Journal of NUCLEAR SCIENCE and TECHNOLOGY, Supplement 1, p. 99–103 (March 2000)

Evaluation of the Shielding Design of the Fuel Handling and Storage System in the Prototype FBR Monju

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Shielding performance tests of the fuel handling and storage system were carried out at Monju in October 1993. The performance of the shielding of the fresh fuel storage rack against neutron and gamma radiation from fresh fuel assemblies, that of the in-vessel fuel transfer machine (IVTM) and the ex-vessel fuel transfer machine (EVTM) against neutron and secondary gamma-rays generated in the shielding or structural materials from the neutron source assembly (^{252}Cf), were assessed. Dose rates were measured using rem-counters and solid-state track-detectors (SSTDs) for neutrons and ionization chamber survey-meters and film dosimeters for gamma-rays. This paper presents an outline of the measurement methods, and results, the design requirements and the design margins, which were evaluated by comparing the measurements with the calculations based on FBR shielding analysis methods.

KEYWORDS : FBR, Monju, shielding measurements, dose rate measurements, shielding design, shielding analysis, fuel handling and storage system

I. Introduction

At the prototype FBR Monju, a series of tests was carried out to verify the characteristics and performance of the shielding. The objectives were to evaluate the design margins and demonstrate the validity of the FBR shielding analysis methods. The tests, which included measurements conducted around the reactor core, in the primary heat transport system cells, in the reactor head access area, and in the fuel handling system, were carried out from October 1993 to December 1995.

The shielding measurements for the fuel handling system were made by inserting a neutron source (^{252}Cf), with a known neutron intensity and neutron spectrum, into the in-vessel fuel transfer machine (IVTM) and then into the ex-vessel fuel transfer machine (EVTM). The neutron and gamma-ray dose equivalent rate distributions were also measured around one outer core fresh fuel assembly and around the fresh fuel storage rack which contained 24 fresh core fuel assemblies. These measurements were carried out in October 1993.

II. Shielding Design Requirements

The function of the IVTM is to transfer the assemblies between the in-reactor fuel handling machine and the EVTM. The EVTM carries the assemblies to and from the ex-vessel storage tank, the fresh fuel storage system, and the spent fuel handling facilities.

The shielding design requirements for all these fuel handling systems are based on the criteria for the fuel transportation cask, that is not exceeding 2000 μSv/h at the surface of the equipment and not exceeding 100 μSv/h at the 100cm from the surface. In the case of the upper guide tube of the IVTM, the requirement is: not exceeding 100 μSv/h at the nearest point for human access on the operating floor; this point is about 400cm from the center of the IVTM in the horizontal direction, as shown in **Fig. 1**.

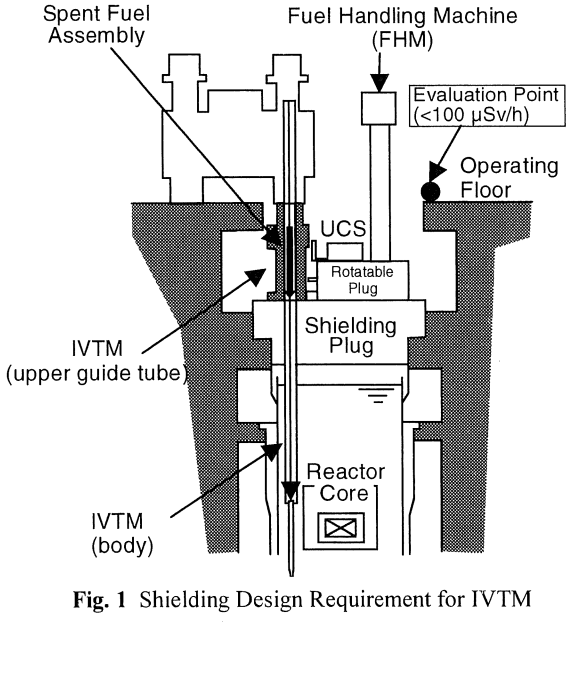

Fig. 1 Shielding Design Requirement for IVTM

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The shielding design requirement for the EVTM is: not exceeding 100 μSv/h at a point about 100cm from the surface of the electrical power supply equipment of the EVTM considering the nearest position for human access which is about 560cm from the center of the coffin of the EVTM in horizontal direction as shown in **Fig.2**.

