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RADIATION SHIELDING ANALYSIS OF THE FUGEN NUCLEAR POWER STATION

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ABSTRACT

The radiation shielding analysis of pressure tube type heavy water reactors was performed using the discrete ordinate transport code, DOT 3.5. The present paper describes the analytical methods developed to calculate the dose rates after shutdown in various regions such as pressure tubes, outside the primary shield and penetrations through the primary shield and calandria tank.

The calculation methods were applied to analyze the dose rates in Fugen measured by neutron and gamma-ray detectors. These calculated results are in good agreement with the measured values within a factor of approximately 3.

INTRODUCTION

In the pressure tube type reactors, there are a lot of penetrations such as pressure tubes, control rod guide tubes, and heavy water piping. And the radiations which leak out through such penetrations are dominant around the primary shields of these type of reactors. These radiations cause the activation of equipment during operation and the radiation exposure of workers after shutdown. It is important to use accurate methods to verify the shielding of such reactors. To verify the validity of shielding calculational method, it is useful to carry out shielding analysis of actual plants. In this study, the calculational accuracy of the shielding analysis of the pressure tube type reactor Fugen was checked by comparing the calculated results with the measured values.

Fugen is a 165 MWe prototype of a heavy-water moderated, boiling light-water cooled, pressure tube type reactor developed in Japan. The core of the reactor consists of a zirconium alloy pressure tubes which pass through vertical tubes forming part of calandria tank containing heavy water as the moderator. The fuel is positioned in the pressure tubes and light water is circulated through these tubes and over the fuel.

The primary (upper, lower and radial) radiation shields surrounding the calandria tank consist of steel tanks containing water and steel plates. The primary shields and their penetrations must prevent neutron activation of the external plant components. In addition they also provide sufficient shielding from the core gamma radiation after shutdown to allow workers to access to all areas outside the primary shields. The schematic view of the Fugen reactor is shown in Fig.1.

