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FAST REACTOR ANALYTICAL SHIELDING
PROGRESS REPORT

FOR JULY AND AUGUST 1973

189a No. 10318, Activity No. HN 04 01 01 1

F. R. Mynatt

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Monthly Progress Report
July and August 1973
FAST REACTOR ANALYTICAL SHIELDING

Activity No. HN 04 01 01 1

189a No. 10318

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CONTENTS

	<u>Page No.</u>
1. CURRENT FFTF STUDIES -----	1
1.1. Coupled DOT-MORSE Calculations of Radiation Streaming Through Penetrations in the Radial Cavity Shield -----	1
1.2. Estimate of Peak Dose on Maintenance Deck Due to Radiation Streaming Through Penetrations of the Radial Cavity Shield -----	6
1.3. Updated DOT Calculations of Heat Temperature Control System Shield -----	8
1.4. Analysis of Branch-Arm Pipe Shielding -----	10
1.5. Calculations of ^{24}Na Gamma-Ray Dose Rates at 28-in. and 16-in. Penetrations from HTS Pipeway to IHX Cell -----	15
2. CURRENT DEMO PLANT STUDIES -----	21
2.1. Calculations of Neutron Attenuation in Lower Axial Shield -----	21
2.2. DOT R- θ and R-Z Calculations of Demo Stored-Fuel Power and Ex-Vessel Detector Response -----	29
3. CODE DEVELOPMENT: SPECIFICATIONS FOR DOT IV -----	30
4. ANALYSIS OF TSF IRON/POLYETHYLENE SECONDARY GAMMA-RAY EXPERIMENT -----	32
4.1. Monte Carlo Calculations (220 Groups) -----	34
4.2. Discrete Ordinates Calculations (39 Groups) -----	38
5. ANALYSIS OF TSF WESTINGHOUSE DEMONSTRATION PLANT EXPERIMENTS -----	56
5.1. Spectrum Modifier -----	56
5.2. Spectrum Modifier Plus 5-in. Radial Blanket -----	65
6. STUDY OF THE SENSITIVITY OF CALCULATIONS OF THE TSF INCONEL EXPERIMENT TO NICKEL CROSS SECTIONS: CORRELATION WITH CALCULATIONS OF NEUTRON DAMAGE TO FTR GRID PLATE -----	72

ABSTRACT

Analytical shielding work performed for the fast reactor program during the months of July and August included calculations for both the FFTF and the Demo plant, computer code development, analyses of TSF experiments, and a cross-section sensitivity study of a TSF experiment. Most of the FFTF studies were aimed at determining the effect of various design changes on the dose rates at the maintenance deck during reactor operation, in particular, the effects of penetrations through the radial cavity shield and the effect of a new design for the head temperature control system shield. Gamma-ray dose rates due to the activated sodium coolant were also calculated for two locations in the system: around one of the branch-arm piping shields and near the entrances of the 28-in. and 16-in. coolant lines into the IHX cell. The main Demo plant study was for the lower axial shield region; another study has been initiated to calculate the Demo stored-fuel power and the ex-vessel detector response. The code development consisted in setting forth a set of specifications for the fourth version of the discrete ordinates code DOT. The TSF experiments that were analyzed were an experiment in which secondary gamma-ray production in iron-polyethylene configurations was investigated and another in which the transmission of the Westinghouse spectrum modifier and 5-in. radial blanket was measured. The sensitivity study was performed to demonstrate that estimates of the accuracy of calculations for the FFTF and Demo plant can be deduced from a sensitivity study of an experimental configuration that only roughly mocks up the real system.

FAST REACTOR ANALYTICAL SHIELDING PROGRAM - FY 1974

	JULY	AUG	SEPT	OCT	NOV	DEC	JAN	FEB	MAR	APR	MAY	JUNE
<u>Design and Analysis of Experiments</u>												
FTR Iron/Poly Secondary Gamma Experiment												
FTR Inconel Experiment												
FTR Preliminary Stored-Fuel Experiment												
Demo Preliminary Stored-Fuel Experiment												
FTR Stored-Fuel Experiment												
Demo Stored-Fuel Experiment												
TLD Check Experiment												
keV Total Cross-Section Experiment												
Demo Radial Shield Experiment												
Demo Lower Axial Experiment												
Support Structure Streaming Experiment												
<u>FTR Design Support</u>												
RCS Penetrations												
HCS Neutron Shielding												
BAPS Neutrons and Gammas												
Center Island Shadow Shield												
HTS-SDO 200 Analysis												
Stored-Fuel 3D Coupling												
<u>Demo Design Support</u>												
Lower Axial Streaming												
R-0 RSP Analysis I												
R-0 RSP Analysis II												
Radial Cavity & Support System												
Head Compartment Shielding												
IHX Shielding												
<u>Code Development</u>												
DOT IV												
COMGOM Improvement												
2-D Sensitivity Code												
FACT Improvement												
MINX-AMPX Modifications												

1. CURRENT FFTF STUDIES

On July 25, 1973, a meeting was held at ORNL for the purpose of reviewing the FFTF shield design support activity. At that time most of the studies of the effects of penetrations or irregularities in the radial cavity shield (RSC) reported in Section 1.1 below had been performed, as had the flux peaking studies given in Section 1.2, and ORNL agreed to prepare a letter report of the results. ORNL also agreed to extend the study to analyze the effect of the 3-in. makeup line penetration and to attempt a calculation containing all of the penetrations in the hope of getting better statistics. For the latter, ARD agreed to provide ORNL with drawings describing a revision of the 16-in. pipe penetration of the RCS. Our letter report (No. FTRL-100), including the results for the 3-in. makeup line, was mailed on August 29.

At the same meeting ARD presented drawings showing the new head compartment shielding concept in which borated polyethylene is placed on the Head Temperature Control (HTC) system ducting. ORNL agreed to analyze a general elevation view model and follow with a study of the interface between the HTC shield and head components. The results to date are given in Section 1.3.

Additional work planned included neutron and gamma-ray attenuation analyses of two sections of the revised Branch Arm Pipe Shield (BAPS) designed by ARD, and an analysis of the SDO-200 conditions for the HTS pipeway and IHX vault (see Sections 1.4 and 1.5).

1.1 Coupled DOT-MORSE Calculations of Radiation Streaming Through Penetrations in the Radial Cavity Shield

Discrete ordinates (DOT) calculations of radiation penetrating to the FTR maintenance deck through the radial cavity shield have been under way for several months¹ using the cylindrical geometry shown in Fig. 1.1. Since in such calculations the components of the assembly must be described as symmetrical regions around the cylinder axis, penetrations

Fig. 1.1. Geometry for DOT FTR Calculations Including Radial Cavity Shield.

through the radial cavity shield, which require three-dimensional descriptions, were not included. The effects of these penetrations have now been investigated by utilizing the MORSE Monte Carlo code for the region in which the penetrations exist and using the DOMINO code to couple the MORSE runs with DOT runs for the regions above and below the shield. Although these coupled calculations were begun in May, none of the results from this phase of the study have been reported previously. Thus the following paragraphs summarize all the radial-shield penetration studies performed through August.

The penetrations investigated are the SISI transporter duct, the 16-in. inlet line, the 6-in. overflow line, and the 3-in. makeup line as described in Bechtel Drawings H-4-18221, Workprint 2/5/73. In all cases the calculations were performed for a simple head compartment shield of 3 in. of borated polyethylene (6 wt% natural boron) and 1 in. of steel at the maintenance floor level as shown in Fig. 1.2.

From one of the earlier forward DOT calculations a boundary angular flux tape had been saved for $z = 532.42$ cm. The DOMINO code used this tape and other information from the DOT run to generate tables of probabilities for selecting the source group, position and direction for five forward MORSE runs -- one for each of the penetrations listed in the preceding paragraph plus a run for the solid radial shield. The MORSE calculations covered the region up to $z = 631.2$ cm, at which point a collision tape was written for all boundary crossings. The geometry used for the MORSE runs, slightly more detailed than that used in the DOT runs, is shown in Fig. 1.3.

