Journal of NUCLEAR SCIENCE and TECHNOLOGY, Supplement 1, p. 83–88 (March 2000)

Measurement of Shielding Characteristics in the Prototype FBR Monju

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In the prototype fast breeder reactor Monju, shielding measurements were made around the reactor core, the primary heat transport system (PHTS), and the fuel handling and storage system during the system start-up tests at different power levels between 0% and 45%. The objectives of the tests were to evaluate the margins by which the shielding performance exceeds the original design requirements, to demonstrate the validity of the shielding analysis method, and to acquire basic data for use in future FBR design. This paper summarizes the important features of the Monju shielding structures and the shielding measurements.

KEYWORDS : FBR, Monju, shielding measurements, dose rate measurements, shielding design, shielding analysis, primary heat transport system (PHTS), fuel handling and storage system

I. Introduction

Monju is a prototype fast breeder reactor designed to have an output of 280 MWe (714 MWt), fueled with mixed oxides of plutonium and uranium and cooled by liquid sodium. The principal data on plant design and performance are shown in **Table 1**. The shielding design of Monju was carried out in 1980~1981, using methods developed following the radiation leakage experienced on the nuclear ship Mutsu in August 1974[1, 2].

As part of the system start-up tests on the Monju initial core, the minimum critical core tests and the other core physics tests were carried out from October 1993 through November 1994. The power raising tests (from 2% to 45% of rated power) were carried out from February through December 1995. The test schedule is shown in **Fig. 1**.

During these test periods, the shielding characteristics tests were conducted in order to evaluate the design margins of the shielding performance, to demonstrate the validity of the shielding analysis method, and to acquire basic data for use in future FBR design. Shielding measurements were made around the reactor core, the primary heat transport system (PHTS), and the fuel handling & storage system.

This paper describes the important features of the shielding structures and the shielding measurements.

Table 1 Principal Design and Performance Data of Monju

Reactor type :	Sodium-cooled loop-type
Thermal / Electrical output :	714 / 280 MW
Fuel material :	PuO_2-UO_2
Core dimensions : Equivalent diameter / Height	1,790 / 930 mm
Plutonium enrichment (inner core / outer core) :	
(Pu fissile %) Initial core : 15 / 20, Equilibrium core : 16 / 21	
Fuel inventory : Core (U+Pu metal) 5.9 t, Blanket (U metal) 17.5 t	
Average burnup at discharge :	80,000 MWd/t
Cladding material :	SUS316
Cladding outer diameter / thickness :	6.5 / 0.47 mm
Blanket thickness : upper / lower / radial	30 / 35 / 30 cm
Breeding ratio :	1.2
Primary sodium temperature : Reactor inlet / outlet	397 / 529 °C
Secondary sodium temperature : IHX inlet / outlet	325 / 505 °C
Reactor vessel : height / diameter	18 / 7 m
Number of loops :	3
Type of steam generator :	Helical coil
Steam pressure / temperature (turbine inlet) :	12.7MPa /483°C
Refueling system : Single rotating plug with fixed arm FHM	
Refueling interval :	6 months

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II. Shielding Structures in Monju

1. Shielding Structures around the Reactor Core

The reactor core consists of 198 core fuel assemblies surrounded by 172 radial blanket fuel assemblies. All fuel assemblies are of identical overall size and have hexagonal steel wrappers. A core fuel assembly contains 169 pins filled with pellets of mixed oxide (MOX) and a blanket fuel assembly contains 61 pins filled with pellets of depleted uranium oxide. The upper and lower parts of the core fuel pins also contain depleted uranium, forming axial blankets to the core.

