactor nuclear designs. Because an experiment with a fusion device has not been possible so far, simulated and small-scale experiments must be considered. In the Japan Atomic Energy Research Institute (JAERI)/U.S. Department of Energy (U.S. DOE) collaborative program on fusion blanket neutronics, the experimental systems were selected by consideration of three stages of geometrical steps and a variety of material configurations.

In the experimental planning, specific material inventory, flexibility of assemblies, detector placement, material preparation, etc., need to be considered. On the other hand, general measurement techniques are essential to performing the experiments. Through the collaboration, many techniques have been developed or applied to measure nuclear parameters such as the tritium production rate (TPR), the neutron spectrum, and the reaction rate. Some of them are active (on-line) methods in which the data are taken at the same time as neutron irradiation, and the others are passive methods in which the data are taken after irradiation. The former requires a shorter measurement time with lower neutron intensity while the latter requires heavy (long and intense) neutron irradiation. The possibility of the desired experiments depends mostly on what kind of techniques can be applied.

In this paper, general consideration for the design of the experimental system is presented based on the simulation of neutronics characteristics discussed through the JAERI/U.S. DOE collaborative experiments. In the practical experimental setups, construction of the assemblies, material preparation, and the neutron source are described. As another aspect of the collaborative program, the measurement techniques developed and applied are outlined and discussed with their features. These techniques are one of the most valuable results of the collaborative programs and will be referenced by the accompanying papers.

**DESIGN OF EXPERIMENTAL SYSTEM**

Scaling and Simulation

In the experimental analysis of a nuclear reactor, a neutronics test is usually applied to a small-scale test assembly instead of to a mock-up system to reduce material inventory and cost. Neutronics behaviors in those systems are calculated by using a neutron transport code with neutron cross sections. The steady-state neutron transport equation is written as follows:

\[
\frac{\partial \phi}{\partial r} + \Sigma \cdot \phi = \int dE' d\Omega' \cdot \phi \cdot \Sigma(E', \Omega' \rightarrow E, \Omega) + S ,
\]

where

- \( \phi \) = neutron flux
- \( \Sigma \) = macroscopic cross section
- \( \Sigma_n \) = neutron energy
- \( \Omega \) = reaction angle
- \( r \) = space coordinates.

Here, when we replace \( r \) by \( r' = \xi r \) using a scaling factor \( \xi \) for a small system, and if we put \( \Sigma' = \xi \Sigma \) and \( S' = \xi S \), the equation is kept in the same representation. This means that when an experimental system is smaller by \( \xi \), if a material density and source intensity are increased by \( \xi \), the experiment is equivalent to the original system. However, because increasing material density is usually impossible, neutronics simulation of a real fusion reactor by a small system cannot be achieved this way. The characteristic size of the experimental system is measured by neutron mean free path in the material. If the density is not changed, the size of components such as a blanket should be kept the same as the design.

On the other hand, the distributions of nuclear parameters are changed not only by density but also by geometry. For example, the local tritium breeding ratio (TBR), defined as the line integral of TPR along the source to the blanket axis, is a measure of the geometrical effect. According to the calculation, the local TBR as a function of radius of a source cavity behaved as in Fig. 1 for two kinds of geometry. Figure 1 was obtained from calculation by the ANISN code with the
The local TBR of $^7$Li ($T_7$) showed $1/r$ and $1/r^2$ dependencies on the radius of the source cavity (even for $T_8$) for cylinder and sphere, respectively. These dependencies reflected directly from the source geometry. It is also suggested that the volume integrations over the whole blankets in both geometries were similar to each other.

As far as a total breeding ratio, which is one of the most important parameters for a D-T fusion reactor design, the volumetric integration over the whole blanket is of key interest. This type of integrated parameter can be characterized by the volume-integrated fluence as follows:

$$\text{total breeding ratio} = \int_{\text{whole volume}} N d r^2 \int \sigma(E_n) \cdot \phi_n(E_n, r) \, dE_n$$

$$= \int \frac{N \sigma(E_n) \, dE_n}{\text{whole volume}} \int \phi_n(E_n, r) \, dr^3$$

$$= \int \Sigma(E_n) \cdot \Phi_n(E_n) \, dE_n \quad \text{(2)}$$

where $\sigma(E_n)$ is the tritium production cross section and $\phi_n(E_n, r)$ is the neutron fluence at position $r$. Then, $\Phi_n(E_n)$ denotes the volume-integrated fluence for the whole system if one assumes that the material density is uniform. From Eq. (2), one can easily see that the system considered is characterized by the volume-integrated fluence (or the volume-averaged fluence by dividing by the whole volume). In this representation, the total breeding ratio was calculated by ANISN, and the result showed dependence on the radius of the source cavity, as predicted earlier. The dependence on the source cavity radius was almost the same in both cylinder and sphere geometries.

In the JAERI/U.S. DOE collaborative program, the experimental systems with a lithium oxide (Li$_2$O) breeder were selected for three source geometries, as shown in Fig. 2. The first was an open source geometry (Phase-I) (Ref. 2), the second was a closed geometry with source enclosure (Phase-II) (Ref. 4), and the third was an annular geometry (Phase-III) (Ref. 5). These experimental arrangements were designed to simulate a fusion reactor blanket, although a compromise was made with the material inventory and the facility restriction. Consequently, the material configurations and the geometry were simulated, although the dimensions were different from the reactor design.

To see how these experimental systems would simulate a reactor, we calculated the volume-integrated fluence for the test breeding zone as mentioned earlier. For the experimental systems, by assuming a sphere or a cylinder corresponding to a point source or a line source, respectively, we obtained the volume integration, as shown in Fig. 3. The integration along the measurement axis (axis from the source to the test blanket) in the Li$_2$O zone was performed by using the DOT3.5 two-dimensional code. The line integration is defined as

$$\Phi(E_n) = \begin{cases} \int \phi(r) 4\pi r^2 \, dr & \text{for a point source} \\ \int \phi(r) 2\pi r \, dr & \text{for a line source} \end{cases} \quad \text{(3)}$$

Those were compared with the volume-integrated flux in the model reactor, which was calculated for the breeding zone by ANISN with cylinder models, as shown in Fig. 4. The experimental configurations examined are summarized in Table I, and the reactor models with a solid breeder taken from Ref. 6 are shown in Table II. Figure 4 shows that the calculations for two models of mixed beryllium blankets showed softer spectra than
the best simulation was given by the annular blanket system. However, from a practical point of view for the implementation of experiments, the Phase-II system has many advantages for flexibility of configuration.

