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ARTICLE

Measurement and Calculations of ^{17}N Concentration in the Main Steam System of the BWR Plant

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The radioactive concentration of ^{17}N in the main steam system of the boiling water reactor (BWR) plant has been evaluated for the aging countermeasures and the planning of the plant decommissioning of the BWR. The neutron flux on the surface of the main steam pipe under the plant operation was measured by using a solid-state track detector. The flux due to ^{17}N has been obtained by the Monte Carlo code calculations. It has been found that the concentration of ^{17}N at the reactor pressure vessel outlet nozzle of the main steam pipe is approximately 3 Bq/cm^3. The concentration has turned out to have small dependency on nuclear thermal power and to be almost common among the BWR plants.

KEYWORDS: ***^{17}N concentration, Solid-State Track Detector, main steam equipment, neutron flux, MCNP5***

I. Introduction

Old nuclear power plants in Japan have been operating for more than 30 years. Nowadays, some of the nuclear power plants arrive at the stage where the aging countermeasures and planning of the plant decommissioning are necessary to be investigated. One of the issues for the investigations is evaluation of the radioactive concentration due to the activation reaction by neutron. As for around the reactor of the boiling water reactor (BWR) plants, relatively many evaluations have been performed. On the other hand, the evaluations for the main steam system of the BWR plants have hardly been published. Major nuclide for neutron source in the main steam system is ^{17}N. ^{17}N, produced by the activation of the coolant due to the fast neutron induced $^{17}O(n, p)^{17}N$ in the reactor core, is carried into the turbine equipment with the steam. The radioactive concentration of ^{17}N has not been investigated so far because ^{17}N is not so important nuclide in the shielding design of a BWR plant.

In this study, the radioactive concentration of ^{17}N in the main steam system of the Hamaoka nuclear power station unit-5 (H-5) has been evaluated. H-5, an advanced boiling water reactor (ABWR), is the highest output nuclear power plant in Japan. The output power of H-5 is 3926 MWt. Since ^{17}N is difficult to be directly measured by the main steam sampling method due to its short half-life. The concentration of ^{17}N was evaluated by the neutron flux measurement and calculations.

II. Measurement

1. Measurement Configuration

The steam generated in the reactor core is carried in the main steam system pipes via Main Steam Isolation Valve (MSIV), High Pressure Turbine (HPT), Moisture Separator and Heater (MSH), Combined Intermediate Valve (CIV) and Low Pressure Turbine (LPT), and arrives at the steam condenser finally. ^{17}N is also carried to the equipments of the main steam system with the steam. Therefore, the radioactive concentration of ^{17}N can be measured at any places within area of the main steam system. However, measurement locations are necessary to be selected considering following points;

(a) to be at the upper stream of the main steam system because of the short half-life of ^{17}N,
(b) to be simple configuration to simplify the calculation model and
(c) to be easy access of workers.

Considering above points, measurement locations were selected. **Figure 1** shows the outline of the main steam system and the measurement locations. As shown in **Fig. 1,** the neutron flux measurement was performed at two locations (A and B) in the main steam pipe areas of H-5.

The location A is located at the upper stream of the main steam system and near the 2nd MSIV, where the measurement sensitivity is expected to be relatively high because of the high radioactive concentration of ^{17}N. As shown in **Fig. 2,** there are 4 main steam pipes around the location A. Measurement was performed for 2 pipes among the 4 pipes. Radiation detectors were located at about 7 m apart from the 2nd MSIV toward down stream side. The measurement time was approximately 3 days.

The location B is near the CIV near the low pressure turbine which is one of subject equipments for the measurement, and suitable for easy measurement. There is one main steam pipe around the location B. Radiation detectors were located at about 1 m apart from the CIV toward the down stream side. The measurement time was approximately 20 days. For both the locations A and B, radiation detectors were located on the right, left, upper, and

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lower sides of the steam pipe as shown in **Figs. 2 and 3**.