The design requirements for the shielding around the reactor core were : (1) to prevent radiation damage to the in-vessel structures and the reactor vessel by limitation of the cumulative irradiation, (2) to limit the dose equivalent rate to not exceeding 500 μSv/h (60 μSv/h in specified areas) in the reactor head

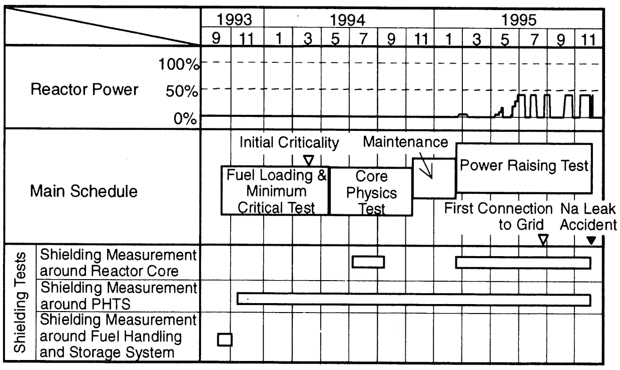


Fig. 1 Realized Test Schedule

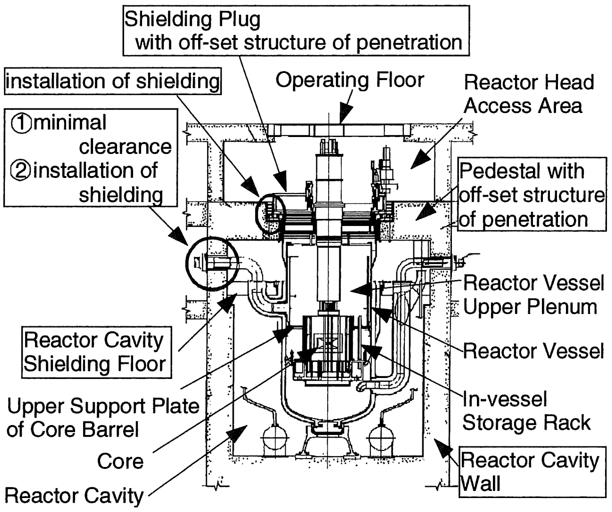


Fig. 2 Shielding Design around Reactor Core

access area, (3) to prevent the activation of the secondary sodium, and (4) to minimize the background level of the delayed neutron method failed fuel detection system (DN/FFD).

The shielding structure around the reactor core is shown in **Fig. 2**. The main structures are the reactor cavity wall in the radial direction and the shielding plug and pedestal in the upper axial direction. The reactor cavity wall, consisting of a concrete-filled steel structure, is about 2 m thick and of hexagonal geometry. The shielding plug is constructed of multiple layers of stainless steel, carbon steel and polyethylene to provide neutron and gamma-ray shielding and also thermal shielding. The pedestal is constructed of a serpentine concrete-filled steel structure, and forms part of the head access area around the shielding plug.

The countermeasures against radiation streaming include the following : (1) the radial and axial stainless steel neutron shielding around the reactor core, (2) the reactor cavity shielding floor, comprised of a 1 m thick serpentine concrete-filled steel structure, (3) the off-set structure of the penetration in the shielding plug and the pedestal, and (4) minimal clearance and inclusion of shielding in the penetrations.

2. Shielding Structures around the PHTS

The shielding structure around the primary heat transport system (PHTS) is shown in **Fig. 3**. The PHTS is composed of 3 loops ; the primary sodium coming from the reactor vessel, passes through the intermediate heat exchanger (IHX) and the primary circulation pump, and goes back to the reactor vessel along the route shown.

The design requirements for the shielding around the PHTS were : (1) the total neutron flux on the surface of the IHX should not exceed 1×10^4 n/cm^2·s in order to prevent the activation of the secondary sodium circulating in the IHX, and (2) in order to increase the sensitivity of the DN/FFD, the neutron background level should be minimized. The chosen limit was that the total neutron flux on the biological shield wall should not exceed 1×10^4 n/cm^2·s at the outlet of the second penetration and not exceed 1×10^3 n/cm^2·s elsewhere.