**Preparation of Experiments**

**Assembly of Test Blanket**

The most important requirements for an experimental assembly are flexibility to enable easy change of the configurations and effective use of material inventory. The block method for constructing the experimental system can meet such a requirement, although the system is restricted to a rectangular shape. The system can be made by directly stacking the blocks without additional structure. Because the breeder blanket generally consists of light materials, the structure with a large fraction of atomic density may disturb nuclear performance. On the other hand, an experimental assembly should have a geometrical shape that can be treated by a deterministic neutron transport code, i.e., at least a two-dimensional code. Thus, the assembly was designed as a pseudocylindrical slab or a rectangular prism.

The dimensions of the system also must be precisely known for the distance between the source and the detector and for the depth in the assembly. These dimensions were measured by scale after the construction of each assembly. An example of the system construction is shown in Fig. 5 for the Phase-I experiment where the blocks were stacked inside the penetration in the room wall and the experimental drawer could be operated from the back side of the assembly. For all the test blank, the experimental drawers were designed to accommodate detectors. The experimental drawer was made of 0.3-mm-thick stainless steel, and the sheet to insert the drawer was made of 0.2-mm-thick steel. Slightly smaller blocks were put in the drawer, and special blocks with 20-mm-square holes were used to pass the signal cables of the detector.

**TABLE I**

<table>
<thead>
<tr>
<th>Source</th>
<th>Phase-I REF</th>
<th>Phase-II REF</th>
<th>Phase-II REF</th>
<th>Phase-IIIA</th>
<th>Phase-IIIA BEF*</th>
</tr>
</thead>
<tbody>
<tr>
<td>Assembly</td>
<td>Point</td>
<td>Point</td>
<td>Point</td>
<td>Line</td>
<td>Line</td>
</tr>
<tr>
<td>Open</td>
<td>Cylinder</td>
<td>Rectangular</td>
<td>Rectangular</td>
<td>Annular</td>
<td>Annular</td>
</tr>
<tr>
<td>Distance from source</td>
<td>250 cm</td>
<td>78 cm</td>
<td>78 cm</td>
<td>Finite cylinder</td>
<td>Finite cylinder</td>
</tr>
<tr>
<td>Multiplier</td>
<td>None</td>
<td>None</td>
<td>21.3 cm</td>
<td>None</td>
<td>20 cm</td>
</tr>
<tr>
<td>Blanket</td>
<td>600 mm</td>
<td>600 mm</td>
<td>50-mm-thick beryllium</td>
<td>400 mm</td>
<td>50-mm-thick beryllium</td>
</tr>
<tr>
<td></td>
<td>Li$_2$O</td>
<td>Li$_2$O</td>
<td>600 mm</td>
<td>Li$_2$O</td>
<td>Li$_2$O</td>
</tr>
</tbody>
</table>

*This experiment, where beryllium is placed on the blanket surface, was not performed; only the calculation was done.
### Table II

Reactor Models Taken for Comparison with the Experimental Systems

<table>
<thead>
<tr>
<th></th>
<th>PCA/Beryllium/Li$_2$O/H$_2$O</th>
<th>Molybdenum Alloy/Beryllium/Li$_2$O/H$_2$O</th>
<th>PCA/Beryllium-Li$_2$O/H$_2$O</th>
<th>Molybdenum Alloy/Beryllium-Li$_2$O/H$_2$O</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>First wall</strong></td>
<td>200-cm radius</td>
<td>200-cm radius</td>
<td>200-cm radius</td>
<td>200-cm radius</td>
</tr>
<tr>
<td></td>
<td>16.5 mm thick</td>
<td>16.5 mm thick</td>
<td>16.5 mm thick</td>
<td>16.5 mm thick</td>
</tr>
<tr>
<td></td>
<td>67% PCA and 37% H$_2$O</td>
<td>80% molybdenum and 20% H$_2$</td>
<td>67% PCA and 33% H$_2$O</td>
<td>80% molybdenum and 20% H$_2$</td>
</tr>
<tr>
<td><strong>Multiplier</strong></td>
<td>50-mm-thick beryllium</td>
<td>50-mm-thick beryllium</td>
<td>Mixed</td>
<td>Mixed</td>
</tr>
<tr>
<td><strong>Blanket</strong></td>
<td>450 mm thick</td>
<td>450 mm thick</td>
<td>500 mm thick</td>
<td>500 mm thick</td>
</tr>
<tr>
<td><strong>Breeder</strong></td>
<td>89% Li$_2$O (30% $^6$Li)</td>
<td>82% Li$_2$O (30% $^6$Li)</td>
<td>22% Li$_2$O (natural lithium)</td>
<td>20% Li$_2$O (natural lithium)</td>
</tr>
<tr>
<td></td>
<td>7% H$_2$O and 4% PCA</td>
<td>13% H$_2$ and 5% molybdenum</td>
<td>67% beryllium, 4% PCA</td>
<td>62% beryllium, 13% H$_2$ and 5% molybdenum</td>
</tr>
</tbody>
</table>

*Primary candidate alloy is designated by PCA, and 0.25% titanium is added to almost the same composition as Type 316 stainless steel.

**Materials**

Lithium oxide, lithium carbonate (Li$_2$CO$_3$), and beryllium were prepared for the experiments. Lithium oxide is a main breeding material adopted in the reference design of ITER. Because the inventory was not enough because of the cost, Li$_2$CO$_3$ instead of Li$_2$O was chosen as the surrounding material of the container in the Phase-II experiment. Beryllium is a neutron multiplier.

![Diagram of experimental assembly](image)

*Fig. 5. The material block structure of Phase-I experimental assembly.*
Three types of Li$_2$O were made to construct a test zone. The outside dimensions of these blocks had an area of 50.6 × 50.6 mm and lengths of 50.6, 101.2, and 202.4 mm. The bricks, almost cubic, were made from Li$_2$O powder by cold pressing. The impurity of the metal elements in the powder was <0.2%. The density of Li$_2$O bricks was 75.5% of theoretical density. One, two, and four Li$_2$O bricks were encapsulated, respectively, in a 0.2-mm-thick Type 304 stainless steel box, edge welded by the Tig method with a 2.5-mm-deep rim. All the procedures were performed in a glove box filled with argon gas and a little helium, which was used for leak detection. By this special procedure, the impurities of H$_2$O and CO$_2$ in the bricks were kept at <0.1%, respectively.

Lithium carbonate blocks were individually machined to final dimensions (51 × 101.5 × 203 mm). An epoxy surface sealer was applied principally as a tritium barrier but also to prevent moisture uptake. On the basis of an estimated 50 μm of triton recoil range, the epoxy was applied to an ~0.1-mm thickness. Major steps in the fabrication of the Li$_2$CO$_3$ blocks were (a) mixing and drying the constituent powders, (b) cold pressing into block shapes, (c) sintering, (d) machining to final dimensions, and (e) coating the blocks to provide a tritium-impermeable surface. The raw materials included Li$_2$CO$_3$, Na$_2$CO$_3$, K$_2$CO$_3$, and an organic binder.

