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ABSTRACT

Neutron and gamma ray measurements were made within the reactor cavity of the Fast Flux Test Facility (FFTF) to establish the operating characteristics of the Ex-Vessel Flux Monitoring (EVFM) system as a function of reactor power level. A significant effort was made to obtain absolute flux values in order that the measurements could be compared directly with shield design calculations. Good agreement was achieved for neutrons and for both the prompt and delayed components of the gamma ray field.

INTRODUCTION

The Fast Flux Test Facility (FFTF), located at the Hanford Engineering Development Laboratory (HEDL) near Richland, Washington, is to serve as the primary test facility in the United States for the development of commercial fast breeder reactors. The Fast Test Reactor (FTR) in this facility is designed to operate at 400 MW and to irradiate material and test fuels in hot flowing sodium at temperatures up to 1,400°F and at neutron fluxes up to 10^{16}n/cm^{2}sec.[1] The FTR, illustrated in Figure 1, contains 91 core positions for fuel, control, and test assemblies with a hexagonal lattice spacing of 4.715 inches. The core is surrounded by 108 reflector positions with the same hexagonal lattice pitch. Radial shielding is provided by stainless steel plates that create a twelve-sided configuration within the structural core barrel. The core, reflector and radial shield are supported by the coolant inlet distribution structure, which is attached to the reactor vessel. The reactor vessel is approximately 20 feet in diameter and 43 feet tall. The annular region between the core barrel and the reactor vessel is filled with sodium and contains transfer and storage positions for up to 60 core or reflector assemblies. The physical arrangement is illustrated in Figure 2. Construction of the facility was begun in 1971 and completed in the fall of 1978. Introduction of sodium into the system began in July 1978 and fuel loading began in late 1979. The reactor achieved initial criticality on February 9, 1980. After a few low power nuclear tests, including this one to characterize the nuclear environment within the reactor cavity, operation of the reactor was terminated to permit replacement of a pump,

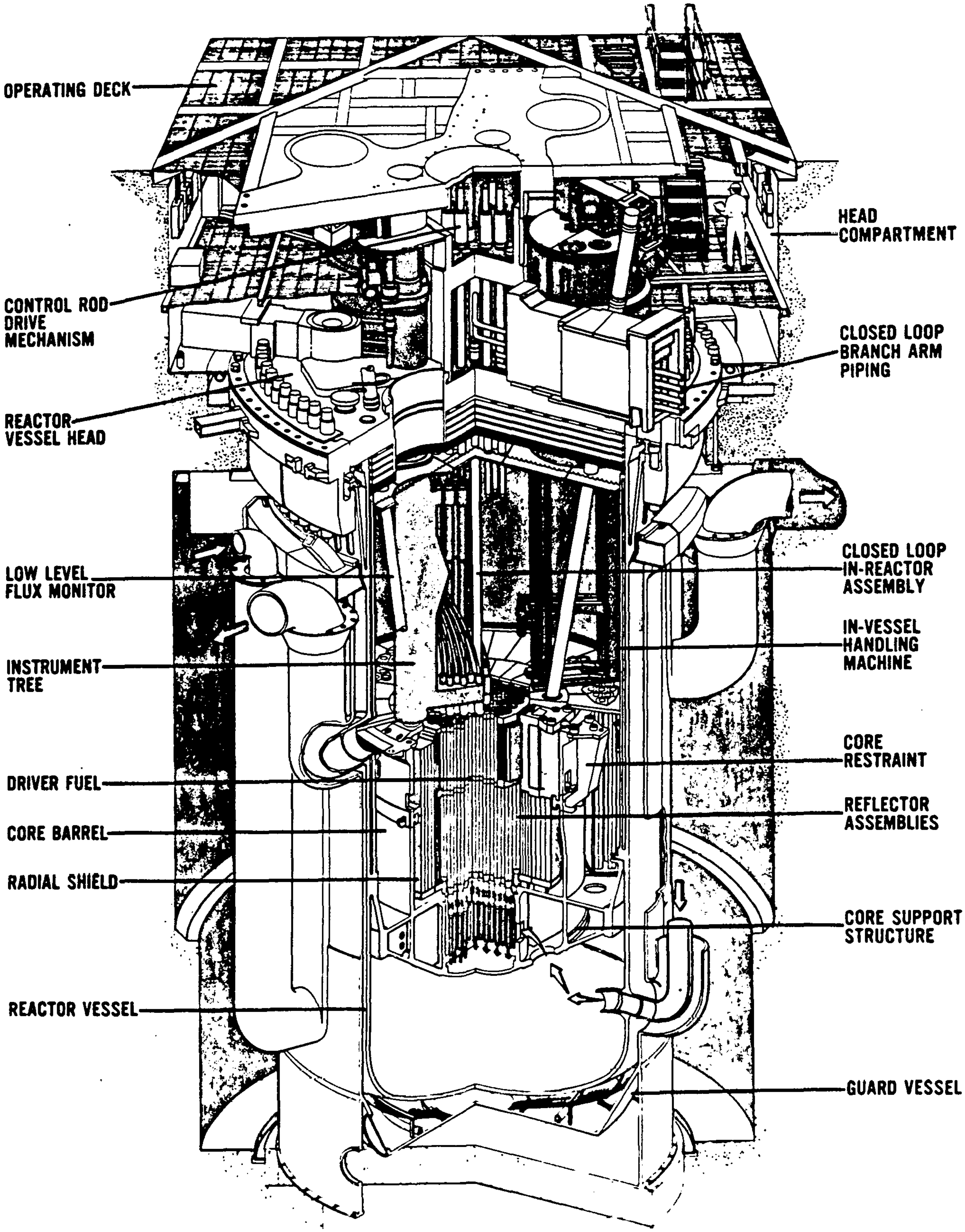

Figure 1. Cutaway view of Fast Test Reactor.