The shielding design requirement for the fresh fuel storage rack is different from those of the above-mentioned fuel handling systems. It is based on the area separation principle for human access, that is: not exceeding 10 μSv/h at the entrance point of the rack which is the floor level of the fresh fuel handling room, as shown in **Fig.3**.

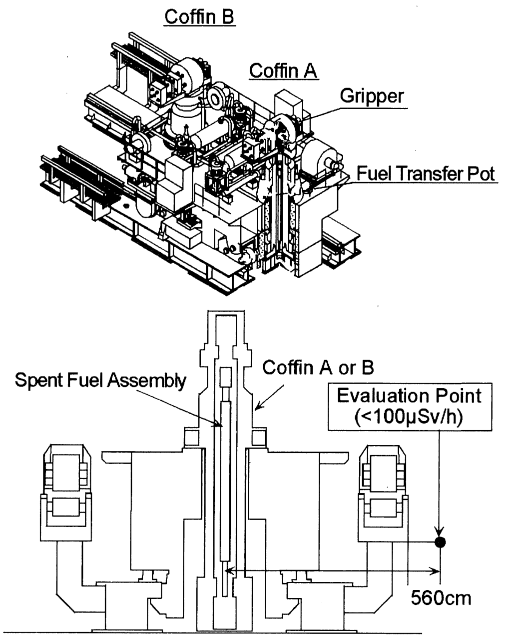

Fig. 2 Shielding Design Requirement for EVTM

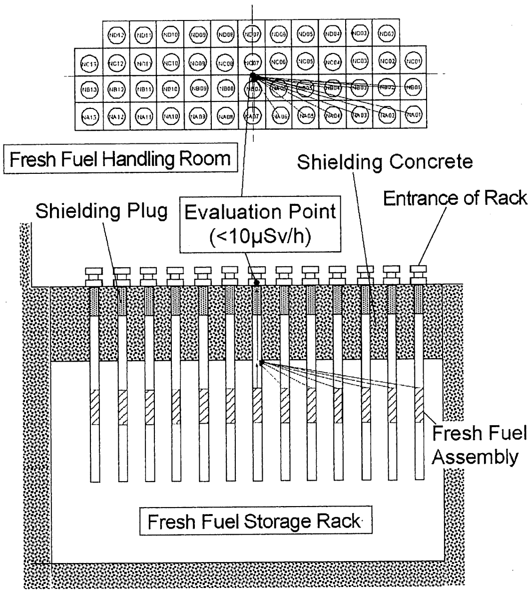

Fig. 3 Shielding Design Requirement for Fresh Fuel Storage Rack

The IVTM has a cylindrical geometry and is shielded with carbon steel, stainless steel, lead (Pb), and boron-containing polyethylene.

The coffin of the EVTM also has cylindrical geometry and, with the exception of lead, the same shielding materials.

The fresh fuel storage rack consists of 50 vertical stainless steel tubes each 470cm high, with inner/outer diameter: 15.1cm/16.52cm, respectively. They are located under the floor of fresh fuel handling room in a 40cm pitch, 13×4 arrangement. Each rack position has a stainless steel shielding plug, 10cm thick. The floor of the fresh fuel handling room is 150cm thick concrete.

III. Outline of Measurements

1. Dose Equivalent Rate Measurements around the IVTM and the EVTM

The shielding performance test for the IVTM and the EVTM was carried out using the neutron source assembly instead of one spent fuel assembly. The neutron source was ^{252}Cf, with a source strength of 6.5×10^9 n/s. The structure of the neutron source assembly is shown in **Fig. 4**.

Dose equivalent rates around the upper guide tube of the IVTM and Coffin-A of the EVTM were measured using rem-counters for neutrons and ionization chamber survey -meters for gamma-rays. The representative measurement points around the IVTM are shown in **Fig. 5** and the measured values are given in **Table 1**; those around the EVTM are shown in **Fig. 6** with measured values in **Table 2**.