The countermeasures against radiation streaming at the PHTS

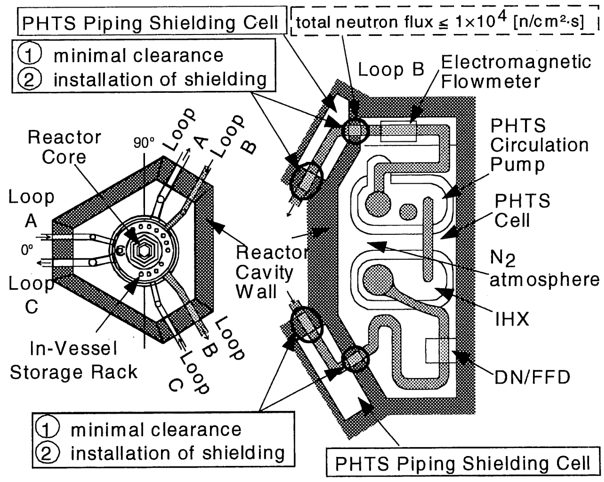


Fig. 3 Shielding Design around PHTS (Horizontal Cross Section)

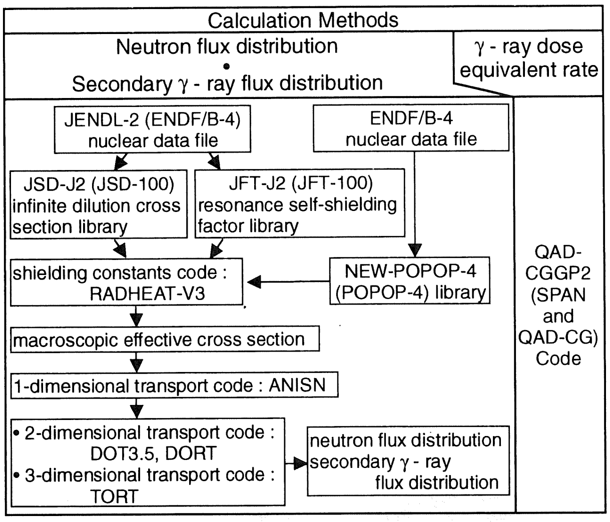
piping penetration include (1) the design of the PHTS piping shielding cells between the reactor cavity and the PHTS cell and (2) minimal clearance and inclusion of shielding, such as B_4C collars, in the piping penetrations.

III. FBR Shielding Analysis System

The shielding analysis system of Monju design calculations performed around 1980 is indicated in parenthesis in **Fig. 4**. In order to analyze the neutron flux distribution and the secondary gamma-ray distribution, the infinite dilution cross section library JSD-100[3] and the resonance self-shielding factor library JFT-100[3] were made from the nuclear data file ENDF/B-4[4, 5]. POPOP-4 library[6] was used as the neutron induced secondary gamma-ray yield cross section library. The macroscopic effective cross sections (100 neutron and 20 gamma-ray groups) were calculated by using the shielding constants code RADHEAT-V3[7]. These macroscopic effective cross sections were collapsed to 21 neutron and 7 gamma-ray groups by the 1-dimensional transport code ANISN[8]. The neutron flux distribution and the secondary gamma-ray flux distribution were obtained by the 2-dimensional transport code DOT3.5[9]. For the calculation of the gamma-ray dose equivalent rates, the SPAN[10] and the QAD-CG code[11] were used.

The effectiveness of this method had already been confirmed through the system start-up tests in Joyo and it has been updated through the experiments in JASPER (the Japanese-American Shielding Program for Experimental Research).

The updated FBR shielding analysis system is shown in Fig. 4. The infinite dilution cross section library JSD-J2[12] and the resonance self-shielding factor library JFT-J2[12] were made from the nuclear data file JENDL-2[13]. The neutron induced secondary gamma-ray yield cross section library NEW-POPOP-4 was made from the POPOP-4 library and ENDF/B-4. The macroscopic effective cross sections (100 neutron and 20 gamma-ray groups) were calculated by using the updated shielding constants code RADHEAT-V3. These macroscopic effective cross sections were collapsed to 21 neutron and 7