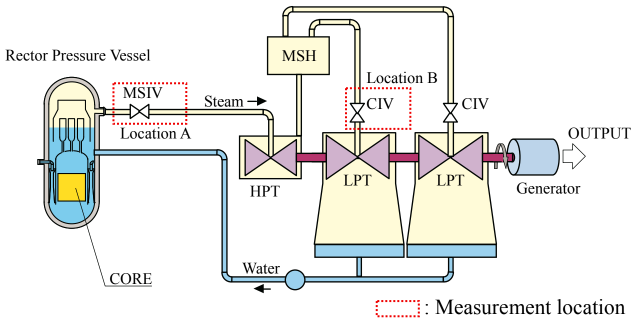

Fig. 1 Location of neutron flux measurement

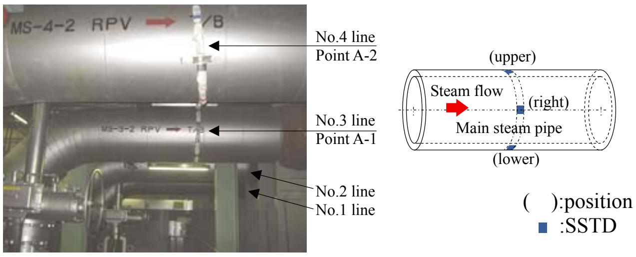

Fig. 2 Measurement points at the location A

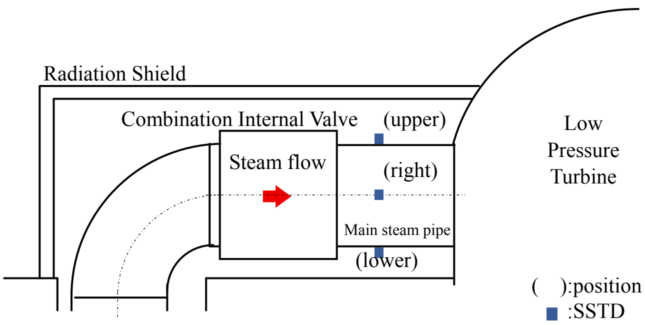

Fig. 3 Measurement points at the location B

2. Detector

A Solid-State Track Detector (SSTD) was used for the measurement of the neutron flux. The SSTD was selected by reasons that the SSTD can be easily attached at the pipe due to its small size and can identify neutrons from the source in the main steam pipe because of its good direction sensitivity. It is difficult to obtain information of neutron source from low energy neutrons because scattered components of high energy neutrons are included. Therefore, high energy neutrons are necessary to be measured to get directly information of the neutron source. The SSTD has 2 channels. One channel is the PE channel for fast neutron region, and the other channel is PA channel for the fast / thermal / epithermal neutron region. The PE channel works with high-density polyethylene converters, and the PA channel works with polyamide (Nylon-6) converters. These converters change neutrons to protons or alpha particles, and these particles make a track on the SSTD. Information of the neutron source can be obtained from the data of the PE channel. The SSTD requires a response between the neutron flux and the track numbers in order to convert the tracks to the neutron flux. The PE channel is sensitive to neutron above 100 keV. Therefore, the neutron energy dependent responses of the PE channel were measured in advance. Three kinds of mono-energetic neutrons, i.e., 144 keV, 565 keV, and 5.0 MeV ones were used for the measurement of the neutron energy dependent response.

The obtained responses of the SSTD are shown in **Table 1** and **Fig. 4**. The obtained tracks on the SSTD were converted to the neutron flux using the responses.

Table 1 Neutron energy dependent response of the SSTD

Neutron energy	Response (tracks・fluence^{-1}・cm^{-2})
144 keV	2.5E-6 (2.8E-7)
565 keV	6.1E-5 (1.1E-5)
5.0 MeV	1.4E-4 (2.0E-5)