Above the radial cavity shield, results were used from four different adjoint DOT runs, the angular flux tapes from each run having been processed by DOMINO. The sources for the adjoint calculations were the neutron flux-to-dose conversion factor and the gamma-ray flux-to-dose conversion factor, each being placed at the top of the maintenance deck and distributed radially from 260 to 450 cm in one pair of cases and from 325 to 400 cm in the other pair (see Fig. 1.2). The DOMINO tapes generated from these adjoint calculations were used as scoring functions for each MORSE run.

Fig. 1.2. Geometry for DOT Adjoint Calculations for FTR Region Above Radial Cavity Shield.

Fig. 1.3. Geometry for MORSE Calculations for FTR Region Containing Radial Cavity Shield.

These calculations indicated that the number of boundary crossings occurring in the significant groups and regions is quite small. To limit the amount of machine time required to obtain acceptable statistics, source energy biasing parameters were calculated using fluxes from an adjoint DOT run and the source position was restricted to the most significant spatial regions.

Table 1.1 gives the integrated dose rate on the maintenance deck from each coupled case calculated, as well as from an all-DOT run for the solid radial cavity shield. Since a maximum increase in the integrated dose rate due to a penetration in the shield is 20%, it appears that the penetrations have no significant effect. However, in these calculations only a single penetration was included in the shield in each case. Calculations now in progress will consider the combined effect of the three SISI transporters, the three 16-in. inlet lines, the 6-in. overflow line, and the 3-in. makeup line.

1.2. Estimate of Peak Dose on Maintenance Deck Due to Radiation Streaming Through Penetrations in Radial Cavity Shield

The DOT-DOMINO-MORSE-DOMINO-DOT calculations discussed in Section 1.1 have given the average dose in the adjoint source area (on the FTR maintenance deck) for various radial cavity shield penetrations. To get an estimate of the peak dose, boundary crossing detectors were placed in the MORSE cases at $z = 631.2$ cm (see Fig. 1.1), and these indicated that in most cases the peak neutron dose in the radial cavity directly above the radial cavity shield was increased by approximately a factor of five when the shield configurations included a penetration. In order to estimate the effect that this higher dose above the shield would have on the dose penetrating to the maintenance deck, two 21-18 group DOT cases were run to estimate the local neutron peaking after penetration through the 21.75-in. reactor vessel head. In one case, a neutron source was distributed uniformly over the bottom of a 21.75-in.-thick, 78.74-in.-diam iron cylinder; in the other case, the source out to $r = 5.5$ in. was increased

Table 1.1. Integrated Dose on Maintenance Deck for
Various Radial Cavity Shield Penetrations

Configuration ^a	Integrated Dose (cm ² -mrem/hr)	
	Gamma Rays	Neutrons
<u>Adjoint Source at R = 260-450 cm</u>		
Solid Shield (DOT)	5.81+5	7.64+5
Solid Shield	5.82+5(0.06) ^b	7.68+5(0.11)
SISI Transporter Duct	6.12+5(0.05)	9.25+5(0.06)
16-in. Inlet Line	6.33+5(0.08)	8.83+5(0.10)
6-in. Overflow Line	6.61+5(0.14)	6.97+5(0.11)
3-in. Makeup Line	6.34+5(0.14)	9.12+5(0.12)
<u>Adjoint Source at R = 325-400 cm</u>		
Solid Shield (DOT)	3.0+5	4.65+5
Solid Shield	3.01+5(0.06)	4.72+5(0.11)
SISI Transporter Duct	3.02+5(0.05)	5.64+5(0.06)
16-in. Inlet Line	3.28+5(0.08)	5.4+5(0.10)
6-in. Overflow Line	3.43+5(0.14)	4.25+5(0.11)
3-in. Makeup Line	3.30+5(0.15)	5.55+5(0.12)

^aExcept for solid-shield DOT calculations, all calculations are coupled DOT-MORSE-DOT calculations.

^bRead: 5.82×10^5 cm²-mrem/hr with 6% fractional standard deviation.

by a factor of five. In the second case the neutron dose at the top center of the iron cylinder was 1.6 times higher than that in the first case.

On the basis of this peaking factor, together with the new value obtained for the stored-fuel power relative to the core power,¹ we can now update the peak dose rates on the maintenance deck previously calculated for the FTR with the solid magnetite shield.¹ These dose rates were 0.6 mrem/hr for neutrons and 0.4 mrem/hr for secondary gamma rays. The effect of the new value for the stored-fuel power would be to increase the neutron dose rate by a factor of 1.21 and the gamma-ray dose by a factor of 1.14, resulting in values of 0.85 mrem/hr and 0.53 mrem/hr for neutrons and gamma rays respectively. With the neutron local peaking factor of 1.6 applied to both of these, which would be conservative for the gamma rays, the new values for the neutron and gamma-ray dose rates are 1.36 mrem/hr and 0.854 mrem/hr respectively. This is reasonably close to a total dose rate not to exceed 2 mrem/hr.

1.3 Updated DOT Calculations of Head Temperature Control System Shield

A general evaluation view of the head temperature control system shield based on HEDL Drawing H-4-23442 was mocked up for a 21-18 group DOT run. The geometry, shown in Fig. 1.4, is updated over that used for the radial cavity shield calculations (compare with Fig. 1.2). The mockup includes 4 in. of 2% borated polyethylene and 1 in. of steel approximately 4 in. above the vessel head from $r = 320$ cm to $r \approx 555$ cm. Inboard of $r = 320$ cm, a steel cylindrical box (2-in.-thick walls) simulates the reactor appurtenances above the head. A 3-in.-thickness of borated polyethylene is adjacent to the outer wall of the steel box. The reactor model used in this calculation is that used in the April 1973 study, and the stored fuel/core power ratio was the old value of 5.3×10^{-4} .

Fig. 1.4. Geometry for DOT FTR Calculations of Head Temperature Control System.

Figures 1.5-1.7 are contour plots of the calculated total neutron fluxes, neutron dose rates, and gamma-ray dose rates for this geometry. It is apparent that there is a significant neutron streaming effect just in-board of the steel box at $r = 320$ cm. As a result, the peak neutron dose rate at the maintenance deck ($z = 933$ cm) is approximately 8 mr/hr at $r = 280$ cm and the peak gamma-ray dose rate is 0.33 mr/hr at $r = 374$ cm.

The analysis will continue by examining in more detail the interface of various reactor components with the vessel head. In particular, it is expected that a better representation of the reactor components above the head will reduce the dose rates to more acceptable levels.

1.4. Analysis of Branch-Arm Pipe Shielding

The analysis of the shield designs for the branch-arm pipes has continued with a DOT calculation for the left side of Section B-B as shown in Dwg. No. H-4-2321. The iron shield thickness is the same on both sides, but the sodium pipes are closer to the left side. Borated polyethylene and steel plates were added to the shield to correspond to the latest design; however, for the first (non-streaming) calculations the insulation regions below the pipes were assumed to be filled with iron. The calculation, performed for an X-Y geometry, used SMUG 18-group P_4 -corrected cross sections and an S_{10} quadrature. The source strength in the sodium pipes was assumed to be 12 mCi/cm³.

Contour plots of the results are shown in Fig. 1.8. (The geometry is flipped in the figure.) The first contour line is 10^5 mrem/hr, decreasing to a maximum of about 8 mrem/hr at the outside edge of the iron shield (approximately 19 in. above the top of the head). The dose rates just left of the borated polyethylene region are in the range of 2 to 4.4 mrem/hr, the maximum again occurring at an elevation of about 19 in. above the head. Dose rates at the top of the maintenance deck are in the range of 0.1 to 0.25 mrem/hr for about 3 ft to the left of the section of

Fig. 1.5. Contour Plots of Neutron Fluxes Calculated in Study of Head Temperature Control System Shield.

Fig. 1.6. Contour Plots of Neutron Doses Calculated in Study of Head Temperature Control System Shield.

Fig. 1.7. Contour Plots of Gamma-Ray Doses Calculated in Study of Head Temperature Control System Shield.

Fig. 1.8. Gamma-Ray Dose-Contour Plots (in mrem/hr) for Left Side of Branch-Arm Piping Shield in Section B-B (Geometry Flipped in Figure).

the shield extending above the maintenance deck. The maximum value is at a point 22 in. to the left of the top section.