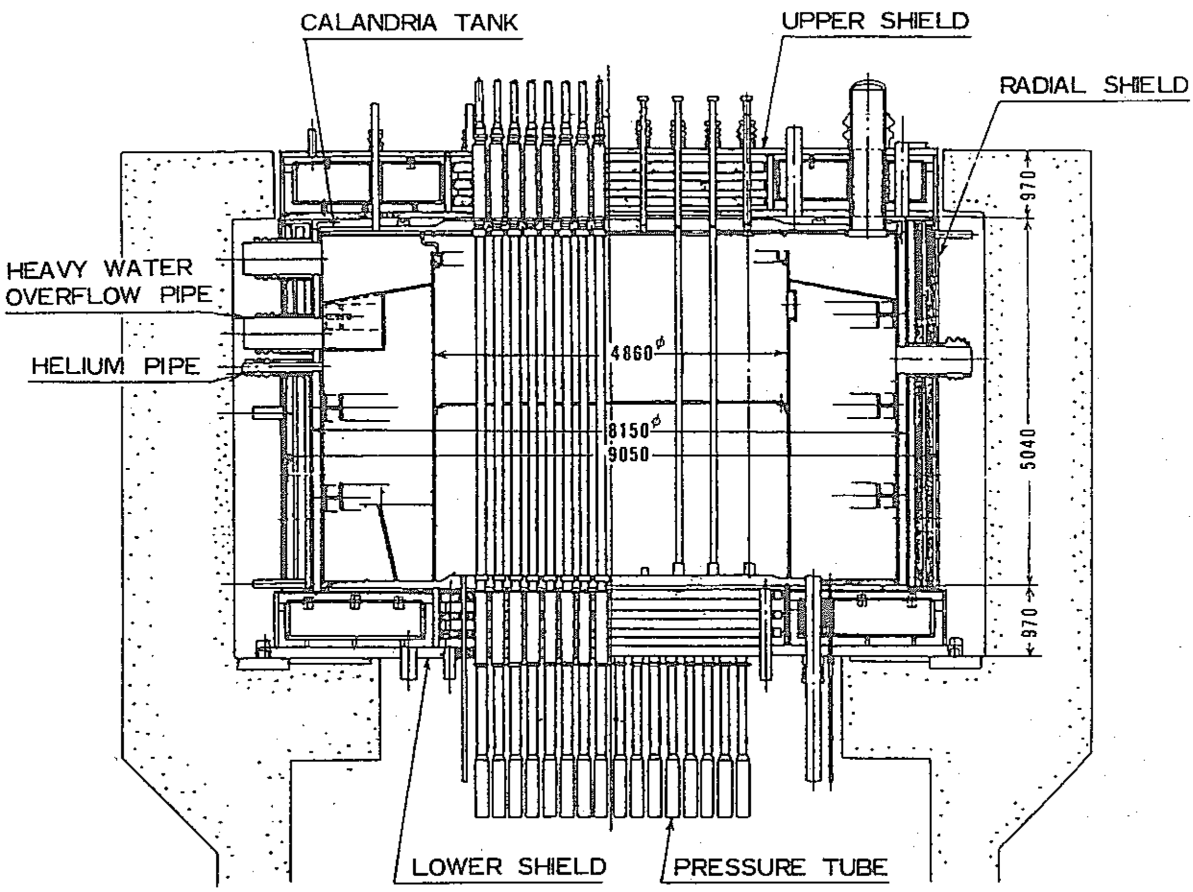

Figure 1. Main Shield Configuration of Fugen

The present paper describes the evaluation of gamma dose rates in the pressure tubes during shutdown, and reaction rates of activation foil and gamma dose rates around the primary shields of Fugen during operation and shutdown.

CALCULATIONS

Gamma Dose Rates in the Pressure Tubes

For the in-service inspection of pressure tubes, it is important to evaluate the gamma dose rates in the pressure tubes and the streaming through the lower axial shield prior to its inspection in order to know the radiation field strength and to eliminate the risk of radiation exposure to personnel.

Pressure tubes, the surrounding calandria tubes, and fuel elements are the main radiation sources used in this study. The induced radioactivity of fuels, pressure tubes and calandria tubes is calculated using the ORIGEN code. [1]

The top view of geometry used in the calculation in this study is shown in Fig.2. The calculation of gamma dose rates in the pressure tube was performed with the DOT 3.5 code using a spatial mesh of 32 for radial and 69 for axial directions.

The effects of source strength from the area beyond this region to the detector positions were neglected because these effects are about 6 ∿ 7% of total strength based on the results of the prior investigation.

The two dimensional DOT 3.5 code that can cover the reactor core region requires homogenization of complicated reactor core regions. Then, the geometry of the pressure tube to be inspected includes an actual tube and a surrounding calandria tube. The moderator zone outside this calandria tube is approximated by an annular moderator possessing the same area, and the structural materials outside of this area are also treated as annular cylinders.