The uncertainty of the measured value was about 4~8% for neutrons, and was about 10~20% for gamma-rays.

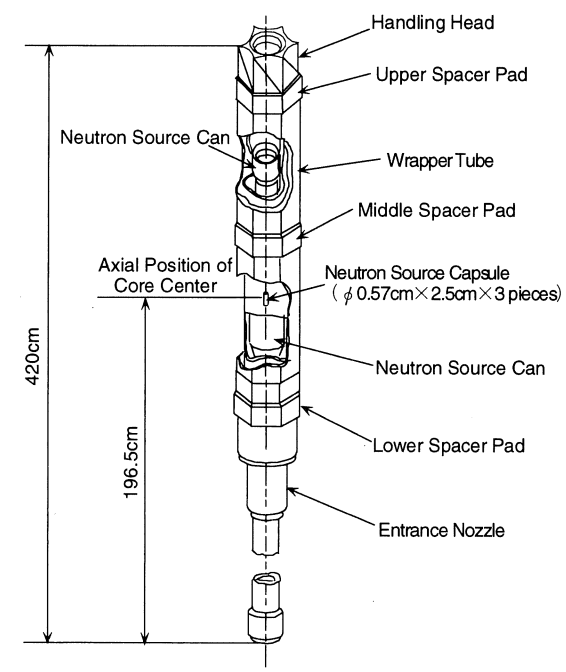

Fig. 4 Neutron Source Assembly

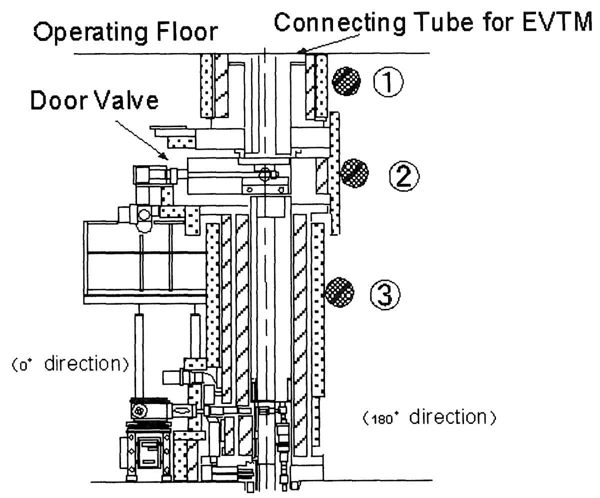

Fig. 5 Measurement Points around IVTM

Table 1 Neutron Dose Equivalent Rates around IVTM

Measurement Point	Neutron Dose Equivalent Rate (μSv/h)		C/E
	Calculated Value(C)	Measured Value(E)	
①	2.5	2.1	1.2
②	0.1	1.3	0.1
③	54	52	1.0

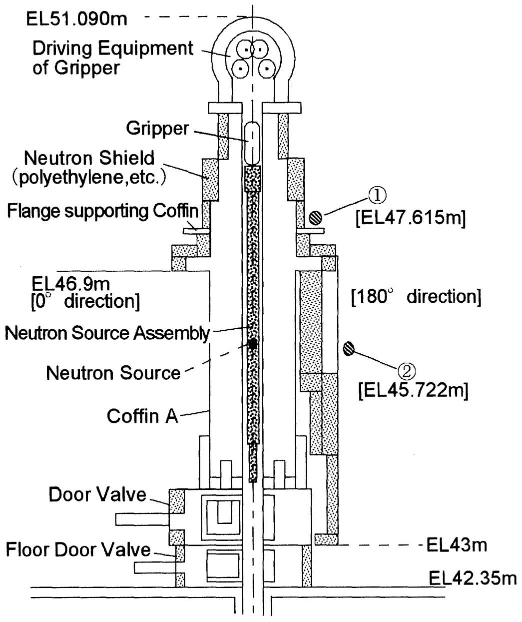

Fig. 6 Measurement Points around Coffin A of EVTM

2. Dose Equivalent Rate Measurements around one Fresh Fuel Assembly and the Fresh Fuel Storage Rack

The shielding performance test for the fresh fuel storage rack was carried out using 24 fresh fuel assemblies, of which the fuel material composition is known. In order to know the source strength precisely, a measurement of dose equivalent rate around one fresh fuel assembly was also carried out. A rem-counter and SSTD were used for neutron measurement, and an ionization chamber survey-meter and film dosimeters were used for gamma-rays.