Small amounts of potassium and sodium carbonate were added to lower the melting point as a sintering aid; the organic binder was used to enhance binding during cold pressing. These raw materials were mixed with water in a weight ratio of 400:15:15:17 (Li$_2$CO$_3$:K$_2$CO$_3$:Na$_2$CO$_3$; Avicel) to form a slurry containing ~50% solids.

Beryllium blocks were also machined to cubes of 50.8 × 50.8 × 50.8 mm$^3$ with some of them machined to half thickness. The density was 1.837 g/cm$^3$, and the weight density of beryllium was 98.92%. The impurities were 1.13% for BeO and a small amount of iron, carbon, and aluminum for the rest.

**Neutron Source**

We based the neutron source on a $^3$T(d, n)$^4$He reaction using a deuterium beam and tritium metal target system. For comparison of calculations with the experiment, it was important to characterize well the energy spectrum and the angular distribution of the emitted neutrons. First, the neutron source yield was monitored absolutely by detecting alpha particles associated with the D-T reaction. The accuracy of this monitor was estimated to be 2%. Second, the energy spectrum and the angular distribution of the neutrons were specified based on detailed Monte Carlo calculations of the target assembly. The calculations were confirmed by experimental information such as the high-energy part of the neutron spectrum and the reaction rate distribution around the target. For the computational analysis of these experiments, this calculated source was used because there was no way to determine experimentally the source condition for the test assembly. This calculated source agreed with the experiment for fast neutrons within an experimental error of 4 to 5%.

**MEASUREMENT TECHNIQUES OF NUCLEAR PARAMETERS**

**TPR**

The measurement of tritium production is one of the most important items in the fusion blanket neutronics experiments. Several kinds of techniques were proposed and applied in the experiments.

**LSC Method with Li-Metal Foil**

Tritium assay with metal foils was originally developed by Bretscher and Redman for a purpose other than fusion blanket evaluation and adapted to the special needs of Li$_2$O breeder blankets. The scheme in metal foils is a passive method that counts tritium beta decay after irradiation of the sample. The Li-metal sample size is 18 mm in diameter and 0.5 mm thick. The sample is capped by aluminum, and the outer diameter of the aluminum capsule is 23.8 mm. Tritium is extracted from the samples by melting the aluminum capsule in a carrier hydrogen atmosphere and converted to water containing HTO on a hot copper oxide surface. This gives the Bretscher method two advantages: First, no tritium escapes during and just after irradiation by capsule; second, all the extracted tritiums are changed only to water form. The foils in the experimental drawer are typically irradiated for >10$^{15}$ neutrons emitted from the source.

A liquid scintillation counting (LSC) system (PACKARD 2000) is equipped with a low-background shield and a software package that retains complete spectrum information. Using this scintillator cocktail in 19-ml samples containing 1.6 ml of water containing HTO, one can regularly obtain a tritium detection efficiency close to 46%, together with a typical background rate of 4 cpm. The detecting efficiency of tritium is calibrated with a standard tritium sample common in LSC techniques. As for vials, thin-walled "low-K" glass is used.

Error contributions due to statistics of counting can usually be reduced by adequate counting time, except for the inherent uncertainty of background averaging. Mass spectrometric composition error of sample, uncertainty of extraction yield, weighing error, pressure measurement, and variability of LSC efficiency with the aqueous fraction (which differs slightly from one sample to the next) each contributes <1%. The overall error comes to ~5% for the TPR measurement, excluding self-shielding effects for $^6$Li.

**LSC Method with Li$_2$O Pellet**

This is a method similar to the Li-metal foil method except for the chemical process, but the advantage is...
to use the same Li$_2$O material as the material of the test blanket; no further consideration is required for the self-shielding effect within the pellet. The Li$_2$O pellet was a 20-mm-diam and 3-mm-thick disk. To expel 100% of the tritium from the irradiated sintered Li$_2$O pellets (with a weight of $\sim$0.38 g and a density of $\sim$83% of theoretical density), we dissolved the pellets in water in a two-forked test tube of the apparatus. This way, without acid, avoids chemical luminescence. The HT gas component, expected to be partially formed in the Li$_2$O, was oxidized in a CuO furnace and condensed in a liquid hydrogen cold trap together with H$_2$ purge gas. The HTO liquid component, which remained in the solution, was pipetted into a reduced pressure distillation apparatus and purified. The amount of tritium that originated from the HT gas component and that of the HTO component were measured separately by the liquid scintillation method to estimate that ratio. This separation procedure was made for some of the pellets to check the ratio. As for the remaining pellets, the HTO components alone were measured, and their total amounts of produced tritium were corrected with the average ratio of (HTO + HT)/HTO obtained from the measurement. The correction factor was estimated to be 1.025 ± 0.010.

To clarify the ambiguity in the tritium escape in the form of recoiled tritons, we irradiated pellets sandwiched between aluminum foils near the neutron source. The tritons implanted in the aluminum foils were expelled by heating at $\sim$800°C and were converted to HTO with the CuO furnace. The liquid scintillation method was also used to measure the tritium amount. The fraction of tritium recoiled into the aluminum foils to that which remained in the pellets was $1.1 \pm 1.5\%$. The estimated errors were due to the atomic number of lithium, weighing, and scintillation counting efficiency, and the total became 4 to 5% after correction, including counting statistical error.

**LSC Method with Li$_2$O Block (Zonal Method)**

The method that utilizes the breeding materials piled as a mass to act as the detectors themselves is named the zonal method. The measuring technique of the zonal TPR was developed for application to the engineering-oriented fusion blanket benchmark experiments. General conventional methods use small pellets as probes placed between stacked Li$_2$O blocks. The zonal method is expected to have the following advantages, especially for heterogeneous configurations:

1. High accuracy of the data of the region-integrated TPR can be obtained around the boundary of different materials, e.g., between beryllium and Li$_2$O where the gradient of TPR distribution is very steep because the data are free from positioning error.

2. A good signal-to-background ratio can be obtained even in deep positions inside the test blanket because of high sensitivity.

Moreover, the first feature gives benchmark data suitable for Monte Carlo calculation because zonal experimental data can be directly compared with the result of estimators of large volume such as track length estimators.