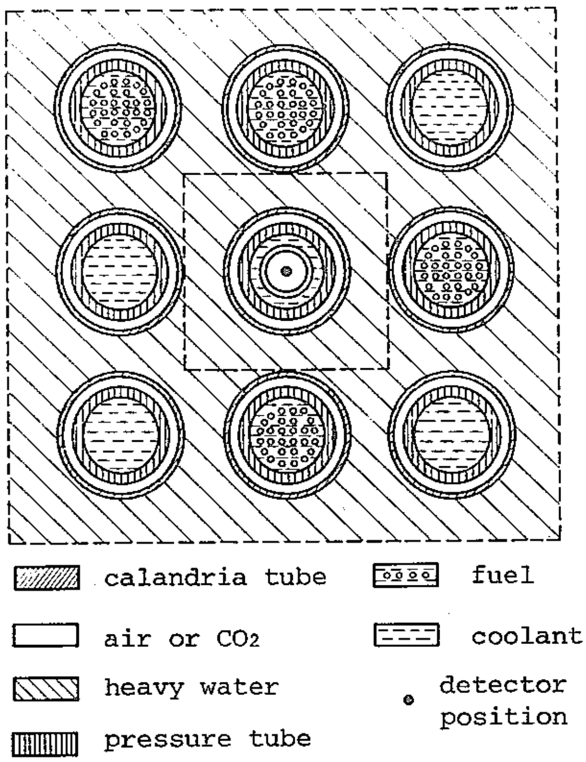

Figure 2. Top View of Geometry Used in DOT Calculations

The transport cross-sections (P_3) for component materials were obtained using the MUG code [2]. In the calculation of streaming at the lower part of the reactor, the "bootstrapping" technique was used : the dose rates below the lower axial shield were calculated using the results of gamma-ray flux at the bottom of the reactor core as the source having a biased 100-angle quadrature set in a subsequent calculation for an adjacent region.

Analyses around the Primary Shield

Procedure of analysis: The 120-group (100 neutron groups and 20 gamma-ray groups) effective macroscopic cross section library was generated by the RADHEAT-V3 code system [3], from ENDF/B-IV library for neutron and POPOP4 library for secondary gamma-ray production, as shown in Fig.3. The 18-group (13-neutron groups and 5 gamma-ray groups) cross section library was obtained by collapsing the 120-group library with the ANISN code. In the first step, using this library, the two dimensional transport calculations of neutron and gamma-ray were carried out focusing on the full reactor system in which the penetrating regions were ignored. Further, gamma-ray sources after shutdown were evaluated by the ORIGEN code using the calculated neutron spectra, and the gamma-ray flux distribution after shutdown in the full reactor system was calculated. In the second step, the streaming calculations of the penetrating regions were performed. The boundary sources for this step were obtained by editing the angular fluxes from the previous step.

Boundary sources: In the perpendicularly connected two cylindrical geometries shown in Fig.4, angular fluxes for angles with positive μ's at the internal boundary of the 1st geometry are used as a bottom boundary source of the 2nd geometry [4].

The transformation from angular flux ϕ to ϕ' (boundary source for the 2nd geometry) shown in Fig.4 and Fig.5 is performed both for axial mesh intervals of the 1st geometry to radial mesh intervals of the 2nd geometry and for angles. For the transformation of mesh intervals, boundary source φ' are obtained from the angular fluxes at the mesh interval which is most close to that of the 2nd geometry. In the case shown in Fig.5, the angular flux of the 1st geometry, $\phi(\mu,\eta,j+1)$, was employed for $\phi'(\mu',\eta',i')$. For the transformation of angles, the angular flux which also has the most close angle (μ,η) of the 1st geometry to angle (μ',η') of the 2nd geometry is assumed as the boundary source for this angle.

The transformation program used for this analysis can normalize the transformed boundary source to equalize neutron current into next leg before and after the transformation. The boundary source is multiplied by a normalization factor (f) as shown below.

Normalization factor: $f=L_1/L_2$

$$L_1 = \sum_{\mu>0} \phi(\mu,\eta,j+1)\cdot wt(\mu,\eta): \text{ Leakage before transformation}$$

$$L_2 = \sum_{\eta'>0} \phi'(\mu',\eta',i')\cdot wt(\mu',\eta'): \text{ Leakage after transformation}$$