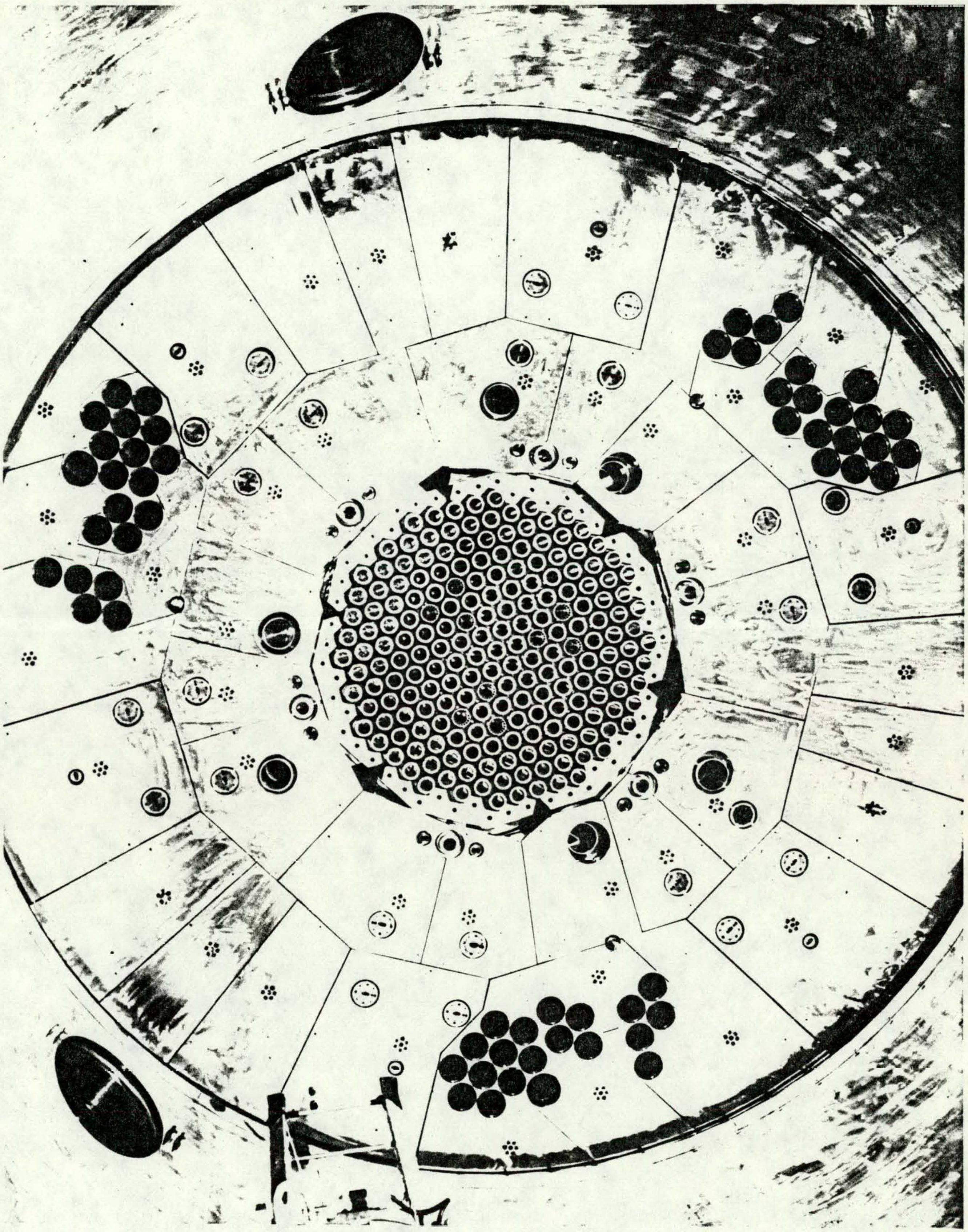

Figure 2. Top view of Fast Test Reactor showing core and reflector positions, radial shield region, and location of three storage modules.

installation of additional piping insulation, and completion of additional plant features to support power operation.

During the initial approach to critical, the Ex-Vessel Flux Monitor (EVFM) system did not respond as anticipated. To provide the technical information to define any required system modifications to meet design and operating objectives, a series of neutron and gamma measurements was made within one of the thimbles that normally houses an EVFM detector. In addition to providing data for the EVFM system, the measurements also establish the attenuation characteristics of the reactor shield. It is the latter result that will be described in this paper.

EXPERIMENTAL METHOD

The EVFM detectors are normally positioned within a block of graphite (19x32x48 inches) located in the reactor cavity as illustrated in Figure 3. The midplane elevation of the graphite block is at the same elevation as the core midplane. There are three graphite blocks arranged nearly symmetrically in the reactor cavity to provide three redundant EVFM systems. Each EVFM system is in turn comprised of a fission counter and a compensated ionization chamber to provide diversity. During this low power nuclear test, the compensated ion chamber was removed from its thimble in Channel A (see Figure 3) enabling us to make axial traverses with a number of different detector arrangements. Because the thimble sealed the inert (nitrogen) reactor cavity atmosphere from the operating deck, it was possible to make a number of different measurements in a short period of time; however, the thimble restricted the measurements to the one azimuthal and radial position.

A. NEUTRON MEASUREMENTS

Neutron measurements employed both a fission counter and a boron coated compensated ion chamber that were procured for the EVFM system. Special electronics were set up on the operating deck adjacent to the thimble to coordinate detector movement and recording of data. To provide the detailed information required for characterizing the neutron environment adequately for modifying the EVFM system and evaluating the shield performance, measurements were made at a number of elevations with a number of different materials surrounding the sensors. Results are summarized in Table I.