Table 2 Dose Equivalent Rates around EVTM

Measurement Point	Radiation Ray	Dose Equivalent Rate (μSv/h)		C/E
		Calculated Value(C)	Measured Value(E)	
①	Neutron	2.5	7.0	0.4
	Gamma	4.7	3.3	1.4
②	Neutron	0.82	2.5	0.3
	Gamma	102	179	0.6

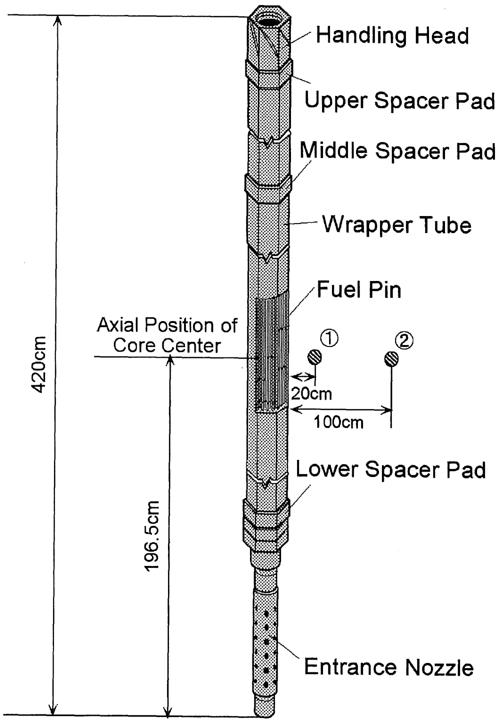

Fig. 7 Measurement Points around Fresh Fuel Assembly

The representative measurement points around one fresh fuel assembly are shown in **Fig.** 7 and measured values are given in **Table** 3. For the fresh fuel storage rack, this information is in **Fig.** 8 and **Table** 4.

The uncertainty of the measured value was about 1~2% for neutrons, and was about 10~20% for gamma-rays.

IV. Calculation Method

The dose equivalent rates around the IVTM and the EVTM were calculated using the neutron and gamma-ray fluxes and the neutron and gamma-energy dependent dose equivalent coefficients. The neutron and gamma-ray fluxes were calculated based on the FBR shielding analysis methods. Firstly, 100-group neutron effective cross-sections and 20-group gamma-ray effective cross-sections were made using the RADHEAT-V3[1] code system with the JSD-J2[2] and the JFT-J2[2] cross-

Table 3 Dose Equivalent Rates around one Fresh Fuel Assembly

Measurement Point	Radiation Ray	Dose Equivalent Rate (μSv/h)		C/E
		Calculated Value(C)	Measured Value(E)	
①	Neutron	480	400	1.2
	Gamma	750	420	1.8
②	Neutron	56	44	1.3
	Gamma	93	52	1.8