wt: weight of quadrature

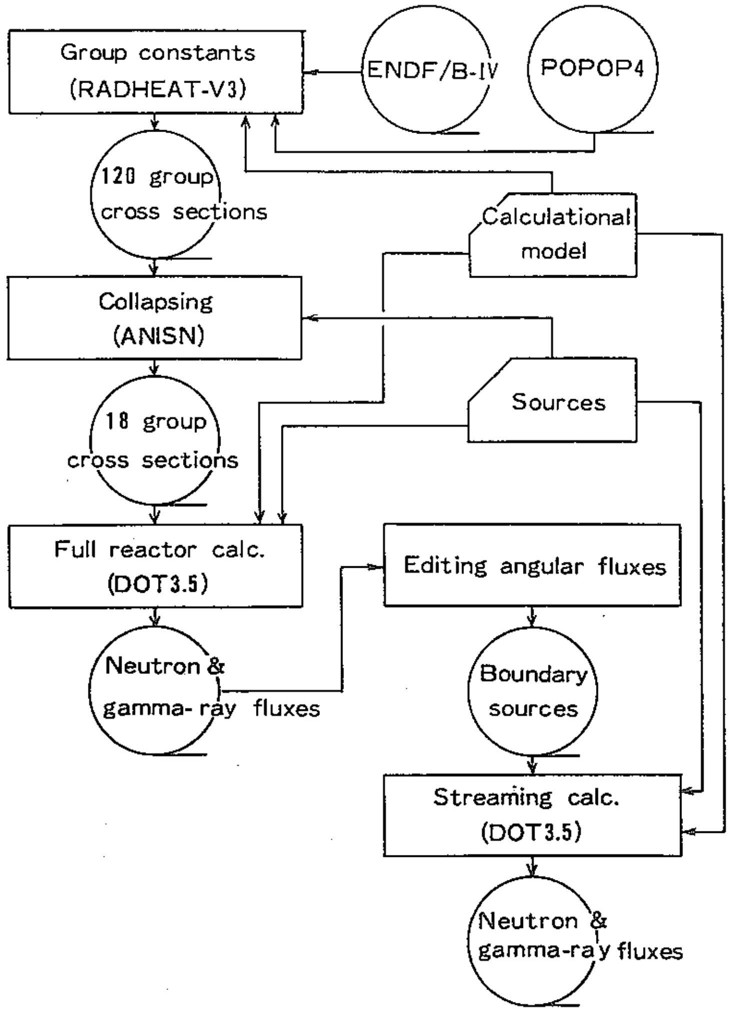

Figure 3. Schematic flow diagram for shielding analyses around the Fugen reactor

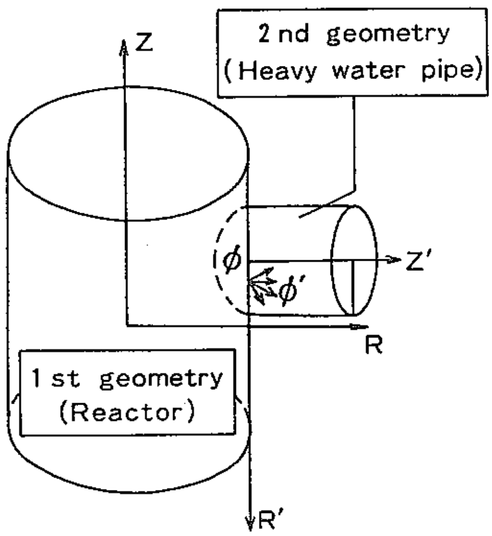

Figure 4. Perpendicularly connected two cylindrical geometries

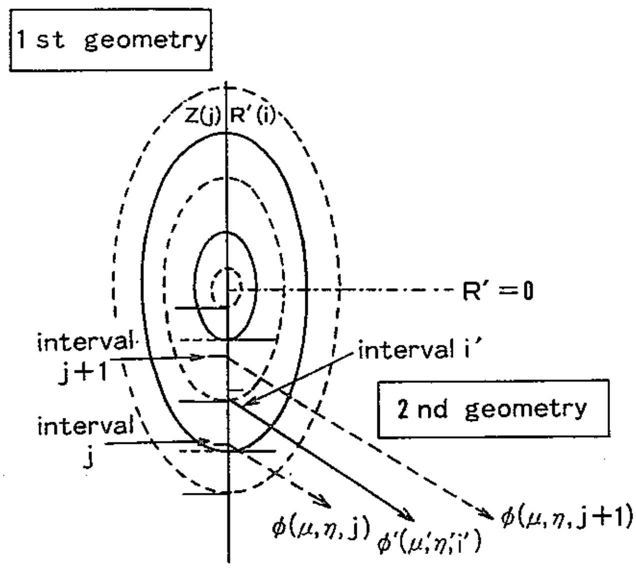

Figure 5. Transformation from angular fluxes to a boundary source

Sn quadrature: In this study, symmetric S30 quadrature was adopted to the full reactor calculation. On the other hand, some finer Sn quadrature sets were employed to the streaming calculations of the penetrating regions. According to the width and length of streaming path, S48 $\sim$ S164 quadrature sets were used for the calculational geometries mentioned below [5].