B. GAMMA MEASUREMENTS

The gamma field in the EVFM location was measured using a sodium iodide scintillation probe operating in the current mode. As for the neutron measurements, the electronic equipment was located adjacent to the thimble to expedite data taking. Immediately after the reactor was taken to the desired operating power, the first gamma traverse was made to obtain the prompt gamma field associated with fission and neutron capture in the absence of activation gammas. Two traverses were made, one with the bare detector and one with a 0.9-inch thick lead shield surrounding the NaI probe. After completion of the neutron traverses, the gamma probe was reinserted into the thimble prior to shutting down the reactor to measure the field, which at that time included significant decay gammas from activation of sodium. Following shutdown of the reactor, traverses were made both with and without

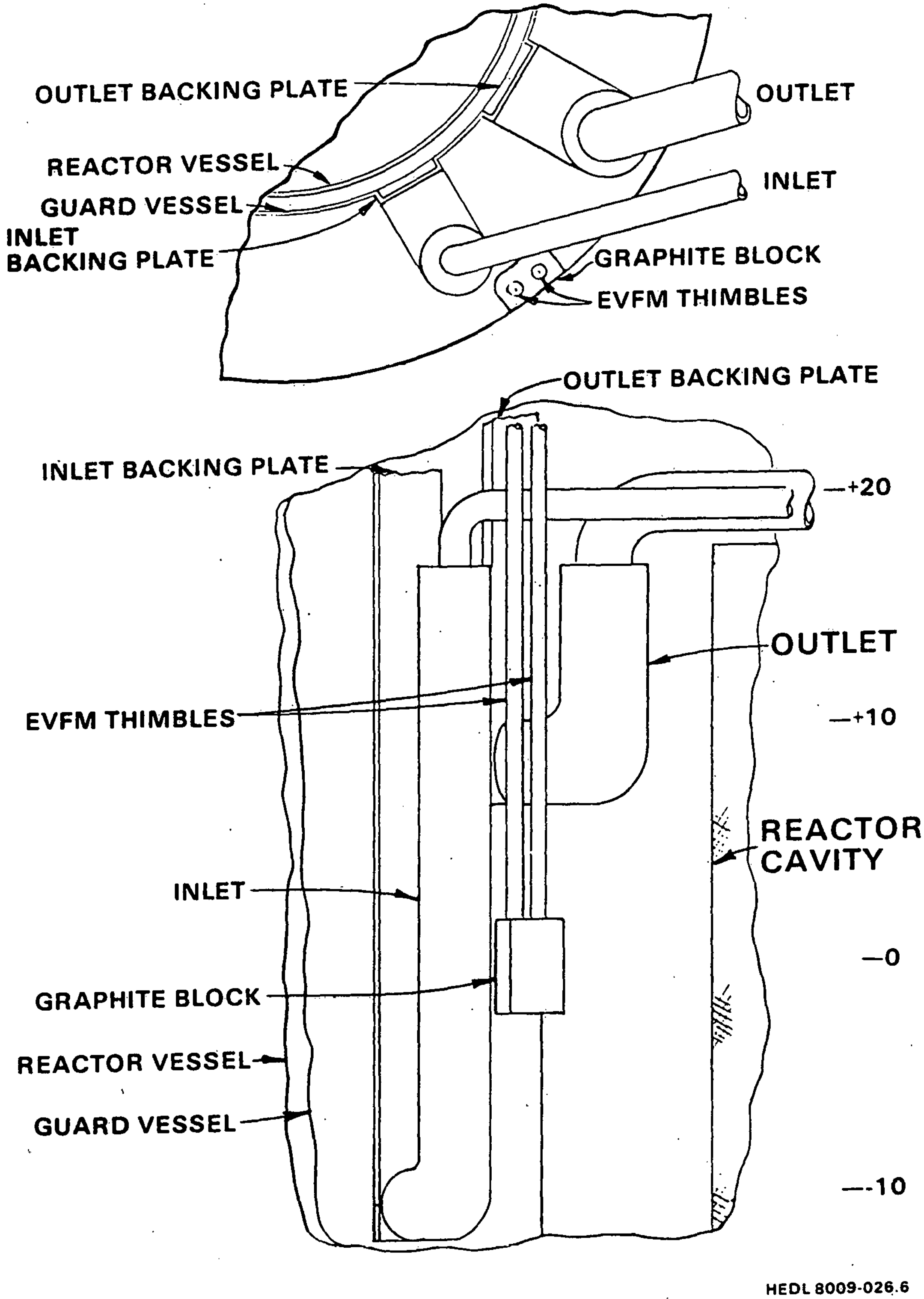

Figure 3. Location of Ex-Vessel Flux Monitor thimbles and moderator with respect to vessel and piping.

TABLE I

MEASURED VALUES FOR FISSION CHAMBER AND COMPENSATED IONIZATION CHAMBER

Elevation (Feet)		Bare Detector	Polyethylene Moderator 0.47 in.	Polyethylene Moderator 0.69 in.	Cadmium Cover	Lead Cover	Boroplaster Cover
FISSION CHAMBER (counts per second)							
-2		1979					
-1		2814					
* 0	Graphite	2994	3391	3170	∿100	2581	214
1		2994					
2		2271					
3		2299					
4		2223	7254	8237	534	1917	555
8		1227	3498	3781	231	1018	252
12		546					
16		278					
COMPENSATED IONIZATION CHAMBER (amps)							
-2		1.55×10^{-10}					
-1		1.82×10^{-10}					
* 0	Graphite	1.93×10^{-10}	1.8×10^{-10}	1.5×10^{-10}	0.12×10^{-10}		0.47×10^{-10}
1		1.85×10^{-10}					
2		1.70×10^{-10}					
3		1.75×10^{-10}					
4		1.60×10^{-10}	4.0×10^{-10}	3.9×10^{-10}	0.5×10^{-10}		0.8×10^{-10}
8		0.85×10^{-10}	1.9×10^{-10}	1.7×10^{-10}	0.21×10^{-10}		0.37×10^{-10}
12		0.3×10^{-10}					
16		0.13×10^{-10}					

* Nominal core midplane elevation of detectors in normal use.

the lead shield surrounding the probe. Finally, the gamma probe was placed at a fixed position to measure the decay of the gamma field as a function of time. Results of the gamma traverses are summarized in Figure 4.