Additional calculations were performed to determine the effect of streaming through the insulation sections below the pipes. For these calculations the geometry was cut off just above the bottom pipe and calculations were performed in one case with iron in the insulation regions and in another case with air in the insulation regions. For these calculations a 210-angle quadrature was used. The resulting contour plots (not flipped in these cases) are shown in Figs. 1.9 and 1.10. A comparison of the two figures indicates that the air regions significantly increased the dose rates penetrating to the outside of the shield just above the head (by a factor of approximately 400).

1.5. Calculation of ^{24}Na Gamma-Ray Dose Rates at 28-in. and 16-in. Penetrations from HTS Pipeway to IHX Cell

MORSE calculations have been performed to obtain ^{24}Na gamma-ray dose rates at several locations near the entrances of the 28-in. and 16-in. sodium inlet and outlet lines to the HTS cell when the isolation valves were closed and the sodium in the pipes in the HTS cell itself had been allowed to decay. The results are to be compared by ARD with the dose rates they calculated with a single-collision point-kernel code. The configuration used was the cell geometry for MORSE given in FRP-1165 (Fox to Mynatt, December 19, 1972) with some minor modifications of the materials and dimensions.

The relative locations of the detector points are shown in Fig. 1.11, where $x, y = 0$ at the center of the reactor vessel and the z values are at the horizontal midplanes of the pipes. Preliminary calculations with poor statistics were made for all points except I to determine the relative magnitudes of the dose rates (see Table 1.2). These calculations considered only single scattering and did not include the stainless steel piping. The results indicated that the highest dose rates were at the

A = Na
B = Air

C = Fe
D = Borated Polyethylene

E = Air in Insulation Gap

Fig. 1.9. Gamma-Ray Dose-Rate Contour Plots (in mrem/hr) for Lower Region of Branch-Arm Piping Shield (Section B-B): Air (Void) in Insulation Gaps.

A = Na C = Fe E = Fe in Insulation Gap
 B = Air D = Borated Polyethylene

Fig. 1.10. Gamma-Ray Dose-Rate Contour Plots (in mrem/hr) for Lower Region of Branch-Arm Piping Shield (Section B-B): Fe in Insulation Gaps.

Fig. 1.11. Detector Locations for Calculations of ^{24}Na Gamma-Ray Dose Rates in IHX Cell.

Table 1.2. ^{24}Na Gamma-Ray Dose Rates at 28-in. and 16-in. Entrances to HTS Cell

Configuration	Detector Location (see Fig. 1.11)	Calculational Method	Dose Rates (r/hr)	
			Uncollided	Total
28-in. Pipe, Na Only ^a	A	MORSE, single collision	2.88-6(2.12) ^b	2.75-1(57)
	B	MORSE, single collision	1.14-4(2.61)	1.12(71.5)
	C	MORSE, single collision	2.04-1(0.70)	8.54-1(14.2)
	D	MORSE, single collision	5.85-1(0.60)	3.27(20.0)
	D	Point kernel		3.98
16-in. Pipe, Na Only ^a	E	MORSE, single collision	9.28-5(1.67)	1.56(83.7)
	F	MORSE, single collision	6.82-3(2.28)	1.69(50.5)
	G	MORSE, single collision	1.07(0.80)	4.96(18.6)
	H	MORSE, single collision	6.07(2.08)	16.96(19.6)
	H	MORSE, single collision (Rerun) ^c	6.38(0.60)	16.84(6.5)
	H	Point kernel		28.9
	H	MORSE, multiple collisions	6.3(1.9)	31.6(21.3)
16-in. Pipe, Na in SST 304 Pipes, with Shield Patches	I	MORSE, multiple collisions	1.16-2(1.2)	1.17(9.88)

^aNo stainless steel piping was included in the calculation.

^bRead: 2.88×10^{-6} with 2.12% standard deviation.

^cRerun to obtain better statistics.

outermost points (D and H), the highest being at point H. The calculation for point H was then rerun to obtain statistics less than 10%. In addition, a point-kernel calculation was performed for point H, as well as for point D, to provide a rough check on the magnitude of the MORSE results. For the 16-in. pipeway, the kernel results were higher by a factor of about two than the MORSE results, indicating that the single-collision approximation was inadequate. An additional MORSE run for this case with multiple scattering included also gave results higher by a factor of approximately two.

Since the dose rate at point H was much too high, patches were added to the concrete wall as shown in the inset in Fig. 1.11, and the dose rate was calculated for point I. (The patch adjacent to point I is doughnut-shaped.) The dose rate was reduced by a factor of 30, but this was still inadequate. The results indicate that the shield patches are much more effective in reducing the uncollided component of the dose rate than the collided component, and that a more judicious placement of the shield patches might do the job. However, further calculations are indicated before a more definitive statement can be made.

Section 1. Reference

1. F. R. Mynatt et al., "Fast Reactor Analytical Shielding Progress Report for May and June, 1973," ORNL-TM-4401 (to be issued).

2. CURRENT DEMO PLANT STUDIES

2.1. Calculations of Neutron Attenuation in Lower Axial Shield

DOT calculations of the transport of fission neutrons down through the axial blanket and on through the lower axial shield of the Demo plant were performed to obtain estimates of the lifetime of the grid plate located below the shield. As currently designed, the blanket is a 14-in.-thick region of uranium oxide and the shield, which is separated from the blanket by a 4.1-in.-thick fuel rod attachment, is a 29.5-in.-thick region of type 316 stainless steel. The shield is to be fabricated as closely fitting pieces that are hexagonal in cross section and penetrated by orifices to allow the flow of sodium coolant over their full lengths. The location of the coolant holes in each piece is shown in Fig. 2.1.

The geometry for the calculations included the lower half of the core (half-height = 18 in.) and extended radially into the sodium region beyond the radial restraint. Below the core it extended well below the grid plate -- a 2-in.-thick type 304 stainless steel plate located about 3 in. below the shield. The various regions in the geometry are numbered in Figs. 2.2 - 2.4 (to be discussed below) and are identified in Table 2.1.

The fission source distribution within the core was obtained from DOT diffusion theory calculations. In one calculation the central control-rod channel was assumed to be filled with sodium, and in a second calculation it was assumed to be filled with fuel. The corresponding k_{eff} values were 0.979 and 0.983 respectively. Because of these low values, a separate S_4P_1 calculation was performed for the case of sodium in the central channel. When this calculation yielded a k_{eff} value of 0.984, it was assumed that the diffusion theory calculations were correct. For the shield calculations with the sodium-filled channel, however, the fission source distribution obtained from the S_4P_1 run was used rather than that from diffusion theory.

Fig. 2.1. Cross Section of Axial Shield Section in Demo Plant Showing Location of Coolant Holes. Dashed lines indicate regions used in calculation (see text).

The calculations for the full geometry were performed for two shield cases: one in which the entire lower axial shield was represented by a homogenized region of 74.6% type 316 stainless steel and 25.45% sodium, and another in which the center section of the shield more nearly resembled the real case. It was described as a central cylinder of sodium (representing the center hole) and three surrounding cylindrical annuli as shown by the dashed lines in Fig. 2.1. The first annulus was steel, the second a homogenized region of sodium and steel (to include the other six coolant holes), and the third a final region of steel. The remainder of the shield was the same as in the homogenized case.

The calculations were performed with 50 energy groups, 147 angles, and variable P_ℓ scattering. Dose-rate results from the homogenized-shield runs are shown in Figs. 2.2 and 2.3 for the cases of the central control-rod channel being filled with sodium and fuel respectively. (When sodium was in the central channel the region extended down through the blanket.) The results from the run in which the center section of the shield was mocked up are shown in Fig. 2.4, the fission source distribution being that calculated for the case of fuel in the central control-rod channel.

The total neutron fluxes calculated for specific locations within the assembly are given in Table 2.2. With the fluxes shown for the grid plate, together with a maximum total fluence of 9.5×10^{21} nvt to be allowed at the plate (an FFTF criterion), the life times of the grid plate would be 166 yr, 194 yr, and 193 yr for Cases 1, 2, and 3 respectively. Folding energy-dependent damage functions supplied by HEDL with the fluxes yields lifetimes of 773 yr, 906 yr, and 906 yr for Cases 1, 2, and 3 respectively. However, the normalization of the energy-dependent response function is suspect and we will determine the effect of more recent damage functions.