- symmetric S48 : Heavy water overflow pipe
- asymmetric S124: Calandria manhole, helium pipe
- symmetric S126 : Cavity region above upper shield
- asymmetric S164: Upper extension tube, narrow gap between upper shield and concrete shield

Among these quadrature sets, the symmetric sets were obtained from DOQ program and the asymmetric ones from ADOQ program.

MEASUREMENTS

The first measurement of gamma dose rates in the pressure tube was performed using an ion chamber detector (6.4 mm diameter and 31 mm long) mounted on the top of the borescope lifter after 247 days of shutdown. The whole fuel element was removed from the core, the pressure tube was emptied of fuel and drained of primary coolant. The dose rates obtained were compared with the calculated values [6]. The measured values in the pressure tube were in good agreement with the calculated ones by a factor of about 2. The dose rate in the middle of the pressure tube was slightly low because the aluminium baffle plate was located in this position. The observed dose rate below the lower radiation shield were slightly overestimated compared with the calculated ones. With the primary shields containing water and steel plates, the dose rates below the lower shield were reduced by a factor of approximately 50, while the evaluated one being about 85.

Gamma dose rates in the pressure tube without the removal of fuel from adjacent channels were measured for two times: one is surrounded by four fuels [6] and the other is surrounded by eight fuels [7]. Gamma dose rates were measured using chemical dosimeters and thermoluminescent dosimeters (TLDs) mounted on a pipe (5.89 m long and 13.8 mm outer diameter) at each 10 cm intervals. This pipe was inserted in the stainless steel vessel (6.98 m x 60.5 mm O.D.) and loaded in the pressure tube to be measured. Three chemical dosimeters (1.5 x 10 x 40 mm^3) were attached to each position. They are wrapped in aluminium foil to keep them clean during handling. The channel to be inspected was defuelled but filled with primary heat transport coolant (light water).

After the irradiation in the pressure tube for an appropriate period, the chemical dosimeters were taken off from the pipe. The change of absorptivity by irradiation was determined by a Hitachi Model 228 spectrophotometer. The TLDs were also used in the region where the evaluated gamma dose rate are not so high ($10^3 \sim 10^5$ mR/h) in the preceding calculation.

Irradiation time of chemical dosimeters was determined by preanalysis of gamma dose rates in the pressure tube and the measurable range of chemical dosimeter (about 0.5 Mrad to 6 Mrad). The results are shown in Table 1, compared with the evaluated values. The power distribution along the axial direction was also included in the calculation. Calculated results agreed within a factor of two with the experimental results in the pressure tube.

Also, radiation measurements around the primary shield of Fugen were performed at the outer surface of the radial shield, the heavy water overflow pipe and the helium pipe, and at the upper part of the upper shield (see Fig.1). In the radial regions, ^{197}Au (n,γ) reaction rates were measured by bare and cadmium-covered foils, ^{58}Ni (n,p) reaction rates by Ni foils, gamma-ray dose rates under reactor operation by thermoluminescent dosimeters (TLDs) and gamma-ray dose rates after shutdown by cobalt glass dosimeter, respectively [5]. In the upper regions, only ^{197}Au (n,γ) reaction rates were measured. The reason why the other three measurements were omitted is that the preliminary analyses showed that the sufficient fast neutron intensity for ^{58}Ni (n,p) reaction would not been obtained in these regions and the dominant gamma-ray sources would be ^{16}N under reactor operation and corrosion products such as ^{60}Co after reactor shutdown in the primary cooling system. The Au foils are cylindrical in shape with 0.1 mm thickness and 12 mm diameter. The thickness of cadmium cover is 0.5 mm. The Ni foils are also cylindrical in shape with 1 mm thickness and 12 mm diameter. The cobalt glass dosimeter was irradiated for 3900 effective full power hours (corresponding to $10^5 \sim 10^6$ R of accumulated dose). The dose rates (R/h) were evaluated by the measured accumulated dose and the irradiation time. Also, the effective irradiation time of TLD chips was 17 hours, and the dose rates were obtained by the accumulated dose and the irradiation time.