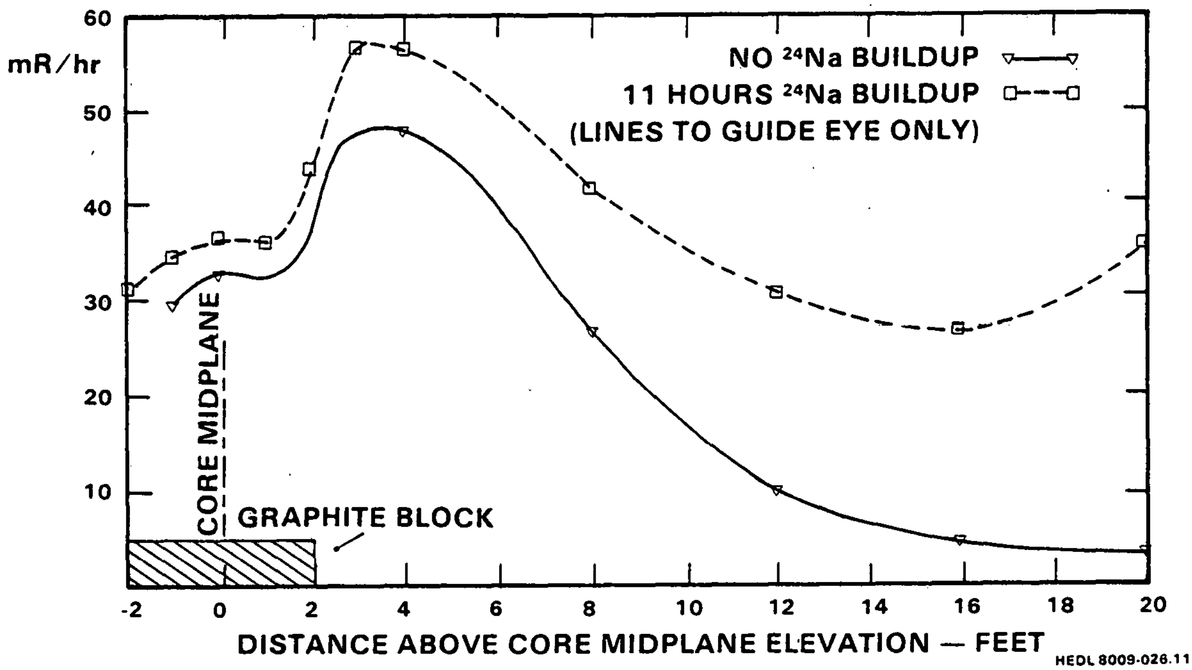

Figure 4. Gamma intensity profiles in Ex-Vessel Flux Monitor thimble.

C. REACTOR POWER LEVEL

In order to be able to relate the measurements to design calculations, it was necessary to establish the reactor power level during the test. The reactor has not been taken to power; therefore, a calorimetric determination has not been made. Two independent estimates of the power level were made and found to agree within expected uncertainties. The first power level estimate was made by relating the fission count rates of selected monitors to the subcritical status of the reactor based on source multiplication. The inherent neutron source of the fuel resulting from spontaneous fission and (alpha,neutron) reactions is relatively well known based on inventory of radioisotopes and decay half-lives. With the reactor relatively close to critical, the reactivity (multiplication factor) can be determined relatively accurately. This method of estimating the power is independent of the detector sensitivity. The second method of estimating the power relies on a calculation of the efficiency of the counter together with the flux distribution throughout the core. The greatest uncertainty with this estimate is associated with establishing the efficiency of the startup chamber, which was located near the geometric center of the core.

The detailed responses of the Low Level Flux Monitors (LLFM) during the test are shown in Figure 5. The safety rods were withdrawn from the core and the control rods were withdrawn (banked) until the reactor was slightly