Table 2.1. Description of Regions in Demo Axial-Shield Calculations

Region No.	Identification	Material
1a	Control rod channel No. 1 (central channel) (radius = 2.516 in.)	Fuel
1b*	Control rod channel No. 1 (central channel)	Sodium
2	Inner core (18-in. half-height)	Fuel
3	Control rod channel No. 2	~95% sodium, ~5% stainless steel
4	Inner core	Fuel
5	Control rod channel No. 3	~95% sodium, ~5% stainless steel
6	Inner core	Fuel
7	Control rod channel No. 4	~95% sodium, ~5% stainless steel
8	Outer core	Fuel
9	Radial blanket	Depleted uranium oxide
10	Radial reflector	~90% stainless steel, ~10% sodium
11	Radial restraint	Inconel
12-13	Radial sodium zones not included in figures	
14	Lower inner axial blanket (14 in. thick)	Depleted uranium oxide
15	Lower outer axial blanket	Depleted uranium oxide
16	Radial blanket extension	Depleted uranium oxide
17	Fuel rod attachment (4.1 in. thick)	80% sodium, 20% type 316 stainless steel
18	Center of mockup of shield section (radius = 0.4375 in.)	Sodium
19	First annulus of shield section mockup (thickness = 0.75 - 0.4375 in.)	Type 316 stainless steel
20	Second annulus of shield section mockup (thickness = 1.625 in. - 0.75 in.)	55.26% sodium, 44.74% type 316 stainless steel
21	Third annulus of shield section mockup (thickness = 2.516 in. - 1.625 in.)	Type 316 stainless steel
22	Shield (29.5 in. thick)	25.4% sodium, 74.6% type 316 stainless steel
23	Transition zone (2.9 in. thick)	65% sodium, 35% type 316 stainless steel
24	Grid plate (2 in. thick)	36% sodium, 64% type 304 stainless steel
25	Transition zone (14 in. thick)	65% sodium, 35% type 316 stainless steel

*Zone 1b extends down through blanket.

Fig. 2.2. Neutron Dose-Rate Contours (mrem/hr) in Demo Lower Axial Shield and Grid-Plate Region: Sodium in Central Control-Rod Channel; Homogenized Shield (Case 1).

Fig. 2.3. Neutron Dose-Rate Contours (mrem/hr) in Demo Lower Axial Shield and Grid-Plate Region: Fuel in Central Control-Rod Channel; Homogenized Shield (Case 2).

Fig. 2.4. Neutron Dose-Rate Contours (mrem/hr) in Demo Lower Axial Shield and Grid-Plate Region: Fuel in Central Control-Rod Channel; Center Section of Shield Represented by Concentric Annuli (Case 3).

Table 2.2. DOT-Calculated Neutron Fluxes on
Center Line of Demo Plant

Location	Neutron Flux* (neutrons·cm ⁻¹ ·sec ⁻²)		
	Sodium in Control-Rod Channel; Homogenized Shield (Case 1)	Fuel in Control-Rod Channel	
		Homogenized Shield (Case 2)	Orificed Shield (Case 3)
Core midplane	9.62x10 ¹⁵	9.61x10 ¹⁵	9.61x10 ¹⁵
Top of blanket	4.65x10 ¹⁵	4.30x10 ¹⁵	4.30x10 ¹⁵
Bottom of blanket	8.30x10 ¹⁴	6.01x10 ¹⁴	6.01x10 ¹⁴
Top of shield	5.30x10 ¹⁴	4.33x10 ¹⁴	4.28x10 ¹⁴
Top of grid plate	1.81x10 ¹²	1.55x10 ¹²	1.56x10 ¹²

*Based on total source of 4.19x10¹⁹ neutrons/sec.

2.2. DOT R- θ and R-Z Calculations of Demo Stored-Fuel Power and Ex-Vessel Detector Response

On July 27 a meeting was held at ORNL to discuss the in-vessel, ex-vessel concepts for the Demo low-level flux monitors and the complications arising from the in-vessel stored fuel. The stored-fuel fission source in the FTR dominates all ex-vessel shielding problems, and when the reactor is shut down, the inherent source in the FTR stored fuel prevents the use of an ex-vessel low-level flux monitor. In the present Demo design concept, the number of in-vessel storage locations is greatly reduced compared with the FTR, and a combination shield, shadow shield, and window is provided to improve the performance of an ex-vessel low-level detector. An analysis program was planned utilizing DOT R- θ and R-Z calculations backed up by three-dimensional MORSE calculations. The objective is to determine the effectiveness of the stored-fuel shielding both at full power and in the fully shutdown condition.

The primary effort in August was in the preparation of the R- θ calculations. With the exception of the fuel transfer port locations, the core and stored-fuel assemblies have 60° symmetry. The R- θ calculations were prepared in two stages -- the first extending from the core center to core barrel, and the second extending from just inside the core barrel to 12 in. in the concrete beyond the radial cavity. FACT coupling is used between the two cases to couple and change the θ mesh at the coupling plane. The inner R- θ case closely mocks up the outer periphery of the hex core restraint and restraint shield assemblies so as to reveal any peaking due to the irregular radial thickness caused by the hex geometry. The full-power calculations were initiated in August, and the results will be included with those from the complete study in the next progress report.

3. CODE DEVELOPMENT: SPECIFICATIONS FOR DOT IV

During the reporting period preliminary specifications for the next version of the discrete ordinates code DOT (DOT IV) were prepared and distributed for comment. The primary objective of the DOT IV code development is to provide a completely new third-generation program incorporating the established techniques of two-dimensional discrete ordinates calculations and coupled discrete ordinates-Monte Carlo calculations. A major feature of the new program will be the ability to operate with less than the full space problem stored in core and therefore provide a capability for solving larger problems on the smaller machines. Incorporated with this is the ability to change radial or "x" mesh and the angular quadrature at specified Z, θ , or Y planes. A later release will provide similar flexibility at radial or "x" planes, thereby allowing independent mesh and quadrature for subzones. Naturally DOT IV will include and improve on the transport diffusion capability of DOT 2DB, DOT III W, and DOT III. The general specifications of DOT IV are as follows:

1. Reduction of permanent in-memory data storage to the minimum amount compatible with efficient computing. The compact storage of DOT III W will be used as a guide.
2. Given the in-memory storage availability as input, DOT IV will determine the number of axial intervals (radial rows) which can be contained in core. The code will then sweep the mesh performing the necessary input-output to switch from one batch of rows to another.
3. CCCC interfaces will be implemented for cross-section input, flux output, card input, and input-output of binary data. DOT IV will also be compatible with DOT III and DOT III W interfaces including the availability of free form FIDO input.
4. Input-output of binary data will use programmer-controlled direct and sequential access. The CCCC standard read-and-write call statements will be used.

5. DOT IV diffusion theory routines will include all options available in DOT III W, including X-Y, R-Z, R- θ , and hex.

6. DOT IV will include a P_1 theory mode of operation for X-Y, R-Z, and R- θ geometry.

7. DOT IV will allow the specifications of a change in the radial mesh and angular quadrature at an arbitrary number of planes between two axial intervals. A further extension of DOT IV will allow a change in the angular quadrature at a specified radial boundary, thereby allowing quadrature specifications by zone.

8. The variable dimensioning technique in DOT will be revised to facilitate the assignment of arrays to fast and slow memory. The entire code configuration consisting of fast memory, slow memory, and peripheral storage may be adjusted at execution time to best utilize the available computer configuration.

9. All of the important DOT III options, such as P_L truncation by group and diffusion theory selection by group, will be provided by DOT IV.

10. DOT IV will be operational on CDC 7600 and IBM 360/65 or larger computers. The programming will conform to the ANS standard and CCCC draft recommendations and documentation will conform to the ANS draft standard.