*: parenthesis was used for design calculations

Fig. 4 FBR Shielding Analysis System

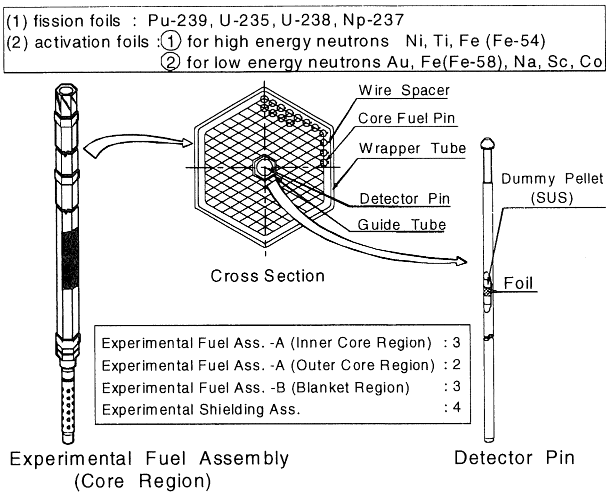


Fig. 5 Experimental Fuel Assembly

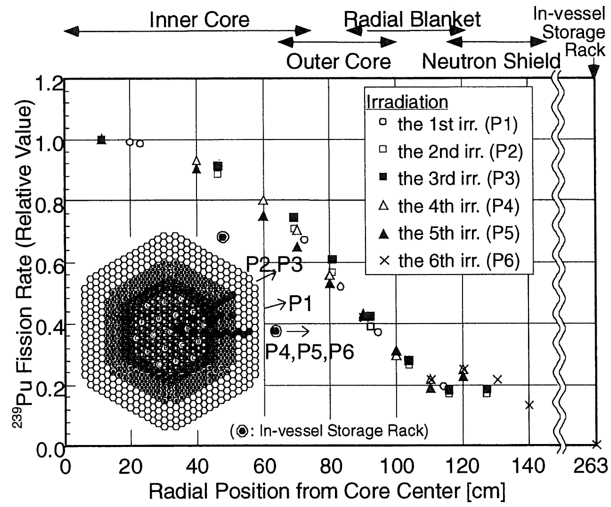


Fig. 6 Radial Distribution of Measured ^{239}Pu Fission Rate on Core Central Plane

gamma-ray groups by the 1-dimensional transport code ANISN. The neutron flux distribution and the secondary gamma-ray flux distribution were obtained by the 2-dimensional transport code DOT3.5 and DORT[14], and by the 3-dimensional transport code TORT[15]. For the calculation of the gamma-ray dose equivalent rates, the 3-dimensional point attenuation kernel integral code QAD-CGGP2[16] reflecting the recommendation from ICRP51 publication was used.

IV. Shielding Measurements around the Reactor Core

1. In-vessel Neutron Reaction Rate Measurements from the Core to the In-vessel Storage Rack[17, 18]

The in-vessel reaction rate distributions were measured by the foil activation method using several types of fission foils and activation foils, which have reactions in different neutron energy ranges. The foils were located at representative positions in the core region, the blanket region, the neutron shield region and the in-vessel storage racks by the use of experimental assemblies. The structure of the experimental fuel assembly is shown in **Fig. 5**. The experimental assemblies, which have the same basic structure as normal assemblies, were loaded in place of standard core elements. The foils were contained in a detector pin located at the center of the pin bundle. In the detector pin, stainless steel dummy pellets were used, and the foils were located at selected axial positions between the dummy pellets. Different foils were irradiated by exchanging the detector pin in the center of the experimental assembly. The irradiation of the experimental assemblies was carried out in six campaigns at reactor powers ranging from 0.02~0.16%. The assemblies were irradiated in different locations and with different control rod configurations, the temperature was 200°C throughout the tests. The gamma-ray spectra of the irradiated foils were measured by use of high purity Ge solid state detectors, and by considering the history of the irradiation and the conditions of the measurement, several types of reaction rate data were obtained. As an example of the measured reaction rate distributions, the relative radial distribution of the measured ^{239}Pu fission rate on the central plane is shown in **Fig. 6**. The uncertainty in the relative ^{239}Pu fission rate distribution from the core to the in-vessel storage rack was in the range of 0.7~2.2 %.