We used a few different sizes of cold-pressed bricks ranging from 12 × 48 × 48 mm to 48 × 48 × 48 mm employing a granule type of Li$_2$O powder. In our earlier development work, we tried to fabricate sintered Li$_2$O plates from a fine powder of Li$_2$O, but we found that samples with thicknesses >15 mm were hard to make because serious capping occurred in cold pressing. When heating under a vacuum is applied instead of a dissolution method for tritium extraction from irradiated bricks, the detector can be repeatedly used. The irradiated samples were heated to a temperature of 650°C for 2 h, as shown in Fig. 6. Tritium assay in the extracted gas was done for each component of HTO.

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**Fig. 6. Tritium extraction apparatus for zonal samples.**
and HT. The residual HTO component in the cold-pressed sample was measured, but tritium in the solution was not detected. The ratio of HT to HTO varied from 0.002 to 0.04 with the cold-pressed samples and from 0.002 to 0.004 in the sintered case. We concluded that the amount of tritium in the HT component is not negligible and varies depending on the fabrication of sample. Therefore, both HT and HTO were measured for each sample. Extracted tritiated water of 6 cm$^3$ was mixed with 14 cm$^3$ of scintillation liquid (Aquadex-2); i.e., a total of 20 cm$^3$ of liquid sample was prepared.

The experimental errors of this technique are mainly divided into the following components: (a) extraction efficiency and (b) calibration of the LSC system. The first component was <0.5% from the procedure mentioned earlier. The second component was in common with the other scintillation counting technique. It could be <2% for fitting of the efficiency curve and 0.86% for the standard of the National Bureau of Standards. Finally, the overall error was expected to be <3% by taking into account counting statistical error.

Self-Irradiation Method with LiF TLDs

Tritium production rates of $^6$Li, $^7$Li, and $^8$Li were also measured by a self-irradiation method with LiF thermoluminescence dosimeters (TLDs). The LiF powder of TLD-600 ($^6$Li: 95.62 at. %), TLD-700 ($^7$Li: 99.993 at. %), and TLD-100 (natural abundance) from Harshaw Chemical Company was sealed in 2-mm-diam × 12-mm Pyrex glass ampoules. The total neutron yield during the irradiation was typically $3.0 \times 10^{16}$ at the target. After waiting a period of 14 days to allow short-lived isotopes in the ampoule to decay, we annealed the TLDs at 400°C for 2 h. After this process, they were stored in a low-background shielded box for 148 days. The thermoluminescence caused by the self-irradiation of the beta-rays from the tritium during this period were measured by a TLD reader.

The TPR data of $^6$Li ($T_\alpha$) were corrected for the atomic ratios of $^6$Li and $^7$Li in the TLD-600. The TPR distributions of TLD are shown in Fig. 7 with those of the Li-glass and the Li$_2$O zonal method. As the TPR distributions of TLD were not measured absolutely, the data of $T_\alpha$ at 48.8 mm shown in Fig. 7 were normalized to $6.0 \times 10^{-29}$ (arbitrary value), and the other TLD data were shifted relatively.

Indirect Method Using NE-213 Scintillation Spectrometer

An active (on-line) method is very useful when quick measurement is required for many measuring points and for comparative study among experimental systems. For this purpose, an NE-213 scintillation spectrometer was applied to obtain TPR. The threshold energy of the $^7$Li($n$, $n'\alpha$) reaction was $\sim 3$ MeV while the measuring energy range of the NE-213 was above 1 MeV. The TPR from $^7$Li was estimated from the neutron spectrum measured by the NE-213 spectrometer and the data of the $^7$Li($n$, $n'\alpha$) reaction cross section. Because the cross section of $^7$Li used in the data reduction was retrieved from the JENDL-3PR2 evaluated at JAERI (Ref. 14), one should note that the obtained TPR distributions were biased by the data of the cross section. Thus, the obtained TPR should include the errors due to the measured neutron flux spectrum and the cross-section data. The overall error possibly becomes >5% when the cross section error is included.

A Pair of Thin Li-Glass Scintillators Method

The TPR of $^6$Li was measured by using the difference between the response of the $^6$Li-glass scintillator and that of the $^7$Li-glass in a mixed neutron-gamma radiation field. A peak caused by the $^6$Li($n$, $\alpha$) reaction appeared on a Compton electron background produced by gamma rays in the pulse-height spectrum (PHS) of $^6$Li-glass while only the gamma-ray background was observed for $^7$Li-glass. Because one could assume that the gamma-ray responses of both scintillators were almost identical, the gamma-ray background could be removed by subtracting the PHS of $^7$Li-glass from that of $^6$Li-glass. The total number of counts in the subtracted spectrum, which was equal to the total...
-number of $^6\text{Li}(n,\alpha)^3\text{H}$ events in the scintillator, corresponded to the TPR of $^6\text{Li}$.

In the experiment, NS15 (6.3% $^6\text{Li}$ from Nikon Corporation) and BS15, corresponding to $^6\text{Li}$ and $^7\text{Li}$ enriched glass scintillators, respectively, 10 mm in diameter and 0.3 mm in thickness, were used. The number of lithium atoms in each scintillator was determined by weighing with chemical and isotopic analyses for scintillators made in the same lot. We coupled the scintillators to 13-mm-diam photomultiplier tubes (Hamamatsu Photonics Corporation R647-02) using 10-mm-diam x 4-mm-thick quartz light guides with silicone gel. The outer diameter and the total length of the detector assembly were 19 and 175 mm, respectively.

It was impractical to achieve a perfect match of the gains of both detector channels by electronic tuning, and there was a difference in gamma-ray detection efficiencies of both scintillators. Thus, it was necessary to adjust the gains (horizontal axis in PHS) and efficiencies (vertical axis in PHS) of the $^6\text{Li}$- and $^7\text{Li}$-glass scintillators in PHS in a numerical way. We performed the adjustment using PHS in the region below the $^6\text{Li}(n,\alpha)^3\text{H}$ peak that included only the gamma-ray background. The PHS of $^7\text{Li}$-glass was converted with a linear transformation and adjusted by fitting to the PHS of $^6\text{Li}$-glass with the least-squares method.

For the case of thin scintillators, in addition to the PHS subtraction, a correction for the edge effect (distortion of pulse height due to the wall escape of alpha particles and/or tritons) was necessary. This effect was taken into account in estimation of tritium production events. However, corrections for flux perturbation were neglected because our scintillator was thin enough to neglect the self-shielding effect.

Systematic errors came mainly from those of the source neutron yield and the number of $^6\text{Li}$ atoms, which in the $^6\text{Li}$-glass scintillator (NS15) were determined by isotope dilution analysis within an uncertainty of $\pm 0.5\%$. Random errors were due to a statistical error of the source neutron counting of $<0.5\%$ and a statistical error of the $^6\text{Li}(n,\alpha)^3\text{H}$ reaction counting of 0.3 to 1.1%. The error of the fitting method in the pulse-height subtraction was estimated to be simply the statistical error.