4. ANALYSIS OF TSF IRON/POLYETHYLENE SECONDARY GAMMA-RAY EXPERIMENT

Emphasis on the analysis of the TSF iron/polyethylene secondary gamma-ray experiment during this reporting period has resulted in calculations for all the major configurations being completed, and all the results, including those for work performed in earlier months, are presented here. These configurations, identified by the same roman numeral and letter combination used in the outline of the experiment (see ORNL-TM-4170),¹ are described as follows:

- I-A. 10 in. mild steel.
- II-A. 22 in. mild steel.
- II-D. 22 in. mild steel + 4 in. borated polyethylene + 1 in. mild steel.
- III-A. 34 in. mild steel.
- IV-A. FTR spectrum modifier (12 in. stainless steel + 12 in. sodium + 6 in. stainless steel + 12 in. sodium). (Sodium canned in aluminum, 1/4-in.-thick walls.)
- V-A. FTR spectrum modifier + 22 in. mild steel.
- V-B. FTR spectrum modifier + 22 in. mild steel + 2 in. borated polyethylene.
- V-C. FTR spectrum modifier + 22 in. mild steel + 4 in. borated polyethylene.
- V-D. FTR spectrum modifier + 22 in. mild steel + 4 in. borated polyethylene + 1 in. Fe.

The last four configurations are shown in Fig. 4.1.

The composition of the stainless steel, aluminum, sodium, and iron used in the calculations are the ones reported in the January experimental monthly;¹ the boron content of borated polyethylene was 4.7 wt%. The use of the latest available compositions and actual rather than nominal dimensions are two of the improvements made over the previously reported calculations.²

Most of the analysis was performed with the discrete ordinates code DOT in conjunction with the computer programs GRTUNCL and FALSTF. GRTUNCL

Fig. 4.1. Experimental Configurations for TSF Iron/Polyethylene Secondary Gamma-Ray Experiment: Spectrum Modifier Plus (a) 22 in. of Iron; (b) 22 in. of Iron and 2 in. of Borated Polyethylene; (c) 22 in. of Iron and 4 in. of Borated Polyethylene; and (d) 22 in. of Iron, 4 in. of Borated Polyethylene, and 1 in. of Iron.

calculates the uncollided flux and the first-collision source* resulting from a point source, which is used as input in the DOT code to calculate the flux within the shield. FALSTF then calculates the flux at points outside the shield. This latter calculation involves a volume integration in which the last-flight contribution of all sources (including scattering sources) is obtained. The calculations yielded both neutron and gamma-ray transport results.

For configurations II-A and III-A (the iron "benchmark" phase of the experiment), neutron transport results were also obtained with the Monte Carlo code MORSE. The results from the full analysis are discussed below, with the Monte Carlo results presented first.

4.1. Monte Carlo Calculations (220 Groups)

Calculations of the neutron spectra behind approximately 22 and 34 in. of iron, with no spectrum modifier present, were made with MORSE using a 220-group structure with $1/E\sigma_T$ iron weighting based on MAT 4180 mod 1 data. The "impurities" of 0.6 wt% manganese and 0.25 wt% carbon found in carbon steel were included. For uncollided flux and collision site determination, total cross sections based on the same iron data were also incorporated into MORSE for approximately 6000 energy points in the range 10 keV - 10 MeV. The results of these neutron calculations after smearing with the detector resolution are shown in Figs. 4.2-4.4, where they are compared with the measurements.

The measurements made with the two-parameter Benjamin counter system (Figs. 4.2 and 4.3) have been slightly revised from the values reported in an earlier report.³ The reported fluxes per MeV were increased by about 15% and the energies decreased by about 15% in order to match the location of the 24-keV window. This represents a revised energy calibration for the two-parameter system. Comparisons of the calculated and measured spectra shown in Fig. 4.2 behind 22 in. of iron and in Fig. 4.3

*In GRTUNCL this source can include both scattering and fission contributions but in this case included scattering only.

Fig. 4.2. Comparison of MORSE-Calculated and Measured Neutron Spectra
18 in. Behind 22 in. of Iron.

Fig. 4.3. Comparison of MORSE-Calculated and Measured Spectra
 24 in. Behind 34 in. of Iron.

Fig. 4.4. Comparison of MORSE-Calculated and Measured Neutron Spectra Above 0.8 MeV 51 in. Behind 22 in. of Iron.

behind 34 in. of iron indicate good agreement in the region 10 keV - 1 MeV. Below ~ 10 keV the measurements are considerably higher than the calculations; this is believed to be caused by complications arising in the pulse-height distributions that are not taken into account in the unfolding procedure.

The comparisons of the NE-213 spectra 51 in. behind 22 in. of iron shown in Fig. 4.4 indicates that above ~ 1 MeV the agreement is somewhat poorer than in the Benjamin counter region, as was expected. The calculations are low in the region $\sim 1-3$ MeV, due probably to the underestimation of the uncollided component in this energy region. Comparisons of the NE-213 spectra behind 12 in. of iron using a different collimator, performed earlier, have indicated the total cross-section valleys in the region ~ 1 to 3 MeV are deeper than those appearing in the iron evaluations.⁴

A comparison of the measured and calculated Bonner ball counting rates using the 220-group MORSE results, is shown in Table 4.1.

4.2. Discrete Ordinates Calculations (39 Groups)

The DOT calculations, performed for all the configurations outlined above, used 21 neutron groups and 18 gamma-ray groups, P_3 scattering, and an S_8 quadrature. The iron cross sections were based on MAT 4180 mod 1 data and were processed by the AMPX code with a $1/E\Sigma_T$ weighting. For the mild steel, the Σ_T value for pure iron was used; for stainless steel the Σ_T value of stainless steel was used. An attempt to use the stainless steel weighting function for the other elements in the stainless steel was also made; however, at the time AMPX could not successfully handle the run. This is unfortunate, since proper weighting of the chromium and nickel cross sections (taken from ENDF-III) could be important for neutron transport in stainless steel. The cross sections used in these calculations for materials other than iron, chromium and nickel came from the previously existing 21-18 library, most of which is several years old.

Table 4.1. Comparison of MORSE-Calculated and Measured Bonner Ball Count Rates Beyond Iron Configurations

	Count Rate (cpm/W)		
	3-in. Ball	6-in. Ball	10-in. Ball
<u>18 in. Behind 22 in. Iron</u>			
Calculated, scattered ^a	244	1630	572
Calculated, uncollided ^a	7.9	57	23
Calculated, total	252	1687	595
Measured	275	1696	629
Calculated/Measured	0.92	0.99	0.95
Uncertainty in the Calculation ^b	±0.15	±0.12	±0.11
<u>24 in. Behind 34 in. Iron</u>			
Calculated, scattered ^a	68.2	363	110
Calculated, uncollided ^a	1.0	7.1	2.5
Calculated, total	69.2	370	112.5
Measured	69.1	336	103
Calculated/Measured	1.00	1.10	1.09
Uncertainty in the Calculation ^b	±0.12	±0.11	±0.12

^aCorrected for the displacement toward the slab of the center of detection of the Bonner Ball. This displacement is assumed to be 0.3 of the diameter of the Bonner Ball.

^bThe uncertainty in the calculation includes the statistical uncertainty in the Monte Carlo results, an assumed error of 50% in the contribution due to multiple scattering effects between the slabs and the collimator, which was not calculated, estimated errors in the extrapolation of the 22-in. results to include the contributions from neutrons leaking from the iron in the energy range below 5 keV, which were not calculated, and estimated errors arising from the center of detection correction.

The source for the calculations included both neutrons and gamma rays. The neutron source was a measured one, and the gamma-ray source was taken from a previously reported ANISN calculation.

While the true geometry of the experiment is three dimensional, the geometry used in the DOT calculation was limited to two dimensions. In the DOT model, the square slabs were replaced with cylinders having the same cross-sectional area as the slab. The problem of how in the calculation to represent the shielding around the experimental slabs was somewhat difficult since different thicknesses were used on the left, right, top and bottom of the real configurations. In all the calculated results reported, the slabs were assumed to be surrounded first by 8 in. of lithiated paraffin, then by 4 in. of lead, and finally by 4 in. of concrete, which corresponds to the experimental configuration on the side of the slabs nearest the sodium iodide detector. One calculation, the spectrum modifier plus 22 in. Fe, was repeated with the slabs surrounded by 4 in. of lithiated paraffin and 12 in. of concrete. This comparison showed that both neutron and gamma-ray results are somewhat sensitive to the surrounding shielding, and thus the symmetric shielding used in the calculations was not an adequate description.