2. Neutron Reaction Rate Measurements in the Reactor Vessel Upper Plenum

The measurements were conducted using the stainless steel guide tube of the core loading monitor (CLM) as shown in **Fig. 7**, which is located about 270 cm from the core center axis in the radial direction and passes from the core central plane through the shielding plug. The ^{10}B proportional counter and the activation foils such as Au, Fe, Co were put into this guide tube, and the neutron reaction rates in the approximately 800 cm range from the core central plane to the sodium surface were measured. At the time of the measurement, the sodium temperature was about 200°C, and the reactor power was about 0.03% (when using the ^{10}B proportional counter) and 0.16% (when using the activation foils). A cadmium cover

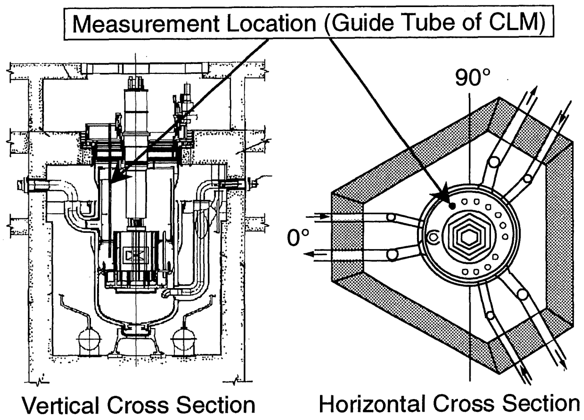


Fig. 7 Measurement Location in Reactor Vessel Upper Plenum

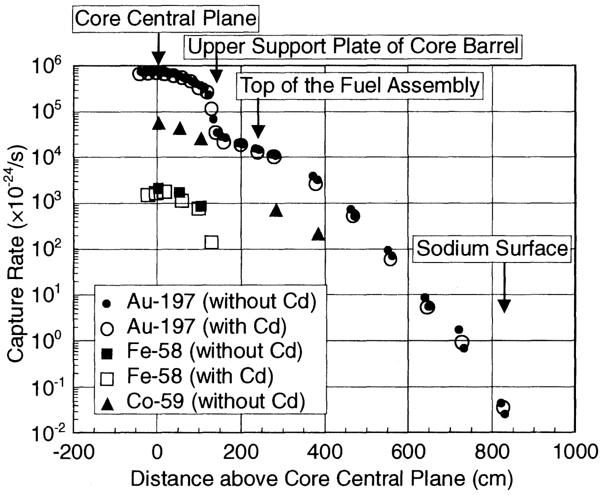


Fig. 8 Vertical Distribution of Measured Reaction Rate

for the ^{10}B proportional counter and foils was also used to evaluate the contribution of thermal neutrons to the reaction rates. **Fig. 8** shows the measurement results of the capture rates of ^{197}Au, ^{58}Fe and ^{59}Co. The neutron flux attenuates rapidly at about 120 cm above the core central plane due to the upper support plate of the core barrel. In all other regions, the neutron flux attenuates uniformly. The uncertainly in the measured ^{197}Au (n,γ) reaction rate was in the range of 7 ~ 40%. The contribution of thermal neutrons to the ^{197}Au (n,γ) and ^{58}Fe (n,γ) reaction rates are low.

3. Neutron and Gamma-ray Dose Equivalent Rate Measurements in the Reactor Head Access Area

In order to evaluate the validity of the shielding design of the upper part of the reactor, the neutron and gamma-ray dose equivalent rates were measured in the reactor head access area (at positions No.1 to No.8 in **Fig. 9**) at the 39% reactor power level. Rem-counters and NaI scintillation counters were used for the neutron and the gamma-ray dose equivalent rate measurement, respectively. The measured dose equivalent rates were below the detection threshold of the detectors, i.e., 1μSv/h for the rem-counter and 0.1μSv/h for the NaI scintillation counter.