**In-System Neutron Spectrum**

A detector for in-system neutron spectrometry basically requires the following characteristics: (a) the detector should be made as small as possible to reduce the perturbation due to the detector and to obtain good spatial resolution, (b) the response of the detector must be insensitive to the direction of the incident neutrons, and (c) gamma rays associated with D-T neutrons can be rejected. Two types of miniature detectors measuring the proton-recoil spectrum have been developed.

**Small Sphere NE-213 Scintillation Detector**

A small spherical detector with an NE-213 liquid scintillator was developed to measure the neutron spectrum inside the test blanket at energies of several hundred kilo-electron-volts to 15 MeV. A 14-mm-diam spherical NE-213 liquid scintillator was used for the probe of neutron detection. The NE-213 liquid of $1.38 \times 10^3 \text{ mm}^3$ was contained in a spherical cell of 1-mm-thick Pyrex glass. A sectional view of the detector is shown in Fig. 8. The scintillator was mounted on an R647-02 photomultiplier tube with an 11-mm-diam and 5-mm-long quartz light guide. One side of the light glass was cut in a spherical shape to fit the surface of the glass cell. The glass cell and the light guide were coated with the NE-560 reflector paint made of MgO.

![Fig. 8. Sectional view of small sphere NE-213 scintillation detector.](image-url)
The output signals from the anode of the photomultiplier tube were split into two signals for two delay-line amplifiers with different gains to supply two different runs simultaneously. The rise-time discrimination technique was used for neutron gamma-ray separation. In addition, a gain-stabilizer was applied for stable measurement, because gain drift of the system was frequently found as a result of counting rate variation or temperature change. Light from a light-emitting diode (LED) was introduced from the side of the light guide using optical fiber. The overall gain of the system was monitored by a peak produced by the light of LED and fed back to the high-voltage supply.

We converted PHS to the proton-recoil energy spectrum using the light output efficiency data. Two proton-recoil energy spectra obtained with different gains were combined at ~2 MeV, and the result was unfolded by the FORIST code by using the neutron response matrix. The Monte Carlo method separately calculated the response matrix in which the responses in specially important energy regions, i.e., 13.6 to 14.8 MeV, were replaced by the responses directly measured with the D-T neutron source. Also, the absolute efficiency was normalized with the absolute neutron flux determined by the associated alpha particles.

The accuracy of the neutron response in the neutron energy range of 13.6 to 14.8 MeV was ~2% from the experimental confirmation. However, the errors of the proton-recoil responses from the higher energy neutrons accumulated in the lower proton-recoil energy range because the response function error of 2% was enhanced in the lower energy recoil spectrum. The calculated neutron response function corresponding to the proton energy below ~2 MeV gave a relatively poor representation because of mixing of the alpha-particle response from carbon reactions by high-energy neutrons. Therefore, in the case where 14-MeV neutrons were dominant, the error of the unfolded spectrum in the range below the 14-MeV peak is attributed to the error of the used response. If the flux above 10 MeV was, for example, ten times larger than the flux below 10 MeV, the uncertainty of the proton spectra in the range of 6 to 10 MeV might become ~10 to ~20% at maximum. On the other hand, the energy calibration error also affected the unfolded results. The error of 2% in the slope of the energy axis influenced the unfolded spectrum by ~3% above 10 MeV and <2% for the 1- to 10-MeV range, respectively. The overall error came to 4% for the flux above 10 MeV and 10 to 20% below 10 MeV depending on the neutron spectrum measured.

**Small PRC**

A small proton-recoil gas proportional counter (PRC) with hydrogen gas was developed by Bennett to cover the energies of a few kilo-electron-volts to 2 MeV. The counter body is fabricated from 0.41-mm-thick Type 304 stainless steel alloy. The inner diameter and the effective length are 18.2 and 127 mm, respectively, as shown in Fig. 9. Field tubes whose length matches that of the internal bridges define the active counter volume by the electric field calculated with a finite difference program. Counter walls and end plugs are crown welded; the pumpout stub is hard soldered. Other joints are made with soft solder. To encompass the neutron spectrum over the energy range from 1 keV to 2 MeV, we used two different gas fillings for identical counters: hydrogen at 5.5 atm with 1% CH₄ for the low-energy component and a 50-50 mixture of hydrogen and argon at 8.8 atm for the upper energy component. Argon increases the stopping power, thereby reducing the proton-recoil range. The filling system employed a commercial cartridge filter to ensure the removal of trace contaminants; in most other respects, standard design practice was followed. A small amount of ³He added to the gas mixture afforded energy calibration in a thermal neutron environment, yet another energy calibration point was available from the lithium and iron resonances. The neutron-detecting efficiency

![Diagram of Small PRC](image-url)

**Fig. 9. Small PRC.**
was calculated from the number of hydrogen atoms and the elastic neutron-proton cross section.

Signals from the counter that corresponded to the rise-time information and the energy were fed to both the fast and slow amplifiers, respectively. Making use of the two-dimensional display mode of the multichannel analyzer system, we set the gain of the slow channel in relation to a stable pulser and adjusted the gain of the fast amplifier to optimize shape discrimination between pulses generated by proton recoils and by Compton electrons. We set the high voltage for the hydrogen counter at 3000 to 4200 V for the five independent measurements to obtain the neutron spectra at the energy region from a few kilo-electron-volts to 150 keV. At the energy region from 150 keV to 1 MeV, we set the high voltage for the hydrogen/argon counter at 2900 to 3600 V from the three different runs. In the case where the line source was used, the high-voltage supply scheme with ramped sweep was developed to get the data for different applied voltages in a single measurement. At first, two-dimensional spectra of the rise-time energy pulses were obtained in 64 × 128 channels from output signals of the fast and slow amplifiers, and neutron pulses were discriminated by the rise-time window. We derived the neutron energy spectrum by unfolding PHS using the differential method.

Possible error sources of this technique are gas pressure (hydrogen atomic number), the neutron-proton elastic scattering cross section, and calibration of the recoil energy. They are expected to be <1% for each and usually ~10% for the fitting error for differentiation of the recoil-proton spectrum due to counting statistics. Moreover, influence of wall end effects in the upper energy region and ambiguity of ionization energy (the W value) in the lower energy region should be accounted for. For each energy point in the obtained flux, the overall error was expected to be 3 to 5% above 100 keV and ~10% below 100 keV, respectively.