Neutron Transport Comparisons

Tables 4.2 and 4.3 present comparisons of calculated and measured neutron spectra for the various shield configurations. Also included are the results from the 220-group MORSE calculations described above. So that the comparison between the two calculations could be made, the 220-group results were binned into the 21-group structure. For the energy ranges and shield thicknesses of interest the 21-group calculations are surprisingly good. The results show that the few-group calculation is inadequate for calculating the uncollided flux, which is important only at high energies. Agreement between the 21-group calculation and experiment in the energy range of the hydrogen counter is somewhat worse for 34 in. of iron than for 22 in., with the calculation being lower than the measurement for 34 in. The uncertainties in the NE-213 measurement

behind 22 in. of iron are somewhat large, which probably accounts for some of the disagreement between the calculation and experiment. The disagreement of the few-group calculations in the MeV range seems consistent and may indicate a severe problem for cases in which an iron zone is followed by a thick sodium zone such that the very high energy neutrons are important.

The calculated neutron spectrum (21 groups) behind the spectrum modifier is in good agreement with the unfolded hydrogen counter measurement (Table 4.4). For the energy range from 0.0409 to 1.35 MeV, the ratio of the calculated and measured fluxes is 1.02 compared with 1.38 for a previous calculation,⁵ which did not use the stainless-steel-weighted iron cross sections. The agreement between calculation and experiment is reasonably good for the spectrum modifier plus 22 in. of iron, with the calculated results being 18% higher than the experiment.

A comparison of calculated and measured Bonner ball count rates are given in Table 4.5. An effective "center of detection" was assumed for each ball which was 0.6 times the radius of the ball closer to the slab than the center of the ball. Most of the calculations agreed with the measurement to within 20%. The majority of the calculations were lower than the experiment with the exception of the configurations including the spectrum modifier and 22 in. of iron. For several of these measurements the calculation is higher than the experiment. The comparisons indicate that the 21-group calculations are good up to 22 in. of iron but are $\sim 20\%$ low for 34 in. of iron. The fact that the spectrum modifier was underpredicted and the spectrum modifier plus 22 in. of iron was overpredicted is mysterious. Comparisons of the measured cadmium-covered BF_3 detector count rates agreed with the calculation to within about 30%. The thermal-neutron flux, as measured by subtracting the cadmium-covered detector's count rate from the bare detector's count rate, was compared to the calculated thermal flux, but the agreement was terrible.

Table 4.6 shows a comparison of the calculated flux incident upon the FTR cover (taken from a design calculation) with the calculated

Table 4.2. Comparison of Calculated and Measured Neutron Fluxes
51 and 18 in. Beyond 22 in. of Fe

Calculation ^a	Group No.	Energy Interval (MeV)	Neutron Flux (neutrons·cm ⁻² ·min ⁻¹ ·W ⁻¹)				Ratio:
			Calculated			Measured	Calc. Flux Meas. Flux
			Collided	Uncollided	Total		
NE-213 Detector 51 in. Beyond 22 in. of Fe							
DOT (21)	4	4.49-3.01	2.1190(-2) ^b	2.8607(-4)	2.1476(-2)	4.144(-2)	0.52
	5	3.01-2.02	7.7702(-2)	1.2499(-3)	7.8952(-2)	2.0644(-1)	0.38
	6	2.02-1.35	3.2058(-1)	1.0305(-2)	3.3089(-1)	1.3005(0)	0.25
	7	1.35-0.907	2.2229(0)	5.0612(-2)	2.2735(0)	5.8668(0)	0.38
	4-7	0.907-4.49	2.6424(0)	6.2453(-2)	2.7048(0)	7.4152(0)	0.36
MORSE (220)	4	4.49-3.01	2.002(-2)	4.349(-4)	2.045(-2)	4.144(-2)	0.49
	5	3.01-2.02	1.071(-1)	1.018(-2)	1.173(-1)	2.0644(-1)	0.57
	6	2.02-1.35	5.119(-1)	1.412(-1)	6.531(-1)	1.3005(0)	0.50
	7	1.35-0.907	3.763(0)	1.228(0)	4.991(0)	5.8668(0)	0.85
	4-7	0.907-4.49	4.402(0)	1.380(0)	5.7819(0)	7.4152(0)	0.78
Single-Parameter Hydrogen Counter 18 in. Beyond 22 in. of Fe							
DOT (21)	8	0.907-0.55	1.2034(+2)	2.1159(-1)	1.2055(+2)	1.1102(+2)	1.09
	9	0.55-0.334	1.2604(+2)	2.7856(-2)	1.2607(+2)	1.5180(+2)	0.83
	10	0.334-0.202	2.1756(+2)	3.1719(-1)	2.1788(+2)	2.0729(+2)	1.05
	11	0.202-0.123	1.6859(+2)	1.0347(-1)	1.6869(+2)	2.0307(+2)	0.83
	12	0.123-0.0409	1.5227(+2)	~ 0	1.5227(+2)	1.6311(+2)	0.93
	8-12	0.907-0.0409	7.8480(+2)	6.6011(-1)	7.8546(+2)	8.3629(+2)	0.94
MORSE (220)	8	0.907-0.55	9.831(+1)	6.807(0)	1.051(+2)	1.1102(+2)	0.95
	9	0.55-0.334	1.424(+2)	5.471(0)	1.479(+2)	1.5180(+2)	0.97
	10	0.334-0.202	2.167(+2)	8.241(0)	2.249(+2)	2.0729(+2)	1.09
	11	0.202-0.123	1.718(+2)	8.055(0)	1.798(+2)	2.0307(+2)	0.89
	12	0.123-0.0409	1.276(+2)	7.160(-1)	1.283(+2)	1.6311(+2)	0.79
	8-12	0.907-0.0409	7.5681(+2)	2.9290(+1)	7.8600(+2)	8.3629(+2)	0.94
Double-Parameter Hydrogen Counter 18 in. Beyond 22 in. of Fe							
DOT (21)	12	0.123-0.0409	1.5227(+2)	3.2976(-4)	1.5227(+2)	1.8684(+2)	0.81
	13	0.0409-0.0117	2.7582(+2)	7.6409(-2)	2.7590(+2)	2.3246(+2)	1.19
	14	0.0117-0.00335	3.6435(+1)	~ 0	3.6435(+1)	3.9383(+1)	0.93
	12-14	0.123-0.00335	4.6453(+2)	7.6739(-2)	4.6461(+2)	4.5868(+2)	1.01
MORSE (220)	12	0.123-.0409	1.276(+2)	7.160(-1)	1.283(+2)	1.8684(+2)	0.69
	13	0.0409-0.0117	2.148(+2)	1.120(+1)	2.260(+2)	2.3246(+2)	0.97
	14	0.0117-0.00335	1.944(+1)	~ 0	1.944(+1)	3.9383(+1)	0.49
	12-14	0.123-0.00335	3.6184(+2)	1.1916(+1)	3.7374(+2)	4.5868(+2)	0.81

^aDot calculation performed with 21 neutron groups; MORSE calculation performed with 220 groups results binned into 21-group structure.

^bRead 2.1190x10⁻².

Table 4.3. Comparison of Calculated and Measured Neutron Fluxes 24 in. Beyond 34 in. of Fe