※( ) experimental error

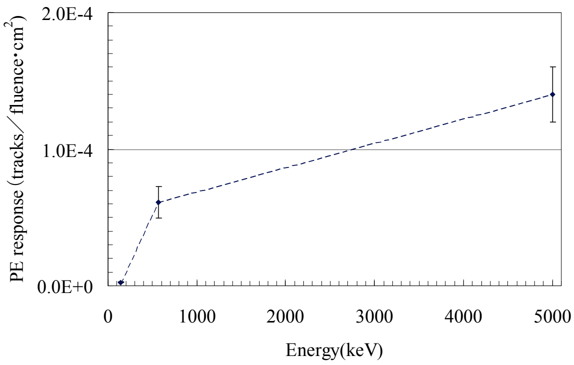

Fig. 4 Neutron energy dependent response of the SSTD

3. Results of measurement

The measured neutron flux above 100 keV which are obtained with the PE converters are shown in **Table 2**. The data shown with the symbol '—' are removed because of poor measurement accuracy. The neutron flux values at the points A-1 and A-2 are 24~37 cm^{-2}・s^{-1}. The experimental error of the measured data is estimated to be about 25%. The measured data for both points are consistent with each other within experimental errors.

As for the point B, two of four data were lost in the etching process of the SSTD. The neutron flux values above 100 keV are about 5 cm^{-2}・s^{-1}. The measured data for the point B are also consistent with each other within experimental errors.

Table 2 Measured neutron flux above 100 keV

Measurement Point		Neutron flux ($cm^{-2}\cdot s^{-1}$)
A-1	Right	31.9 ± 8.0
	Left	28.2 ± 7.1
	Upper	36.7 ± 9.1
	Lower	—
A-2	Right	30.0 ± 7.5
	Left	—
	Upper	28.7 ± 7.2
	Lower	24.4 ± 6.1
B	Right	5.46 ± 1.36
	Left	—
	Upper	—
	Lower	5.77 ± 1.44

III. Calculation of neutron flux

The neutron flux was calculated by the Monte Carlo code MCNP5[1]. **Figure 5** shows the corresponding model to the points A-1 and A-2, drawn with the MCNP input data. The experimental layout was precisely modeled. Neutron sources of the calculation model are located uniformly inside the main steam pipes. The surface tallies were set at the same as the position of attached detectors on the pipes. The tally size decided to be 100 mm × 100 mm, which is larger than the detector size (10 mm × 10 mm) in order to avoid statistical problem within realistic computing time. This prescription is justified due to fact that the deviation of the calculated neutron fluxes in the surface area is small. The neutron cross-section library FSXLIB-J33[2] was used in the MCNP calculations. Number of histories used in the MCNP calculations is 10^8. The weight window method is applied for the calculations as a variance reduction technique to reduce computing time. Main parameters used in the calculations are shown in **Table 3**.

Table 3 Calculation parameters for the MCNP

Item	Value
Tally	Surface detector (100 mm × 100 mm)
Energy bin	above 100 keV
Neutrons emission energy from ^{17}N (Ratio:%)	0.383 MeV (38%), 0.884 MeV (0.6%) 1.171 MeV (50.1%), 1.700 MeV (6.9%)
Library	FSXLIB-J33[2]

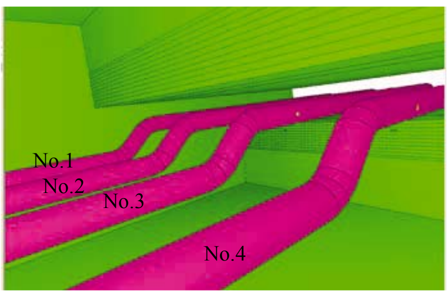

Fig. 5 Model for neutron flux calculation at points A-1 and A-2

IV. Estimation of the ^{17}N concentration

The average ^{17}N concentrations at the measured locations were estimated first by using the measured neutron flux in **Table 2** and the calculated neutron flux. **Table 4** shows the average ^{17}N concentrations at the locations A and B. The ^{17}N concentrations at the reactor pressure vessel (RPV) outlet nozzle which is usually recognized to be the start-point of ^{17}N concentration estimation of the main steam system were estimated using following equation,

$$Q = Q_0 \cdot e^{-\lambda \cdot t} \cdot \frac{V}{V_0} \quad , \tag{1}$$

where Q is ^{17}N concentration at the measurement location ($Bq\cdot cm^{-3}$), Q_0 is ^{17}N concentration at the RPV nozzle ($Bq\cdot cm^{-3}$), t is transit time between RPV nozzle and measured points (s), λ is decay constant. V and V_0 are steam specific volume ($cm^{-3}\cdot g^{-1}$) in the measurement locations and the RPV nozzle, respectively. Design data were used as the values for t, V and V_0. The estimated ^{17}N concentrations at the RPV nozzle are also shown in **Table 4.**