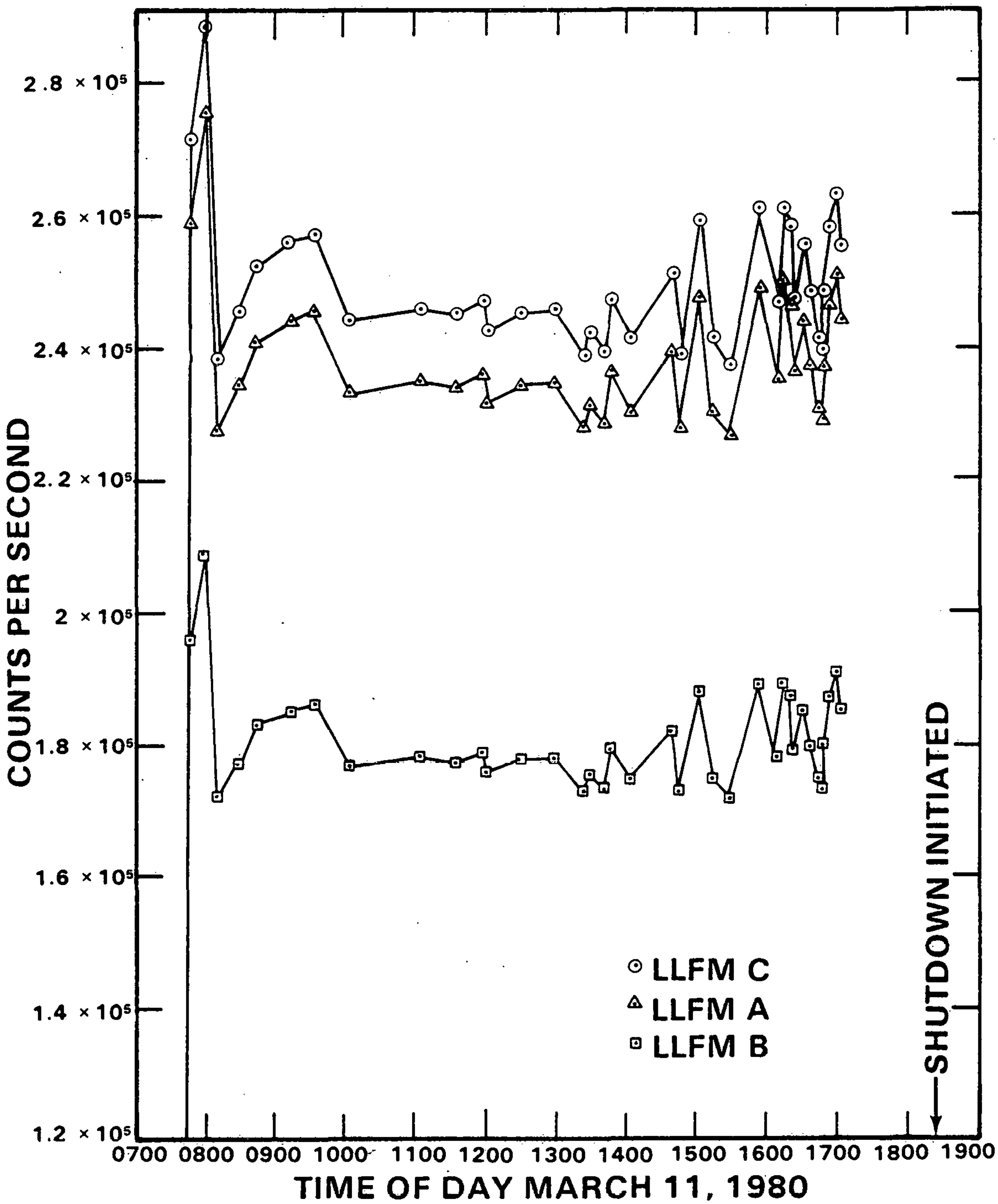

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Figure 5. Relative reactor power history during test period based on Low Level Flux Monitor system response.

subcritical. A single control rod was then moved to bring the LLFM count rate up to about 10% of the desired rate for a few minutes to check out various systems. The rod was then moved again to achieve the final operating count rate for the test. Based on the two methods described above, the reactor power level was estimated to be 6.8±1.1 kilowatts. Very little control rod movement was required to maintain the power level steady within test requirements.

ANALYSES

The attenuation calculations to support the design of the FTR radial shield were made at Oak Ridge National Laboratory (ORNL) using the two-dimensional discrete ordinates code DOT. A relatively detailed R-Z model of the reactor was employed, which required approximating the hexagonal core and reflector regions, and the twelve-sided shield region, as cylinders. As reported in Reference 2, supporting analyses were made to investigate the sensitivity of the results to input parameters such as cross sections. Output of the design calculation is stored on computer data tapes and includes the spatial, angular, and energy distributions of the neutrons and photons throughout the system, including the reactor cavity. An illustration of neutron flux profiles is shown in Figure 6. In the following sections, the present experimental results are compared to the applicable portions of these design calculations.

A. NEUTRON CALCULATIONS

To interpret the experimental results, three-dimensional Monte Carlo calculations were made using the MCNP computer code.[3,4] The energy-dependent incident neutron flux in the reactor cavity was taken from the ORNL design calculations. Only the reactor cavity, the graphite block and EVFM thimbles and sensors, and the reactor cavity wall were included in the Monte Carlo calculations, as illustrated in Figures 7 and 8. The calculations were normalized to a neutron current of one neutron per square centimeter per second incident upon the graphite block, with the angular dependence of the source taken as uniform in $\mu = \cos\theta$ for $0.8 \leq \mu \leq 1.0$. Results of these calculations are contained in Tables II and III, where they are also compared with the experimental values.

The neutron measurements were made both with the bare sensor and with the sensor surrounded by various shields. In part, these measurements were made to determine experimentally the changes that could be made in the EVFM system. However, the results also provide some spectral information. Within the graphite block (Table II), the agreement between calculated and measured response ratios is relatively good. The underprediction of the effect of the cadmium cover indicates the calculated spectrum is too hard. Above the graphite block (Table III), comparison of the calculated and measured ratios also indicates that the calculated spectrum is too hard; however, this may be due in part to the perturbation introduced by the adjacent graphite, which was not included in the Monte Carlo calculations.

In addition to establishing that the design calculations realistically predicted the neutron spectrum in the reactor cavity, it was also desired to compare the absolute flux levels. This was accomplished by measuring the