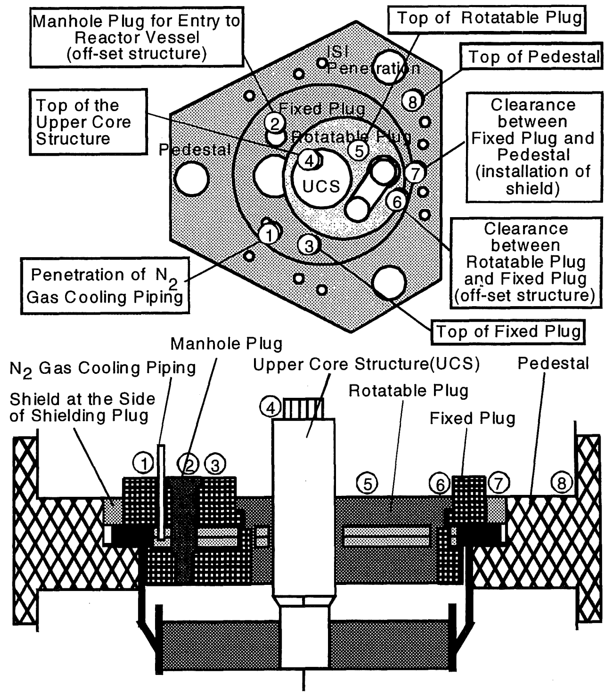


Fig. 9 Measurement Positions in Reactor Head Access Area

V. Shielding Measurements around the PHTS

1. Activation Measurements of the Primary Sodium

^{24}Na and ^{22}Na, created by the activation of primary sodium, are very important nuclides to consider in the design of the wall thickness around the primary heat transport system (PHTS) and in the evaluation of the exposure rates at maintenance activities. So, in order to evaluate the accuracy of the shielding analysis method and the design margins, the activation of primary sodium was measured during the power raising test. Sodium was sampled from the primary sodium purification system line and the activity of the sampled sodium was measured by the Ge semiconductor detector.

Because the half-life of ^{24}Na (15 hr) is relatively short, the measured activities of ^{24}Na were calibrated with the instantaneous complete mixing model[(19)] composed of the overflow tank and the cold trap. For ^{22}Na (half-life : 2.6 years), the measured value obtained by sampling was used. Taking into account the power history during the power raising test and extrapolating the measured values on the basis of the typical operation cycle, the activation of primary sodium in the radioactive equilibrium condition at the rated power was obtained. The results were 2.9×10^8 Bq/cm^3 for ^{24}Na and 3.3×10^4 Bq/cm^3 for ^{22}Na.

2. Gamma-ray Dose Rate Measurements in the PHTS Cell

Because of the activation of primary sodium, the PHTS cell and the PHTS piping shielding cell are exposed to high gamma-ray dose rates. Gamma-ray shielding is provided by the thickness of the shield wall, and the gamma-ray dose rate measurements in the PHTS cell were carried out in order to verify the shielding performance of the PHTS cell and to obtain the basic data about the shielding design. Semiconductor

Core Physics Test		Power Raising Test
γ-ray dose equivalent rates	γ-ray dose rates	
Si semiconductor detector	TLD	TLD-800, Cobalt glass dosimeter, Alanine dosimeter, Film dosimeter
3 points in Loop A (▲)	77 points	20 points in Loop A (●), 9 points in Loop B

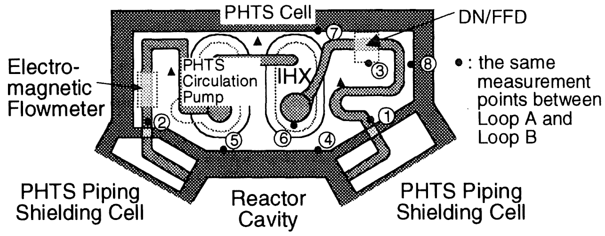


Fig. 10 Measurement Points in Primary Heat Transport System (PHTS) Cell

Table 2 Results of Gamma-ray Dose Rate Measurements in the PHTS Cell

(unit : R/h)