Comparing two estimated results of the ^{17}N concentration at the RPV nozzle, the concentration derived from the data of the location A is different from that from B. It is guessed that the difference between A and B is due to the measurement error and the difference between the real and the applied values of the parameters t, V and V_0. The result derived from the data of the location B was decided to be representative ^{17}N concentration at the RPV nozzle because the result from B is more conservative than that from A and the location B is nearer the LPT which is one of subject equipments of this study, than the location A.

The calculated ^{17}N concentration at the RPV nozzle by the radiation shielding design is also shown in **Table 4**. In radiation shielding design, the ^{17}N concentration in the reactor core is given by following equation,

$$Q_0 = \frac{K \cdot P}{Wt} \cdot \left\{ \frac{1+M}{M(1-e^{-\lambda \cdot t_{in}}) + (1-e^{-\lambda \cdot t_{ex}})} \right\} \quad , \tag{2}$$

where Q_0 is ^{17}N concentration in the core ($Bq\cdot g^{-1}$), P is thermal power (W), Wt is mass flow rate of the coolant ($g\cdot s^{-1}$), K is production rate of the ^{17}N ($Bq\cdot W^{-1}\cdot s^{-1}$), M is jet pump flow ratio (-), λ is decay constant (s^{-1}), t_{in} is transit time of the inner loop of the coolant (s), and t_{ex} is transit time of the outer loop of the coolant (s).

In the case of the ABWR, the outer loop is removed. Thus, the equation (2) can be simplified as follows:

$$Q_0 = \frac{K \cdot P}{Wt} \cdot \frac{1}{\left(1-e^{-\lambda \cdot t_{in}}\right)} \quad . \tag{3}$$

The calculated concentration at the RPV nozzle is overestimated approximately 10 times higher than the estimated one based on the measurement as shown in a column C/E of **Table 4**. The reason of the discrepancy is not investigated in detail in this paper, but it is supposed that the overestimation is owing to the uncertainty in the values of the parameters used in the calculation. Especially, the value of the parameter K is to be investigated since the applied value: 5.8 (Bq・W^{-1}・s^{-1}) was derived by the use of a semi-empirical method.

Table 4 Estimated and calculated ^{17}N concentration

Measurement Location	Estimated concentration (Bq・cm^{-3})		Calculated concentration at RPV nozzle (Bq・cm^{-3})	C/E
	At measurement location (Q)	At RPV nozzle (Q_0)		
A	1.9E+00	2.1E+00	2.7E+01	1.3E+01
B	1.3E-01	3.3E+00		8.1E+00

V. Evaluation of ^{17}N concentration for the other BWR

Applicability of the ^{17}N concentration at the RPV nozzle estimated in this paper to the other BWR plants than H-5 was studied.

The estimation of the ^{17}N concentration for the other BWRs which have a different thermal power was performed using the equation (2) or (3). The ^{17}N concentrations at the RPV nozzle of four different BWRs including H-5 were calculated. The parameters and the calculated results are shown in **Table 5**.

The ^{17}N concentration for H-5 (ABWR) is the highest value among four types of BWR. Comparing the ^{17}N concentrations in four BWR plants, the ^{17}N concentrations at the reactor cores are almost constant among four BWR plants. As for the ^{17}N concentration at the RPV nozzle, the discrepancy is only within 15%. This shows that the ^{17}N concentration is almost constant among different BWR plants, and this value might not be dependent upon the thermal power of plants.