Foil Activation Technique

This passive dosimeter method has, in general, several advantages: (a) the activation technique is well established, (b) the responses of the activation detectors have been well studied and evaluated in the dosimetry files, and (c) the small size of the detector foils allows good spatial resolution and less perturbation of the field. Relatively small experimental error enables neutronics discussion to be precise. The data have been used to validate the other experimental data with both off- and on-line techniques. The reliability as a detector for the spectrum indices has been achieved by improvement of the accuracy in the dosimetry cross-section data.

Throughout the series of experiments, a common set of reactions was used for making comparisons between systems. In Table III, selected reactions are listed with their effective threshold energies. In particular, the general use of the common dosimetry reactions, e.g., $^{37}$Al($n,\alpha$)$^{24}$Na, $^{60}$Fe($n,p$)$^{56}$Mn, $^{58}$Ni($n,p$)$^{58}$Co, $^{58}$Ni($n,2n$)$^{57}$Ni, $^{90}$Zr($n,2n$)$^{89}$Zr, $^{93}$Nb($n,2n$)$^{92}$mNb, $^{115}$In($n,n'$)$^{113}$mIn, $^{197}$Au($n,\gamma$)$^{198}$Au, etc., over all the

### TABLE III

<table>
<thead>
<tr>
<th>Reactions</th>
<th>Half-Life</th>
<th>Abundance (%)</th>
<th>Gamma-Ray Energy (keV)</th>
<th>Gamma-Ray Branching (%)</th>
<th>Threshold (MeV)</th>
</tr>
</thead>
<tbody>
<tr>
<td>$^{27}$Al($n,\alpha$)$^{24}$Na</td>
<td>15.02 h</td>
<td>100.0</td>
<td>1368.6</td>
<td>100.0</td>
<td>5</td>
</tr>
<tr>
<td>$^{47}$Ti($n,x$)$^{46}$Sc</td>
<td>83.83 days</td>
<td>100.0</td>
<td>889.3</td>
<td>99.98</td>
<td>4</td>
</tr>
<tr>
<td>$^{47}$Ti($n,x$)$^{41}$Sc</td>
<td>3.341 days</td>
<td>100.0</td>
<td>159.4</td>
<td>68.0</td>
<td>1.5</td>
</tr>
<tr>
<td>$^{47}$Ti($n,x$)$^{40}$Sc</td>
<td>1.821 days</td>
<td>100.0</td>
<td>983.5</td>
<td>100.0</td>
<td>5</td>
</tr>
<tr>
<td>$^{53}$Mn($n,\gamma$)$^{53m}$Mn</td>
<td>2.579 h</td>
<td>100.0</td>
<td>846.8</td>
<td>98.9</td>
<td>---</td>
</tr>
<tr>
<td>$^{54}$Fe($n,p$)$^{54}$Mn</td>
<td>312.2 days</td>
<td>5.8</td>
<td>834.8</td>
<td>99.98</td>
<td>2</td>
</tr>
<tr>
<td>$^{54}$Fe($n,p$)$^{54}$Mn</td>
<td>2.579 h</td>
<td>91.72</td>
<td>846.8</td>
<td>98.9</td>
<td>5</td>
</tr>
<tr>
<td>$^{55}$Ni($n,p$)$^{58}$Co</td>
<td>70.92 days</td>
<td>68.26</td>
<td>810.8</td>
<td>99.5</td>
<td>2</td>
</tr>
<tr>
<td>$^{58}$Ni($n,2n$)$^{57}$Ni</td>
<td>1.503 days</td>
<td>68.27</td>
<td>1377.6</td>
<td>77.9</td>
<td>12.5</td>
</tr>
<tr>
<td>$^{59}$Co($n,\alpha$)$^{56}$Mn</td>
<td>2.579 h</td>
<td>100.0</td>
<td>846.8</td>
<td>98.9</td>
<td>6</td>
</tr>
<tr>
<td>$^{60}$Co($n,2n$)$^{58}$Co</td>
<td>70.92 days</td>
<td>100.0</td>
<td>810.8</td>
<td>99.5</td>
<td>10</td>
</tr>
<tr>
<td>$^{64}$Zn($n,p$)$^{64}$Cu</td>
<td>12.70 h</td>
<td>48.6</td>
<td>511.0</td>
<td>74.2</td>
<td>1.5</td>
</tr>
<tr>
<td>$^{60}$Zr($n,2n$)$^{59}$Zr</td>
<td>3.268 days</td>
<td>51.45</td>
<td>909.2</td>
<td>99.01</td>
<td>12</td>
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<tr>
<td>$^{93}$Nb($n,2n$)$^{92}$mNb</td>
<td>10.15 days</td>
<td>100.0</td>
<td>934.5</td>
<td>99.0</td>
<td>9</td>
</tr>
<tr>
<td>$^{115}$In($n,n'$)$^{113}$mIn</td>
<td>4.486 h</td>
<td>95.7</td>
<td>336.3</td>
<td>45.8</td>
<td>0.34</td>
</tr>
<tr>
<td>$^{197}$Au($n,\gamma$)$^{198}$Au</td>
<td>2.694 days</td>
<td>100.0</td>
<td>411.8</td>
<td>95.5</td>
<td>---</td>
</tr>
<tr>
<td>$^{197}$Au($n,2n$)$^{196}$Au</td>
<td>6.183 days</td>
<td>100.0</td>
<td>355.6</td>
<td>87.0</td>
<td>8.5</td>
</tr>
</tbody>
</table>
experimental systems provided beneficial neutronics information for the systematic analysis through comparative study concerning material and geometrical configurations.

The foils were irradiated typically for ~10 h with D-T neutrons, and total neutron yields were $4.3 \times 10^{16}$ neutrons for the irradiations that used experimental drawer channels. After irradiation, gamma rays from the irradiated foils were measured in parallel with several germanium detectors. First, the detecting efficiency of one detector was calibrated accurately as a standard, and the other detectors were calibrated relative to the standard detector by using the same samples irradiated in the experiment. This scheme reduced the whole counting time for measuring activations of many foils without losing accuracy for the detector efficiencies.

The decay data for the half-life (decay constant) and the gamma-ray branching ratio were taken from Ref. 21. The neutron strength $Y_n$, is the time-averaged neutron production monitored with the fission chambers calibrated by the associated alpha-particle scheme. The natural background gamma-ray subtraction can be critical for weak activities located in deep positions in the assemblies. In that case, the contributions were carefully subtracted from the net peak counts, correcting the counting loss due to the coincidental sum-peak in the cascade gamma rays.