Measurment point		Measured Value *1
Hot Leg Piping	①	5.4×10^3
Cold Leg Piping	②	4.0×10^3
Hot Leg Piping	③	2.2×10^3
Wall of the PHTS Cell	④	1.5×10^3
Wall of the PHTS Cell	⑤	1.1×10^3
IHX	⑥	2.8×10^3
Wall of the PHTS Cell	⑦	1.7×10^3
Wall of the PHTS Cell	⑧	1.7×10^3

*1: the average value between Loop A and Loop B, which is extrapolated to the power condition

detectors and TLDs were used for gamma-ray detection in the core physics test ; TLDs, cobalt glass dosimeters, alanine dosimeters and film dosimeters were used and exposed during about 40 effective full power days (EFPD) in the power raising test. As an example, **Fig. 10** shows the locations of the measurements with gamma-ray detectors. **Table 2** shows the results of gamma-ray dose rate measurements in the PHTS cell, which were extrapolated to the rated power condition.

3. Neutron Dose Rate Measurements in the PHTS Cell

Foils of Co, Au and Ni were placed in the PHTS piping shielding cell and the PHTS cell in order to measure the neutron flux. They were exposed during about 40 EFPD. Solid state track detectors (SSTDs) were also fitted to the wall of the PHTS cell to measure the neutron dose equivalent (with energy over 50 keV). In addition, the count rates of the DN/FFD were observed. When the measured values were extrapolated to the rated power, the results were that the extrapolated measured ^{59}Co (n,γ) reaction rate around the first penetration of the PHTS piping shielding cell was 4.6×10^{-22} s^{-1}. The extrapolated measured neutron dose equivalent rates (with energy over 50 keV) in the PHTS cell were 0.4~2.0 μSv/h.

There are three types of neutron source in the PHTS cell : the first is the neutrons leaking from the reactor core, and the second is the photo-neutrons generated in the concrete, especially perlite concrete. The third is the delayed neutrons coming from the fission products of the uranium impurities in the primary sodium (mainly from surface contamination of

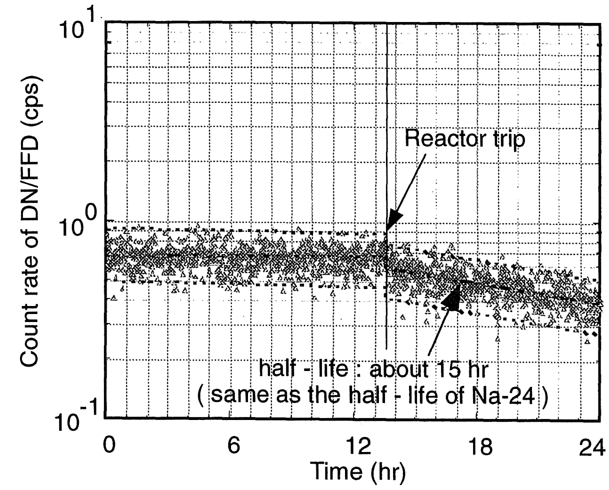


Fig. 11 The Change in Count Rate of Delayed Neutron Method Failed Fuel Detection System (DN/FED) at ReactorTrip

the core fuel assemblies). A photo-neutron is created by the (γ, n) reaction of heavy hydrogen in the concrete with the high energy (2.75 MeV) gamma-ray coming from ^{24}Na in the primary sodium. **Fig. 11** shows the change in measured count rate of the DN/FFD just after the reactor trip from 45% reactor power level. According to Fig.11, the half-life of the count rate of this detection system was almost the same as that of ^{24}Na, which causes the creation of photo-neutrons. Thus, it was clarified that the majority of neutrons in the PHTS cell are photo-neutrons.