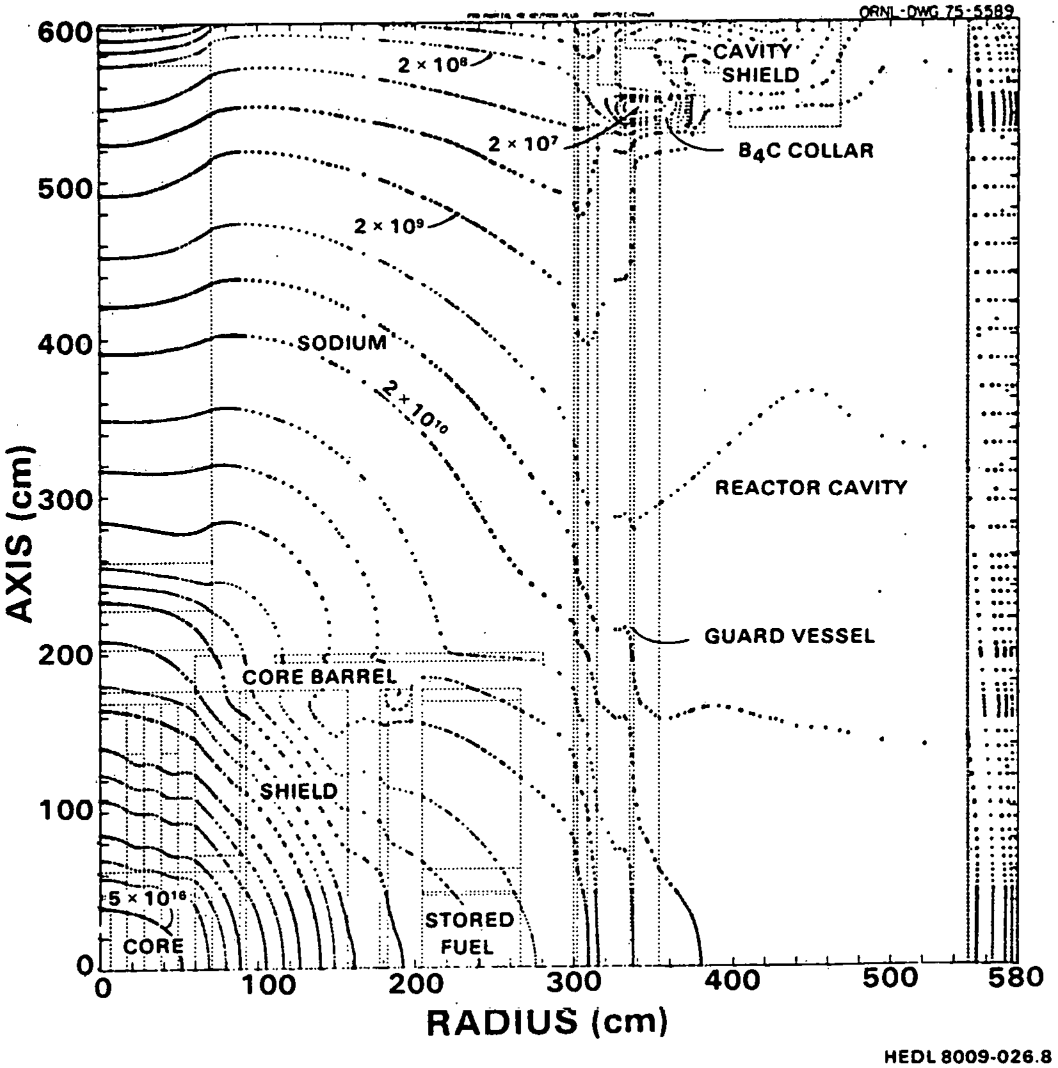

Figure 6. Neutron flux contours from DOT R-Z shield design calculation.

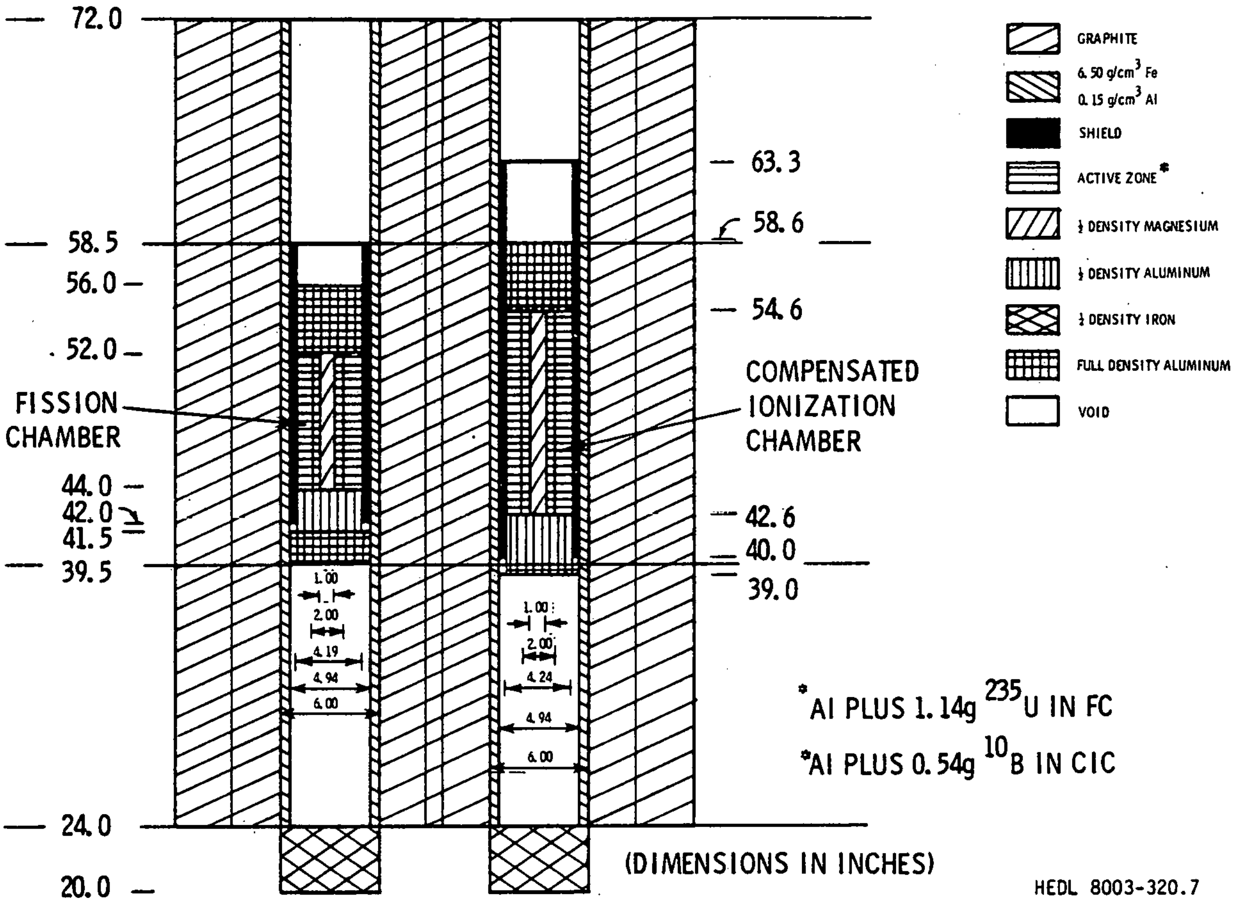

Figure 7. Geometry for Monte Carlo calculation of Ex-Vessel Flux Monitors, elevation view.