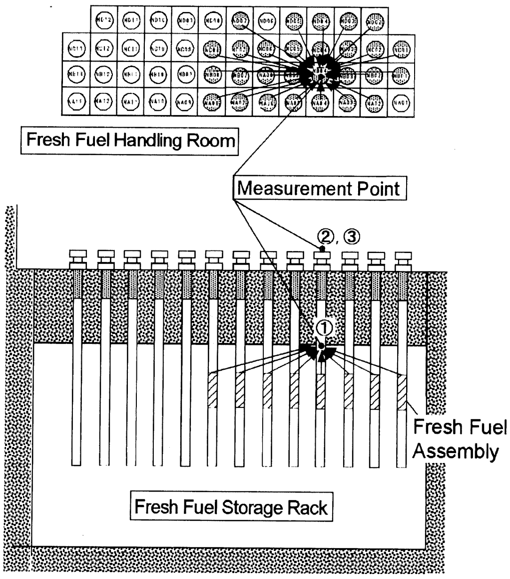

Fig. 8 Measurement Points around Fresh Fuel Storage Rack

Table 4 Neutron Dose Equivalent Rates around the Fresh Fuel Storage Rack

Measurement Point	Neutron Dose Equivalent Rate (μSv/h)		C/E
	Calculated Value(C)	Measured Value(E)	
①	1400	740	1.9
②	0.023	0.009	2.6
③	0.12	0.061	1.9

section libraries generated from the JENDL-2[3] file and the neutron induced secondary gamma-ray yield cross section library NEW-POPOP-4 generated from the POPOP-4[4] library and ENDF/B-4[5),(6)].

Then, the obtained 120-group cross-section set was reduced to 28-groups by means of the one-dimensional transport code ANISN[7], and used in the two-dimensional transport code DOT3.5[8].

For the IVTM and the EVTM, the known ^{252}Cf neutron energy spectra and source strength were used. Because the form is almost a point source, it was decided to adopt the first collision source method using the GRTUNCL code (including DOT3.5) to avoid the ray effect.

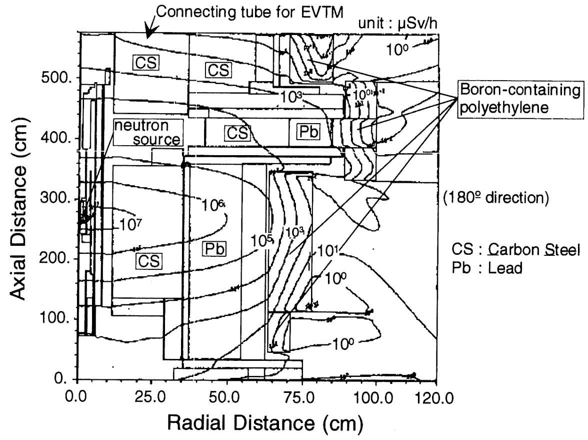

Fig. 9 Calculated Neutron Dose Equivalent Rate Distribution around IVTM

For the calculation of the neutron and gamma-ray fluxes around one fresh fuel assembly and the fresh fuel storage rack, the neutron and gamma source strength of 24 fresh fuel assemblies were calculated using the ORIGEN-2[9] code based on the measured fresh fuel composition of each assembly.

The neutron and gamma-ray fluxes around one fresh fuel assembly and the fresh fuel storage rack were calculated based on the FBR shielding analysis methods[10] using an RZ model.

V. Calculation Results

1. Dose Equivalent Rate Calculations around the IVTM and the EVTM

The calculated neutron dose equivalent rate distribution around the IVTM is shown in **Fig. 9**. The calculated values at the measurement points are given in **Table 1.** The calculated values for the neutron dose equivalent rate are in good agreement with the measured values at ① and ③, but at ②, the calculated value is much lower than measured value. The reason for this difference is thought to be that the neutron reflection effect by the structure around the IVTM (e.g. the upper core shielding), is not considered in the calculation model.

The calculated neutron dose equivalent rate distribution around the EVTM is shown in **Fig. 10**. The calculated values at the measurement points are given in **Table 2.** The calculated values are less than measured values except for the gamma-ray dose equivalent rate at ①. The reason for this underestimation is believed to be that neutron reflection by the structure around the EVTM (e.g. the floor shielding concrete) is not considered in the calculation model.

2. Dose Equivalent Rate Calculations around one Fresh Fuel Assembly and the Fresh Fuel Storage Rack

The calculated dose equivalent rates at the measurement points around one fresh fuel assembly are given in **Table 3.** The calculated neutron dose equivalent rates around one fresh fuel assembly are in good agreement with the measured values. The calculated values for gamma-ray dose rates are greater than the measured values by a factor of about 1.8.