Table 5 Calculated ^{17}N concentration for various types of BWR

Item	A (BWR)	B (BWR)	C (BWR)	H-5 (ABWR)
Parameter				
Thermal power (MW_t)	1593	2436	3293	3926
Flow rate (t/h)	2.29E+4	3.56E+4	4.83E+4	5.22E+4
Jet pump ratio (-)	1.37	1.27	2.19	-
Inner loop time (s)*1	24	21	21	18
Outer loop time (s)*1	19	18	17	-
Concentration				
Reactor core(Bq・g^{-1})	1.5E+3 *2	1.5E+3 *2	1.5E+3 *2	1.6E+3 *3
RPV nozzle (Bq・g^{-1}) *4	6.0E+2	6.4E+2	6.0E+2	7.1E+2

*1: These loop times are calculated with typical data from the plants.
*2: These data are calculated with equation (2)
*3: These data are calculated with equation (3)
*4: Attenuation by transit time between reactor core and RPV nozzle is taken into account.

In addition, applicability of the ^{17}N concentration to different types of the BWR plant was also studied based on the measured data. The measured data of ^{16}N were used instead of ^{17}N because the ^{17}N concentration in the BWR plant has hardly been measured. There are many measured data of ^{16}N since ^{16}N is a major source nuclide for the shielding design of a BWR plant. Both ^{17}N and ^{16}N are produced by the activation reaction (n, p) of the coolant due to fast neutron in the reactor core and, carried into the turbine equipments with the steam. Chemical behavior of ^{16}N is same as ^{17}N because both the nuclides are the radioisotopes of nitrogen element. Therefore, it was judged that the trend of ^{17}N can be estimated by that of ^{16}N. **Figure 6** shows relative values of dose rate due to the gamma ray emitted from ^{16}N on the surface of the main steam pipe of five different BWR plants. As shown in **Fig. 6,** the dose rates are almost constant among the BWR plants. Thus, the dose rates have small dependency on thermal power. Therefore, it has turned out that the ^{17}N concentration has also small dependency on thermal power and can be applied for different types of the BWR plants. Consequently, the ^{17}N concentration at the RPV nozzle, 3.3 Bq・cm^{-3}, is applicable to various types of the BWR plants.

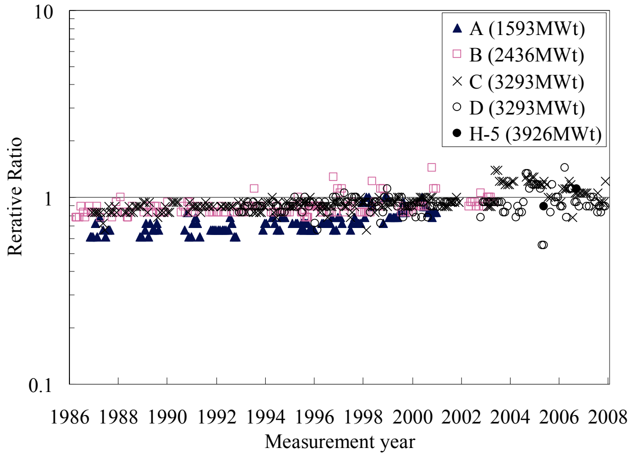

Fig. 6 Trend data of the measured gamma ray dose

VI. Conclusions

Measurements and calculations for the evaluation of the ^{17}N concentration in the main steam system of the BWR plant were performed. Main results has been obtained as follows.

(1) The ^{17}N concentration at the RPV nozzle of the ABWR plants was estimated using the measured results and calculation.
(2) The estimated ^{17}N concentration at the RPV nozzle is 3.3 Bq・cm^{-3}.
(3) The ^{17}N concentration has small dependency on thermal power and the present estimation can be applicable for various types of the BWR plants.

Reference

1) RSICC CODE PACKAGE CCC-710, MCNP5: Monte Carlo N-Particle Transport Code System.
2) K.Kosako, N.Yamano, T.Fukahori, K.Shibata and A.Hasegawa: "The Libraries FSXLIB and MATXSLIB based on JENDL-.3", JAERI-Data/Code 2003-011 (2003).