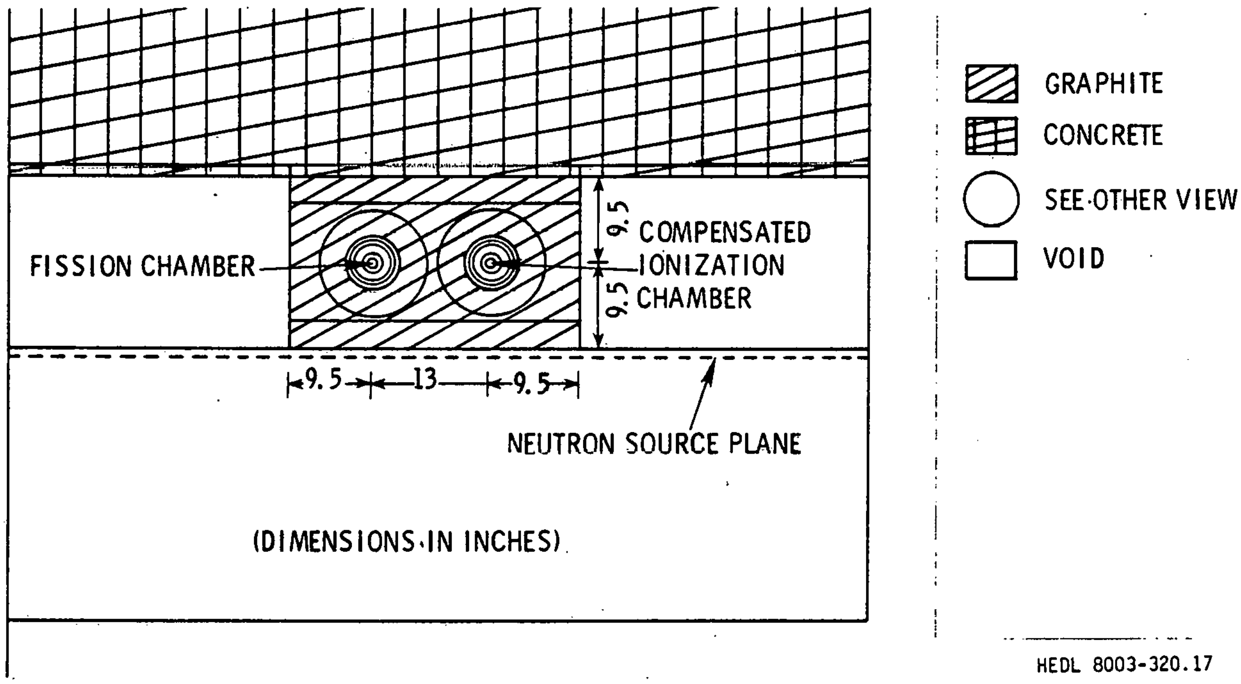

Figure 8. Geometry for Monte Carlo calculation of Ex-Vessel Flux Monitors, plan view.

TABLE II

CALCULATED AND MEASURED COMPARISONS FOR CHAMBERS WITHIN GRAPHITE BLOCK

Shield or Moderator	Compensated Ionization Chamber: Calculated: (n,α)'s per Gram of ^{10}B	Compensated Ionization Chamber: Calculated: Ratio of Response With Shield to Without	Compensated Ionization Chamber: Measured Ratio	Fission Chamber: Calculated: Fissions per Gram of ^{235}U	Fission Chamber: Calculated: Ratio of Response With Shield to Without	Fission Chamber: Measured Ratio
None	59.33	1.00	1.00	0.5095	1.00	1.00
Boroplaster[a]	13.01	0.22	0.24	0.0376	0.074	0.071
Cadmium[b]	5.68	0.096	0.062	0.0236	0.046	0.034
Lead[c]	58.65	0.99	---	0.4785	0.94	0.86
Polyethylene (.47 inches thick)	59.90	1.01	0.93	---	---	1.13
Polyethylene (.69 inches thick)	---	---	---	0.6492	1.27	1.06

(a) Radial thickness = 0.187 inches for compensated ionization chamber and 0.375 inches for fission chamber.

(b) Radial thickness = 0.05 inches.

(c) Radial thickness = 0.40 inches in calculations; subsequent measurements actually employed 0.90 inches.

TABLE III

CALCULATED AND MEASURED COMPARISONS FOR CHAMBERS ABOVE GRAPHITE BLOCK

Shield or Moderator	Compensated Ionization Chamber: Calculated: (n,α)'s per Gram of ^{10}B	Compensated Ionization Chamber: Calculated: Ratio of Response With Shield to Without	Compensated Ionization Chamber: Measured Ratio	Fission Chamber: Calculated: Fissions per Gram of ^{235}U	Fission Chamber: Calculated: Ratio of Response With Shield to Without	Fission Chamber: Measured Ratio
None	51.76	1.00 (0.87)[d]	1.00 (0.83)[d]	0.3582	1.00 (0.70)[d]	1.00 (0.74)[d]
Boroplaster[a]	22.13	0.43	0.50	0.1109	0.31	0.25
Cadmium[b]	19.17	0.37	0.31	0.1369	0.38	0.24
Lead[c]	53.01	1.02	---	0.3813	1.06	0.86
Polyethylene (.47 inches thick)	136.46	2.64	2.50	---	---	3.26
Polyethylene (.69 inches thick)	---	---	---	1.7362	4.85	3.71

(a) Radial thickness = 0.187 inches for compensated ionization chamber and 0.375 inches for fission chamber.

(b) Radial thickness = 0.05 inches.

(c) Radial thickness = 0.40 inches in calculations; subsequent measurements actually employed 0.90 inches.

(d) Ratio of response two feet above top of graphite block to response in center of graphite block for bare (no shield) counters.

inherent efficiency of the fission counter (counts per fission event) in a standard pile where the thermal neutron flux is calibrated by means traceable to the National Bureau of Standards. The spatial and energy dependent fission rate within the counter was calculated using discrete ordinates and Monte Carlo computer codes. The efficiency of the counter can be combined with the calculated fission rate and measured count rate to establish the total neutron flux. Mathematically, this can be stated as follows:

$$N = \int_E \phi(E) \ \Sigma(E) \ V \ p \ e \ dE \tag{1}$$

where N = count rate,
$\phi(E)$ = the energy dependent neutron flux in the counter volume, V,
$\Sigma(E)$ = the energy dependent macroscopic fission cross section of the fissile material in the counter,
p = the perturbation factor introduced by the presence of the counter,
e = efficiency of the counter, counts per fission event.