DISCUSSION

In the calculation of gamma dose rates in the pressure tubes, the DOT 3.5 code requires the homogenization of the core structures such as surrounding pressure tubes, calandria tubes and fuel elements. This induces a cylindrical "pseudo" radiation sources around the detector points in the pressure tubes, and thereby the cylindrical gamma ray source will cause a higher dose rate than that of actual arrangement of the core structures.

By use of the QAD code [8], the radiation dose rates in the center of the pressure tube are calculated with the lattice pitch geometry and with the homogenized (coaxial) geometry. The ratio of these two dose rates is defined with the following equation.

$$f_G = \frac{D_{QAD-HO}}{D_{QAD-LP}} \tag{1}$$

where

f_G : Ratio of homogenized geometry with lattice pitch one using QAD code

D_{QAD-HO}: Dose rate with the homogenized modelling using QAD code

D_{QAD-LP}: Dose rate with lattice pitch modelling using QAD code

In the case of gamma dose rates surrounded by four fuels as described above, the value of f_G was 2.06 and in the case of eight fuels, it was 1.64 [7]. Thus, the cylindrical gamma-ray source tends to cause higher dose rates than that of actual arrangement. If these f_G values are applied to the calculated values by the DOT 3.5 code, the calculated dose rates are in good agreement with the experimental results within a factor of 0.98 ∿ 1.18 (average 1.08) for four fuels (see Table 1), and 1.50 ∿ 1.63 (average 1.57) for eight fuels, respectively.

The comparison of the measured values around the primary shields with the calculated ones is shown in Fig.6. In almost all cases, the calculated results agreed with the measured data within a factor of 3. These data show relatively good agreement considering the neutron flux attenuation of

TABLE 1
Ratio of Dose Rates in the Pressure Tube

Axial Distance (cm)	Dose Rates (mR/h) Calculation C	Dose Rates (mR/h) Measurement E	Ratio C/E	Ratio C/2.06E
121	4.635 E+08	1.905 E+08	2.43	1.18
180	4.396 E+08	2.067 E+08	2.12	1.03
202	4.460 E+08	2.102 E+08	2.12	1.03
214	4.471 E+08	2.205 E+08	2.03	0.98
272	4.338 E+08	1.921 E+08	2.25	1.09
283	4.255 E+08	1.826 E+08	2.33	1.13
364	1.844 E+08	8.412 E+07	2.19	1.06
413	1.007 E+07	6.913 E+06	1.45	0.71
423	2.593 E+06	1.730 E+06	1.45	0.73
436	3.826 E+05	3.806 E+05	1.01	0.49
452	6.702 E+04	2.001 E+04	3.35	1.62
465	3.222 E+04	1.330 E+04	2.42	1.17
481	1.798 E+04	4.899 E+03	3.67	1.78
505	9.677 E+03	2.391 E+03	4.05	1.96
545	3.691 E+03	1.211 E+03	3.05	1.48
Average *			2.21±0.22	1.08±0.10

Effective Full Power Days: 1168 EFPD
Cooling Time: 23 days
Fuel in Adjacent Channels: 4 assemblies
*Average at the distance of 121 cm to 364 cm from the top of the core

more than 8 decades from the core to the measured points. As for the heavy water overflow pipe and helium pipe which penetrate the radial shield three dimensional geometries are constructed. Then the devised boundary sources previously described in this paper in detail were used. These C/E values show the validity of such boundary sources.

At the top of the core the main streaming path is a 2 ∿ 3 mm gap between each pressure tube and the upper shield (the path length is approximately 70 cm). In the calculation of streaming of both neutrons and gamma-rays through narrow gaps, the specially prepared quadrature sets such as S164 are useful. As for the C/E values around calandria manhole, the C/E values are distributed between 5.0 and 10.0, and the agreement between the calculated values and the measured data is not good in comparison with other data. The heavy water which lies on the overflow plate between the core and the calandria manhole was ignored in this analysis, but it is speculated that the influence of this heavy water is rather significant. This problem has not been resolved yet because it is difficult to determine the depth of the heavy water.