Major sources of the errors for the reaction rates were the gamma-ray counting statistics (0.1 to several percent) and the detector efficiency (2 to 3%). The error for sum-peak correction was estimated to be <2% depending on the decay mode and the fraction of the multiple gamma-ray cascade. The error of the half-life of the activity was reflected in the error for the decay correction. If the half-life was accurate, the error for the saturation factor should have been <1% even for the short half-life activities. The other errors associated with foil weight, gamma-ray self-absorption, irradiation time, cooling time, and counting time were negligibly small. The error for neutron yield was estimated to be 2%. The overall error of the reaction rate, therefore, ranged from 3 to 6%.

**Gamma-Ray Heating Rate**

**Interpolation Method by TLDs**

Measurements of gamma-ray heating that use TLDs have been performed for the fast breeder reactor mock-up experiments, but there have been no measurements for fusion neutrons. The interpolation method for gamma-ray heating measurement that uses TLDs was proposed by Tanaka et al. The relation between the absorbed dose in a medium and that in a TLD is written as

$$D_m = f(M, \text{TLD}, E_\gamma)^{-1} \cdot D_{\text{TLD}},$$

where

- $D_m$ = absorbed dose in a medium $M$
- $D_{\text{TLD}}$ = absorbed dose in a TLD
- $f(M, \text{TLD}, E_\gamma)$ = conversion factor
- $E_\gamma$ = gamma-ray energy.

According to Tanaka et al., an absorbed dose in a TLD overestimates that in the surrounding medium with an atomic number smaller than that of the TLD while it underestimates for the opposite case. In addition, the measured values of dose for different types of TLDs increase monotonically with the atomic number. Thus, the absorbed dose in a medium can be obtained as an interpolated or extrapolated value at the atomic number of the medium of interest when the measured values of TLDs are plotted with the effective atomic number of TLD. The great advantage of this method is that it does not require any gamma-ray spectrum information. However, the neutron contribution should be subtracted by the assistance of calculation.

We performed the measurement with four kinds of TLDs: $^7$LiF(MgI) (99.93% enrichment), $\text{Mg}_2\text{SiO}_4$(Tb), $\text{Sr}_2\text{SiO}_4$(Tb), and $\text{Ba}_2\text{SiO}_4$(Tb). The TLD powder was sealed in a 2-mm-diam $\times$ 12-mm-long Pyrex glass ampoule. Four ampoules for each TLD were enclosed in a 0.05-mm-thick polyethylene bag and set inside the test blanket. The irradiated TLDs were cooled down in one day, and the thermoluminescence from them was measured by the TLD reader with a constant temperature increase rate. The responses of TLDs and the reader were calibrated by standard samples irradiated with the calibrated $^{60}$Co gamma-ray source. The thermoluminescence from the neutron interactions was subtracted from total observed thermoluminescence. The neutron thermoluminescence contribution was estimated with the calculated neutron response of TLDs and the calculated neutron spectrum at the detector position. Figure 10 shows the calculated neutron response for each TLD (Ref. 23). The neutron contribution was usually large at the front of the test blanket. The gamma-ray heating rate of the interested material of the assembly was interpolated from the gamma-ray heating rate above several TLDs with respect to the effective atomic number.

The experimental errors were due to dispersion of thermoluminescence measurement of 2 to 15%, gamma-ray response calibration of 2 to 10%, neutron response subtraction of 10 to 20%, and interpolation of <20%. Finally, the overall error was 20 to 30%.

**NE-213 Detector with SWF**

The spectrum weighting function (SWF) method can be applied to measure in-system gamma-ray heating rate distribution. The NE-213 scintillation detector separates the gamma-ray interactions from those of neutrons via pulse-shape discrimination, and the
gamma-ray response of the detector gives the energy information. The kerma factor for gamma rays is obtained by the mass energy absorption coefficient, so the gamma-ray heating rate is obtained by using the following weighting function.

The heating rate $H$ is given by

$$ H = \int K(E_\gamma) \cdot \Phi(E_\gamma) \ dE_\gamma , \tag{5} $$

where $K(E_\gamma)$ is the kerma factor of surrounding material and $\Phi(E_\gamma)$ is the photon flux with energy $E_\gamma$. The kerma factor is expressed by the recoil electron spectrum produced from one photon in the medium, and the electron spectrum is related to the photon response matrix $R$ for the scintillator:

$$ H = \int K(E_\gamma) \cdot \Phi(E_\gamma) \ dE_\gamma \\
= \int \left[ \int R(E_\gamma, E_e) \cdot G(E_e) \ dE_e \right] \cdot \Phi(E_\gamma) \ dE_\gamma . \tag{6} $$

Exchanging the order of integration, we get

$$ H = \int \left[ \int R(E_\gamma, E_e) \cdot \Phi(E_\gamma) \ dE_\gamma \right] \cdot G(E_e) \ dE_e \\
= \int C(E_e) \cdot G(E_e) \ dE_e . \tag{7} $$

In Eq. (7), $C(E_e)$ is an electron energy spectrum measured by the NE-213 scintillator. The SWF $G(E_e)$ is determined by both gamma-ray response matrices and the kerma factor. From the foregoing relation, we obtained the gamma-ray heating rate by the small NE-213 scintillation detector, assuming that the detector would not perturb the gamma field.

First, the gamma-ray response was calculated by the Monte Carlo code and renormalized to the measured response by using a standard gamma-ray source. Second, the weighting function was determined by solving the following integral equation:

$$ K(E_\gamma) = \int R(E_\gamma, E_e) \cdot G(E_e) \ dE_e . \tag{8} $$

The successive approximation method was applied to obtain the solution $G(E_e)$.

The error in the deduced heating rate was estimated by the uncertainties of the source neutrons, weighting function, gamma-ray response matrix, and kerma factor. The differential errors of the response function and the kerma factor were assumed to be 20 and 10%, respectively. Those produced the errors of the weighting function independently under the boundary condition of Eq. (8). The error of the weighting function due to uncertainty of the solution was estimated from the range of the results obtained by a different iteration, and a systematic error of 5 to 50% was obtained. These errors propagated with the weight of PHS to the error of the gamma-ray heating rate by 10% from the first two sources and 3 to 10% from the weighting function, respectively. Finally, the overall error became 5 to 15%, excluding the detector perturbation. \(^{25}\)

**Discussion**

The measurement techniques for TPR are compared in Table IV. The active (on-line) methods had a somewhat larger experimental error while tritium contamination of the sample of the passive method was sometimes found in scintillation counting. In the on-line method, the NE-213 method could be biased by the cross-section data of the $^7\text{Li}(n, n'\alpha)$ reaction. In the LSC method, finding the source of contamination and judging whether or not the sample is contaminated is usually very difficult. Though the spatial resolution was worse, the zonal method had the smallest uncertainty of all the techniques because of a good signal-to-background ratio. Figure 11 shows the comparison of the results among three different techniques measured in the Li$_2$O test blanket \(^4\) in Phase-IIA. The discrepancies of these techniques were 5 to 10% and almost within the experimental errors. In this case, the distribution was smooth, and the differences were expected to be small. Figure 12 gives an appropriate comparison of the methods with necessary measurement (irradiation) time and measurable level of TPR. The on-line detector was more sensitive, and its measurement time was shorter. In the passive method, the zonal method covered a wide area while the TLD self-irradiation method showed the lowest sensitivity.