Calculation ^a	Group No.	Energy Interval (MeV)	Neutron Flux (neutrons·cm ⁻² ·min ⁻¹ ·W ⁻¹)				Ratio: <u>Calc. Flux</u> <u>Meas. Flux</u>
			Calculated			Measured	
			Collided	Uncollided	Total		
Single-Parameter Hydrogen Counter 24 in. Beyond 34 in. of Fe							
DOT (21)	7	1.35-0.907	1.5482(-1) ^b	~ 0	1.5482(-1)	1.177(0)	0.13
	8	0.907-0.55	9.6208(0)	~ 0	9.6208(0)	8.822(0)	1.09
	9	0.55-0.334	1.2202(+1)	~ 0	1.2202(+1)	2.031(+1)	0.60
	10	0.334-0.202	3.0863(+1)	~ 0	3.0863(+1)	3.315(+1)	0.93
	11	0.202-0.123	2.9427(+1)	~ 0	2.9427(+1)	4.486(+1)	0.66
	12	0.123-0.0409	2.9084(+1)	~ 0	2.9084(+1)	3.632(+1)	0.80
	7-12	1.35-0.0409	1.1135(+2)	~ 0	1.1135(+2)	1.4464(+2)	0.77
MORSE (220)	7	1.35-0.907	5.5478(-1)	1.0612(-1)	6.6090(-1)	1.177(0)	0.56
	8	0.907-0.55	8.9745(0)	5.8531(-1)	9.5598(0)	8.822(0)	1.08
	9	0.55-0.334	1.9196(+1)	6.5488(-1)	1.9851(+1)	2.031(+1)	0.98
	10	0.334-0.202	4.8911(+1)	1.0423(0)	4.9953(+1)	3.315(+1)	1.51
	11	0.202-0.123	4.8491(+1)	1.3572(0)	4.9848(+1)	4.486(+1)	1.11
	12	0.123-0.0409	3.5077(+1)	7.4136(-2)	3.5151(+1)	3.632(+1)	0.97
	7-12	1.35-0.0409	1.6120(+2)	3.8199(0)	1.6502(+2)	1.4464(+2)	1.14
Double-Parameter Hydrogen Counter 24 in. Beyond 34 in. of Fe							
DOT (21)	12	0.123-0.0409	2.9084(+1)	~ 0	2.9084(+1)	3.7869(+1)	0.77
	13	0.0409-0.0117	7.1775(+1)	~ 0	7.1775(+1)	6.5294(+1)	1.10
	14	0.0117-0.00335	9.4397(0)	~ 0	9.4397(0)	1.1047(+1)	0.85
	12-14	0.123-0.00335	1.1030(+2)		1.1030(+2)	1.1421(+2)	0.97
MORSE (220)	12	0.123-0.0409	3.5077(+1)	7.4136(-2)	3.5151(+1)	3.7869(+1)	0.93
	13	0.0409-0.0117	7.3660(+1)	1.5445(0)	7.5205(+1)	6.5294(+1)	1.15
	14	0.0117-0.00335	1.0590(+1)	5.9748(-10)	1.0590(+1)	1.1047(+1)	0.96
	12-14	0.123-0.00335	1.1933(+2)	1.6186(0)	1.2095(+2)	1.1421(+2)	1.06

^aDot calculation performed with 21 neutron groups; MORSE calculation performed with 220 groups results binned into 21-group structure.

^bRead as 1.5482x10⁻¹.

Table 4.4. Comparison of Calculated and Measured Neutron Fluxes
24 in. Beyond Spectrum Modifier and Iron Configurations
(Single-Parameter Hydrogen Counter)

Group	Energy Interval (MeV)	Neutron Flux (neutrons·cm ⁻² ·min ⁻¹ ·W ⁻¹)		Ratio: Calc./Meas.
		Calculation	Measurement	
Configuration: Spectrum Modifier				
7	1.35-0.907	3.665(-1) ^a	3.81(-1)	0.96
8	0.907-0.55	1.001(0)	8.941(-1)	1.12
9	0.55-0.334	4.321(0)	3.4653(0)	1.25
10	0.334-0.202	6.961(0)	7.4271(0)	0.94
11	0.202-0.123	1.076(+1)	9.8426(0)	1.09
12	0.123-0.0409	1.786(+1)	1.8399(+1)	0.97
7-12	1.35-0.0409	4.1270(+1)	4.0409(+1)	1.02
Configuration: Spectrum Modifier + 22 in. Fe				
8	0.907-0.55	1.2786(-2)	2.1821(-2)	1.71
9	0.55-0.334	4.1682(-2)	5.5748(-2)	1.34
10	0.334-0.202	1.4564(-1)	2.4616(-1)	1.69
11	0.202-0.123	2.9863(-1)	3.3698(-1)	1.13
12	0.123-0.0409	4.3725(-1)	4.4175(-1)	1.01
8-12	0.907-0.0409	9.3599(-1)	1.1025(0)	1.18

^aRead 3.665x10⁻¹.

Table 4.5. Comparison of Calculated and Measured Bonner Ball
Count Rates 24 in. Beyond Various Configurations

Configuration (No. of Groups in Calc.) ^a	Count Rate (cpm/kW)											
	3-in. Ball			6-in. Ball			10-in. Ball			Cd-Covered Ball		
	Calc.	Meas.	Calc. Meas.	Calc.	Meas.	Calc. Meas.	Calc.	Meas.	Calc. Meas.	Calc.	Meas.	Calc. Meas.
10 in. Fe (21)	5.57(5) ^b	6.12(5)	0.91	4.68(6)	4.90(6)	0.95	2.34(6)	2.48(6)	0.94			
22 in. Fe (21)	1.96(5)	2.06(5)	0.95	1.17(6)	1.24(6)	0.94	4.17(5)	4.57(5)	0.91			
22 in. Fe (220)	2.52(5)	2.75(5) ^c	0.92	1.69(6)	1.70(6) ^c	0.99	5.95(5)	6.29(5) ^c	0.95			
34 in. Fe (21)	5.73(4)	6.91(4)	0.83	2.77(5)	3.36(5)	0.83	8.40(4)	1.03(5)	0.82	1.95(3)	3.14(3) ^d	0.62
34 in. Fe (220)	6.92(4)	6.91(4)	1.00	3.70(5)	3.36(5)	1.10	1.13(2)	1.03(5)	1.09			
22 in. Fe + 4 in. Borated Polyethylene + 1 in. Fe (21)	2.62(3)	2.88(3)	0.91	7.47(3)	7.41(3)	1.01	2.61(3)	2.65(3)	0.99			
Spectrum Modifier (21)	5.17(4)	5.87(4)	0.88	1.30(5)	1.57(5)	0.83	3.30(4)	3.94(4)	0.84			
Spectrum Modifier + 22 in. Fe (21)	1.21(3)	9.76(2)	1.24	4.57(3)	3.67(3)	1.25	1.14(3)	9.43(2)	1.21	1.94(1)	2.52(1)	0.77
Spectrum Modifier + 22 in. Fe + 2 in. Borated Polyethylene (21)	2.35(2)	1.91(2)	1.23	3.30(2)	2.74(2)	1.20	6.55(1)	5.76(1)	1.14	1.83(1)	1.41(1)	1.30
Spectrum Modifier + 22 in. Fe + 4 in. Borated Polyethylene (21)	7.18(0)	6.21(0)	1.15	1.00(1)	9.00(0)	1.11	2.17(0)	1.92(0) ^e	1.13	6.24(-1)	7.1(-1)	0.88
Spectrum Modifier + 22 in. Fe + 4 in. Borated Polyethylene + 1 in. Fe (21)	2.79(0)	3.19(0)	0.87	4.76(0)	5.21(0)	0.91	1.16(0)	1.35(0)	0.86	2.25(-1)	3.2(-1)	0.70

^aThe 220-group Monte Carlo calculations have an estimated maximum uncertainty of 10-15%.

^bRead: 5.75×10^5 .

^cMeasurement 18 in. beyond slab.

^dMeasurement 6 in. beyond slab; background not subtracted.

^eValue corrected over that reported in May, 1973 experimental monthly (ORNL-TM-4284, p. 48).