It is assumed that the flux can be defined by the product of a magnitude ϕ_o and an energy dependent function, $h(E)$:

$$\phi(E) = \phi_o \ h(E) \quad . \tag{2}$$

The constant or energy-independent values can be taken out of the integral and Eq.(1) rewritten:

$$N = e \ \phi_o \ V \ p \int_E h(E) \ \Sigma(E) \ dE = e \ \phi_o \ V \ p \ \bar{\Sigma} \quad . \tag{3}$$

In the standard pile where the flux is known, the relation is used to determine the counter efficiency, e, where the magnitude of the flux ϕ_o is known and the volume response function, $V \ p \ \bar{\Sigma}$ is calculated. The relation is then used to convert count rates in the reactor cavity to fluxes, where it is necessary to calculate a different response function. The total flux measured in this manner can then be compared with the total flux from the design calculations.

One additional correction is required before the experimental and calculational fluxes can be compared directly. During the measurements, the sodium pool and all reactor and shield materials were at a temperature of approximately 400°F. The shield design calculations were based on an operating reactor where the inlet sodium temperature was 600°F and the outlet 900°F. It was assumed the sodium in the annular region between the core barrel and the vessel wall would be at 800°F. One-dimensional discrete ordinates (ANISN) calculations were made to estimate the change in leakage flux associated with the change in sodium temperature. These calculations indicate that leakage flux per unit power will increase by about 25 percent as a result of increasing the sodium temperature from 400°F to 800°F because of the reduced density.

Based on this method, the measured neutron flux in the reactor cavity at the midplane elevation of the core is $1.51 \times 10^9 \ n/cm^2 sec$ when extrapolated to operation at 400 MW. The shield design calculations performed by ORNL yielded a total neutron flux of $2.03 \times 10^9 \ n/cm^2 sec$. This agreement is better than anticipated and is well within the uncertainty limits of the measurements and calculations.

B. GAMMA CALCULATIONS

The design calculations by ORNL[2] also included the prompt gamma field associated with fission and neutron capture. A second important component of the gamma field is that associated with the decay of sodium. The isotope ^{24}Na, which is formed as a result of neutron capture, has a beta decay half-life of 15 hours and emits both a 2.75 and a 1.37 MeV photon during the process. Because the ORNL design calculations did not include sodium decay gammas, we made point kernel gamma ray calculations of this source using the computer code ISOSHLD.[5] Calculations were made for the reactor vessel as well as for the piping, which runs quite close to the EVFM thimble (see Figure 3). Activation of the sodium was based on an earlier two-dimensional diffusion theory calculation of FTR.[6] This calculation, which predicts an equilibrium activity at 400 MW of about 10 mCi/cc, was substantiated by experimental results obtained in the Engineering Mockup Critical tests.[7]

The prompt gamma field in the reactor cavity at the midplane elevation of the core was calculated by ORNL to be 3×10^6 mR/hr. When the measured gamma field of 55±5 mR/hr at a power level of 6.8 kW is extrapolated to 400 MW, the measured prompt gamma field becomes 3.2×10^6 mR/hr, in very good agreement with the design calculations. For sodium decay gammas, the equilibrium activity at 400 MW was calculated to give a field of 4.6×10^6 mR/hr at the same elevation and at the position of the EVFM thimble but in the absence of the graphite moderator block. The measured (extrapolated) value of 25 mR/hr at the core midplane elevation after eleven hours of operation at 6.8 kW yields a 400 MW equilibrium value of 3.7×10^6 mR/hr. For operation of the detectors inside the graphite moderator block, the measurements indicate that the prompt gamma field will be 1.1×10^6 mR/hr and the sodium decay gamma field will be 1.7×10^6 mR/hr, giving a total field of 2.8×10^6 mR/hr. This field is sufficiently low so that it will be unnecessary to add lead gamma shielding, which would reduce the field to 1.2×10^6 mR/hr.

An interesting aspect of the gamma measurements was the discovery of a short half-life gamma field that had not been anticipated. We at first attributed this entirely to the activation of manganese in the steel. The isotope ^{56}Mn, which would be generated as a result of neutron capture, has a half-life of 2.58 hours and emits 0.85 MeV photons. Stripping off the 15-hour sodium half-life indicated the additional presence of a half-life of the order of one-half hour. This indicated that the iodine in the NaI crystal might have become activated, since ^{128}I has a half-life of 25 minutes. In order to confirm the source of the short half-life gamma field, the NaI detector was placed in the standard neutron flux field used to obtain the neutron counter efficiency and irradiated for a sufficiently long time to activate the iodine. When removed from the neutron field, the current from the self-induced gamma field in the crystal was measured to be the equivalent of 0.8 mR/hr compared to a calculated value for the standard neutron flux field of 0.9 mR/hr. An accurate time- and space-dependent irradiation history of the detector in the reactor cavity was not made; therefore, it was not possible to accurately relate the NaI activity to the neutron flux. The laboratory experiment did confirm that activation of the iodine was a major constituent of the short half-life gamma field that was detected.

In a second laboratory experiment, a piece of three-inch schedule 40 steel pipe was irradiated in the standard neutron field for several hours to activate the manganese. When the pipe was withdrawn from the neutron flux, the NaI detector was placed inside the pipe and the gamma field was measured. In this case the gamma field was measured to be about 0.02 mR/hr compared to a calculated field of 0.015 mR/hr for this piece of pipe. Although not precise, these measurements also confirmed the presence of the manganese activity in the reactor cavity. The most important conclusion from this particular effort was to confirm our ability to calculate the gamma field associated with activation of hardware. This was important for the long term operation of the EVFM system to assure that an intolerable background gamma level at the detector would not be generated after several years of operation at power.

SUMMARY AND CONCLUSIONS

Axial traverses in one of the thimbles for the EVFM system of the FTR were made to determine the neutron and gamma field in the reactor cavity. These measurements are in good agreement with the design calculations for the radial shield of the FTR. They also confirm predictions of the activation of sodium. In addition, the measurements provided a firm basis for making adjustments in the electrical portions of the EVFM system in order that the readouts could be realistically related to reactor power level. Finally, laboratory measurements confirmed the calculations that predict that long-term activation of hardware in the reactor cavity will not compromise the EVFM system by introducing excessive gamma fields.

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