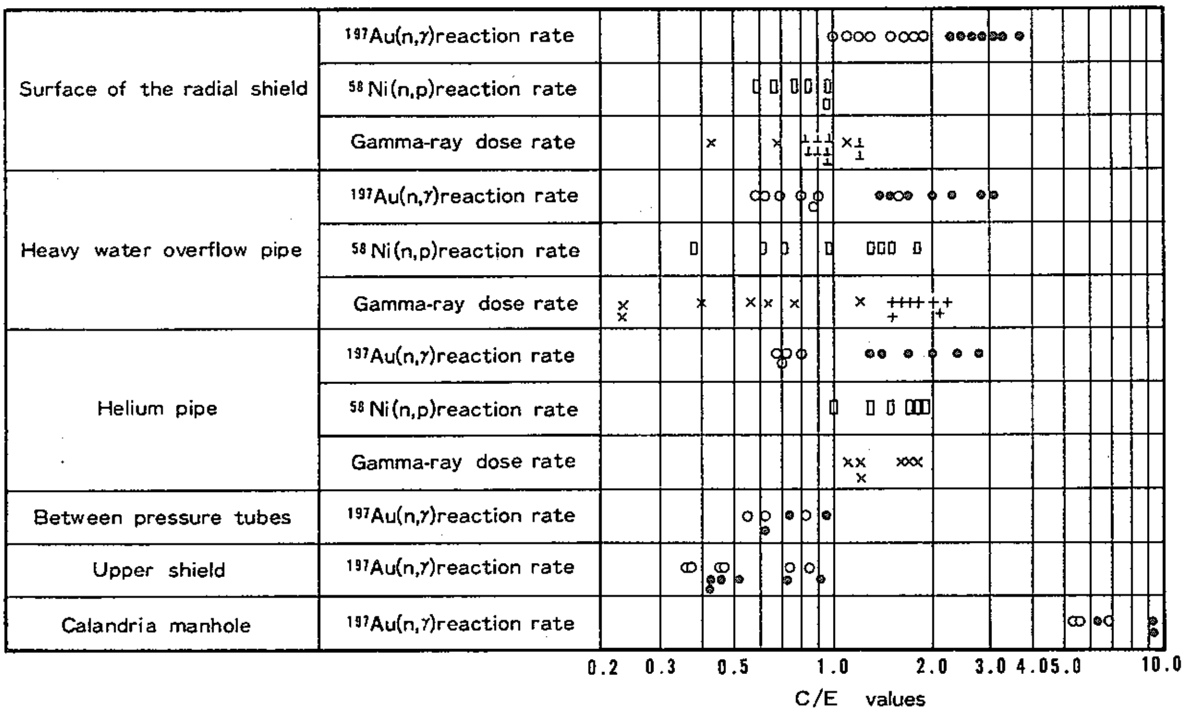

Figure 6. Ratio between calculated and experimental values obtained by using Au, Ni foils and TLDs (open symbols indicate the data obtained with bare foils and the closed ones are that of cadmium-covered foils)

CONCLUSIONS

For the shielding analysis around the reactor core of Fugen, the conclusions of this study are summarized as follows:

(1) The correction factors of the homogenized gamma-ray source strength for the DOT 3.5 was obtained with the three-dimensional QAD calculation in cylindrical geometry. The corrected dose rates with these factors were in good agreement with the measured values.
(2) Accurate results were obtained by using the DOT 3.5 code and the method used to evaluate the proper boundary sources for the bootstrapping calculations was confirmed.
(3) The asymmetric biased quadrature sets are useful for the streaming analyses of the narrow gap, especially for an extremely narrow gap such as 2 ∿ 3 mm width.
(4) The DOT 3.5 code is sufficiently useful for the shielding analyses of the pressure tube type reactors if the three-dimensional geometrical effects are properly represented by a two-dimensional approximation.

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