VI. Shielding Measurements around the Fuel Handling and Storage System

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

Figure 12 outlines the measurements around the in-vessel fuel transfer machine (IVTM) and the ex-vessel fuel transfer machine (EVTM). The neutron source was ^{252}Cf, the intensity (6.5×10^9 n/s) and spectrum of which were known. Rem-counters were chosen for neutron detection and ionization chamber survey-meters for gamma-ray detection.

The IVTM is used to transfer the assemblies between the fuel handling machine (FHM) 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 design requirements for the shielding of the EVTM and the upper guide tube of the IVTM are based on the criteria of the fuel transportation cask, that is not exceeding 2000 μSv/h at the surface of the equipment and not exceeding 100 μSv/h at 100 cm from the surface. However, because the minimum distance for human approach is 400 cm from the center of the IVTM, the design requirement of the IVTM is that the dose equivalent rate should not exceed 100 μSv/h at the evaluation point A in Fig. 12. As a result of the measurements, when a spent fuel assembly is used instead of the neutron source, the estimated dose equivalent rates at the evaluation point A and point B (at 100 cm from the surface of the EVTM) are 2.7 μSv/h and 6.4 μSv/h, respectively.

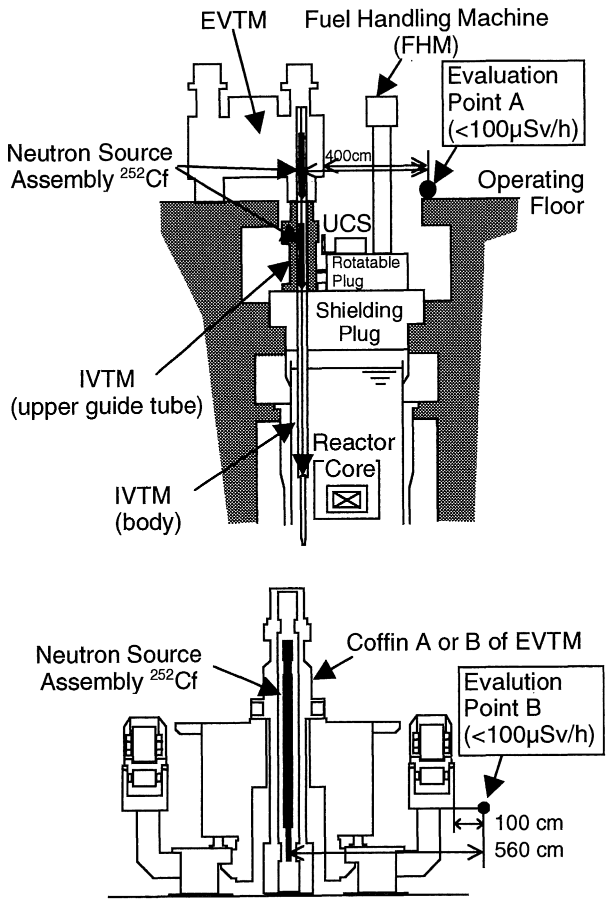


Fig. 12 Shielding Measurements around Fuel Handling and Storage System

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

A rem-counter and SSTDs were used for neutron detection and an ionization chamber survey-meter and film dosimeters for gamma-ray detection. The dose equivalent rates around one outer core fresh fuel assembly were 400 μSv/h for neutrons and 420 μSv/h for gamma-rays at 20 cm from the surface of the assembly. Values of 44 μSv/h for neutrons and 52 μSv/h for gamma-rays were obtained at 100 cm from the surface. The neutron and gamma-ray dose equivalent rates at the entrance of the fresh fuel storage rack, where the 24 outer core fresh fuel assemblies were stored, were 0.009 μSv/h and less than detection threshold, respectively.

VII. Conclusion

The measurements performed during the start-up tests demonstrated that the design of the Monju shielding fully meets the requirements for safe operation of the reactor. The results validate the shielding analysis methods applied and will be of value in design of future FBR plants.

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