The calculated neutron dose equivalent rate distribution

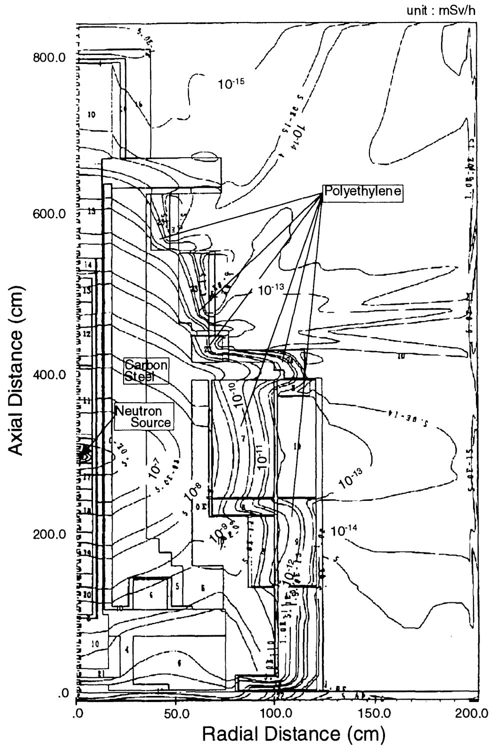

Fig. 10 Calculated Neutron Dose Equivalent Rate Distribution around EVTM

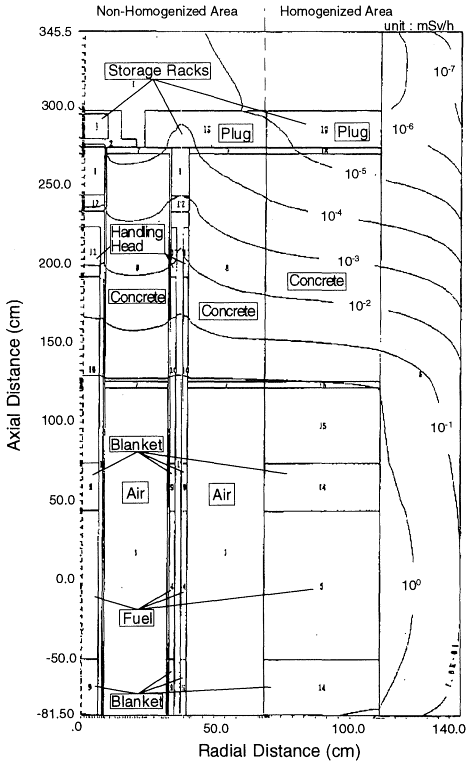

Fig. 11 Calculated Neutron Dose Equivalent Rate Distribution around Fresh Fuel Storage Rack

around the fresh fuel storage rack is shown in **Fig. 11.** The calculated values at the measurement points are given in **Table 4.**

The calculated neutron dose equivalent rates around the fresh fuel storage rack are larger than the measured values by factor of about 2.

VI. Discussion

The dose equivalent rate at the nearest point to the IVTM for human access on the operating floor (evaluation point in **Fig.1**) is estimated to be 2.7 μSv/h using the C/E value for neutrons at ③ in **Table 1,** and the design margin for the shielding structure is derived to be 37.

The dose equivalent rate at the nearest point to the EVTM for human access (evaluation point in **Fig.2**) is estimated to be 6.4 μSv/h using C/E value for neutron, and the design margin for the shielding structure is derived to be 16.

The dose equivalent rate at the entrance point of the fresh fuel storage rack (evaluation point in **Fig.3**) applying the C/E value, if all of the 50 rack position were filled with the fresh fuel assemblies, is estimated to be 0.012μSv/h. The design margin for the shielding structure is derived to be 800.

VII. Conclusion

The neutron and gamma-ray dose equivalent rates around the IVTM and the EVTM containing the neutron source assembly were measured. The neutron and gamma-ray dose equivalent rates around one fresh fuel assembly and the fresh fuel storage racks locating 24 fresh fuel assemblies were also measured. These measured values were compared with values calculated based on the FBR shielding analysis methods.

The differences between the measurements, the calculations and the design margins of these fuel handling and storage systems were assessed and basic data for shielding performance were obtained.

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