For in situ spectrum measurement techniques, only the recoil proton spectrometers were applied with a scintillator and gas proportional counter. Otherwise, the foil activation technique could be used as the spectrum
### TABLE IV
Comparison of Techniques Applied to TPR Measurement

<table>
<thead>
<tr>
<th>Method</th>
<th>Principle</th>
<th>Applicability</th>
<th>Error and Comments</th>
</tr>
</thead>
<tbody>
<tr>
<td>Passive method (postprocessing) Li-foil</td>
<td>Tritium measurement by LSC technique</td>
<td>Large fluence requirement especially for (^{\text{7}})Li</td>
<td>~5%</td>
</tr>
<tr>
<td>Li(_2)O pellet</td>
<td>The same as above, but use the same material as test blanket</td>
<td>Large fluence requirement, especially for (^{\text{7}})Li</td>
<td>Protection for tritium contamination is required ~5%</td>
</tr>
<tr>
<td>Li(_2)O zonal</td>
<td>The same as above, but use the part of test blanket as a block sample</td>
<td>No self-shielding, modest fluence, especially for (^{\text{7}})Li</td>
<td>Protection for tritium contamination is required ~3%</td>
</tr>
<tr>
<td>LiF TLD</td>
<td>Self-dose measurement by decay of produced tritium</td>
<td>Long measurement time for self-irradiation</td>
<td>Better signal-to-background ratio 10 to 20%</td>
</tr>
<tr>
<td>Active method (on-line) NE-213</td>
<td>(Cross-section data) (\times) (measured spectrum)</td>
<td>Counting rate limit</td>
<td>Relative measurement only</td>
</tr>
<tr>
<td>Li-glass</td>
<td>Direct counting of reaction</td>
<td>Counting rate limit, high neutron-gamma ratio is required for measured field</td>
<td>5 to 8% Dependence on used cross-section data void effect should be considered</td>
</tr>
</tbody>
</table>

As for gamma-ray heating measurement, Fig. 14 shows the comparison with the results of the TLD interpolation method in the Phase-II beryllium sandwich experiment. In this case, the data of the TLD method were extrapolated to the effective atomic number of Li\(_2\)O, as shown in Fig. 15. Figure 15 shows that the extrapolation error was very large for low effective atomic numbers, so that error results in a large discrepancy between both the methods in Fig. 14. For light materials, there was another problem with the TLD method; i.e.,

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**Fig. 11.** Comparison of the results of the \(^{\text{6}}\)Li TPR measured by different techniques in the Li\(_2\)O test blanket.

**Fig. 12.** Relation of measurement time and measurable range of tritium production measurement methods.
the error for the subtraction of the neutron thermoluminescence component was larger. In the TLDs with small effective atomic numbers, neutron heating could be greater than gamma-ray heating.

The detector perturbation is also of concern for the in-system experiments. This is separated into the void effect and the flux deformation effect. The former is due to a lack of material around the measured position, and the calculation for the benchmark usually does not take such voids into consideration. The latter case is caused by different materials from the medium, and then the calculation also does not take into account such details. These effects can cause differences between the calculated and the measured values. For example, the void effect of the NE-213 detector corresponded to a lack of neutron attenuation in the ~14-mm-thick material; that is a 5 to 7% increase of measured fast neutron flux above 10 MeV but 1 to 2% of flux above 1 MeV inside the homogeneous Li2O assembly, according to the computational estimate. For the Li-glass detector, which is sensitive to low-energy neutrons, the void effect was neglected, because the low-energy flux

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**Fig. 13.** Energy range of neutron spectrum measurement techniques.

**Fig. 14.** Gamma-ray heating rates measured with the TLD and the SWF methods.

**Fig. 15.** Extrapolation for gamma-ray heating rate with effective atomic number in the Phase-IIA beryllium sandwich assembly.
is not changed by the void in uniform flux distribution. However, in the case that the neutron flux changes rapidly, e.g., at the material boundary, there could be a large effect.4

On the other hand, the flux deformation depends on the detector material and the measured quantity; the detector self-shielding is such a deformation effect for the tritium production measurement. However, this effect can be significantly reduced by the use of a very thin scintillator and the choice of low lithium content. For the $^{6}$Li$_{2}$O pellet method, this effect is corrected by ~4% in the Li$_{2}$O by using the calculation while the zonal method is free from this effect because the detector material is the same as the medium. In the case of the $(n, \gamma)$ reaction measurement, the gold foil thickness was also chosen to be as thin as 1 $\mu$m to reduce the self-shielding effect.

CONCLUDING REMARKS

The characteristics of an experimental system for an engineering-oriented neutronics experiment were examined. The closed system surrounding a neutron source by tritium breeding material simulated well the reflected neutron component incident to the test blanket. Moreover, the calculated volume-integrated spectrum of the annular blanket system, i.e., an integrated nuclear parameter, was very similar to one calculated for a reactor blanket. The block system adopted for the test assembly gave good flexibility for a variety of configurations of materials. The current measurement techniques provided the data with an error range of 3 to 5% for tritium production, 5 to 10% for the neutron spectrum, 3 to 6% for the activation reaction, and 10 to 20% for the gamma-ray heating rate. Consequently, the experimental method and the measurement techniques for the engineering benchmark were established by this series of experiments.

In fusion reactor development, the obtained error ranges indicate the current limitation of experimental confirmation of the design parameters. For example, confirming TPR within 3% is difficult, so that it seems that the 3% error is a lower limit of the design uncertainty. For a future experiment, most of the current techniques could not be applied for measurements in a fusion device where high temperatures and high magnetic fields exist. The active method has a problem due to electric components and insulators. Scintillators do not work for such high temperatures. The gas counter needs research and development for use in high temperatures and magnetic fields. The passive method of tritium production also does not work under high temperature because of tritium escape. Currently, only a foil activation technique can work under the fusion device conditions without any modification. Thus, the development of new techniques is needed for nuclear technology experiments in a fusion reactor such as ITER.

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