Table 4.6. Comparison of Neutron Flux Calculated To
Be Incident on FTR Cover with Flux Calculated To Be
Emerging from Spectrum Modifier

A: Flux Incident on Bottom of FTR Cover

B: Flux in First Interval of 22-in. Fe Slab Following Spectrum
Modifier

C: Flux in Last Aluminum Interval in Spectrum Modifier

Energy Group	A	B	C	B x 3.866(5)	C x 3.866(5)
				A x 1.778(6) (a)	A x 1.953(6) (a)
1	3.06(-2)	5.33(0)	6.33(0)	37.87	40.95
2	2.99(-1)	1.37(1)	1.67(1)	9.96	11.06
3	1.12(0)	1.95(1)	2.37(1)	3.79	4.19
4	2.67(0)	5.35(1)	6.53(1)	4.36	4.84
5	1.58(1)	1.69(2)	2.01(2)	2.33	2.52
6	6.29(1)	7.03(2)	8.13(2)	2.43	2.56
7	1.68(2)	2.17(3)	2.47(3)	2.81	2.91
8	1.99(3)	8.15(3)	8.77(3)	0.89	0.87
9	3.92(3)	3.47(4)	3.76(4)	1.92	1.90
10	7.15(3)	6.30(4)	6.69(4)	1.92	1.85
11	1.01(4)	1.05(5)	1.14(5)	2.26	2.23
12	2.95(4)	2.39(5)	2.52(5)	1.76	1.69
13	4.43(4)	3.03(5)	3.18(5)	1.49	1.42
14	2.38(4)	1.35(5)	1.49(5)	1.23	1.24
15	3.63(4)	1.98(5)	2.27(5)	1.19	1.24
16	3.42(4)	2.47(5)	2.93(5)	1.57	1.70
17	4.40(4)	1.92(5)	2.10(5)	0.95	0.94
18	5.69(4)	1.35(5)	1.46(5)	0.52	0.51
19	5.66(4)	7.76(4)	8.39(4)	0.30	0.29
20	3.40(4)	3.28(4)	3.66(4)	0.21	0.21
21	3.64(3)	5.06(3)	6.64(3)	0.30	0.36
Total	3.866(5)	1.778(6)	1.953(6)		

^aNumbers in fractions are total fluxes from bottom line of table.

spectrum on the center line between the spectrum modifier and the 22 in. of iron. This comparison is somewhat poorer than similar comparisons between ANISN calculations made when the spectrum modifier was being designed. The stainless-steel weighting of the iron cross sections may account for some of the change. The importance of this tilt toward higher energy neutrons in the experiment cannot be assessed until sensitivity calculations are performed.

Secondary Gamma-Ray Comparisons

In previously reported gamma-ray calculations, the results were multiplied by an attenuation factor to account for the 12 in. of lithium hydride between the experiment and the gamma-ray detector. However, there is an additional 2 in. of lithium hydride inside the collimator which was not included in the attenuation factor. The effect of this error is to reduce previously reported gamma-ray calculations by 10% at 8 MeV, 25% at 1 MeV, and 50% at 0.15 MeV. This error has been corrected in the present calculations.

Since all the shielding around the slabs was not included in the DOT geometry, using this geometry to calculate the contribution of gamma rays leaving the right boundary of the DOT calculation to the flux at the gamma-ray detector would overestimate this contribution. This problem was corrected by adding an option to FALSTF, which calculates only the contribution from gamma rays emerging from the bottom boundary of the DOT geometry (i.e., gamma rays from the sides of the configurations were not included).

The calculations include core gamma rays as a part of the point source. These core gamma rays are most important for thin slabs such as 10 in. of iron. However, since the uncollided core gamma rays are all traveling in the same direction and the scattering of gamma rays is extremely anisotropic, a low-order Legendre expansion of the scattering cross section (and thus a low-order spherical harmonic expansion of the scattering source) is not adequate to calculate this problem. A thorough

resolution of this problem was not attempted because the difficulty only occurs for thin slabs that are only of secondary interest compared to the thick slabs. The approximate fix-up used was to calculate the single-scattering contribution of the core gamma rays analytically and to use "P₃-corrected P₂" cross sections to calculate the scattering source in FALSTF. This procedure was used in the calculation for the 10-in. iron slab only.

Table 4.7 gives comparisons of calculated and measured gamma-ray spectra for the various shield configurations. In every case the latest available measurement is used. Each table has two sets of results; one includes all gamma rays from the exit face of the configuration, while the other includes only those gamma rays from the shield slabs themselves (i.e., no gamma rays from the lithiated paraffin, lead, or concrete were considered). Because the three-dimensional geometry of lithiated paraffin, lead, and concrete is not represented explicitly, the gamma rays from these materials are not calculated correctly. Also, in the DOT geometry the outside surface of these materials is flush with the surface of the slab, while, in the real geometry, some materials extend farther than others and thus have a shadowing effect. By placing lead over the lithiated paraffin it has been experimentally determined that none of the measured gamma rays come from the lithiated paraffin. This is a direct contradiction of the calculated results which indicate a lithiated paraffin contribution. It therefore appears that the "slab gammas only" condition is the best model; however, both results are given.

For the configurations without the spectrum modifier, the total gamma-ray flux calculated is about 20% lower than the measured flux in each case. Note that the peak in the 7- to 8-MeV bin is about right for all of these configurations except for the 10-in. iron slab. This latter case was one of the very first measurements made in the experimental series and attempts were made to improve the results as the measurements proceeded. At high gamma-ray energies, where there is no core gamma-ray contribution, there is no apparent reason why the agreement at 10 in. should be worse than the agreement at 22 and 34 in.

Table 4.7 . Comparison of Calculated^a and Measured Gamma-Ray Fluxes
at an Angle of 45 deg Beyond Various Configurations

Group No.	Energy Interval (MeV)	Gamma-Ray Flux (γ 's $\cdot$ cm ⁻² $\cdot$ kW ⁻¹ $\cdot$ sec ⁻¹)			Ratio: $\frac{\text{Calc.Flux}}{\text{Meas.Flux}}$	
		Calculated		Measured	All γ 's in Calc	Config γ 's Only in Calc.
		All γ 's	Configuration γ 's Only			
Configuration: 10 in. Fe						
1	10-8	2.592(-2) ^b	2.590(-2)	3.6496(-2)	0.71	0.71
2	8-7	1.264(-1)	1.262(-1)	1.8402(-1)	0.69	0.69
3	7-6	4.452(-2)	4.292(-2)	7.9814(-2)	0.56	0.54
4	6-5	4.806(-2)	4.744(-2)	7.6001(-2)	0.63	0.62
5	5-4	6.254(-2)	6.156(-2)	7.9103(-2)	0.79	0.78
6	4-3.5	3.781(-2)	3.726(-2)	4.5725(-2)	0.83	0.81
7	3.5-3	5.471(-2)	5.436(-2)	5.4314(-2)	1.01	1.00
8	3-2.5	7.312(-2)	7.268(-2)	6.4956(-2)	1.12	1.12
9	2.5-2	9.580(-2)	9.270(-2)	1.0373(-1)	0.92	0.89
10	2-1.6	1.070(-1)	1.059(-1)	1.0800(-1)	0.99	0.98
11	1.6-1.2	1.636(-1)	1.626(-1)	1.5110(-1)	1.08	1.08
12	1.2-0.9	1.505(-1)	1.498(-1)	1.1218(-1)	1.34	1.33
1-12	10-0.9	9.900(-1)	9.793(-1)	1.095(0)	0.90	0.89
Configuration: 22 in. Fe						
1	10-8	9.5317(-3)	9.5120(-3)	9.0462(-3)	1.05	1.05
2	8-7	5.4129(-2)	5.3858(-2)	5.4999(-2)	0.98	0.98
3	7-6	1.6192(-2)	1.5753(-2)	2.7072(-2)	0.60	0.58
4	6-5	1.6090(-2)	1.5654(-2)	2.1960(-2)	0.73	0.71
5	5-4	1.9990(-2)	1.9426(-2)	2.3906(-2)	0.84	0.81
6	4-3.5	8.0764(-3)	7.7556(-3)	1.2008(-2)	0.67	0.65
7	3.5-3	1.2178(-2)	1.1840(-2)	1.2991(-2)	0.94	0.91
8	3-2.5	1.1024(-2)	1.0626(-2)	1.2476(-2)	0.88	0.85
9	2.5-2	1.5078(-2)	1.0028(-2)	1.7037(-2)	0.89	0.59
10	2-1.6	1.0625(-2)	9.6626(-3)	1.6578(-2)	0.64	0.58
11	1.6-1.2	8.8283(-3)	7.8696(-3)	1.2882(-2)	0.69	0.61
12	1.2-0.9	6.0973(-3)	5.4072(-3)	8.8242(-3)	0.69	0.61
13	0.9-0.6	8.3371(-3)	7.6385(-3)	1.0686(-2)	0.78	0.71
14	0.6-0.4	7.8259(-3)	7.2302(-3)	1.2852(-2)	0.61	0.56
15	0.4-0.21	4.7135(-3)	4.1867(-3)	7.6231(-3)	0.62	0.55
1-15	10-0.21	2.087(-1)	1.964(-1)	2.609(-1)	0.80	0.75
Configuration: 34 in. Fe						
1	10-8	3.4451(-3)	3.4327(-3)	3.4679(-3)	0.99	0.99
2	8-7	2.2257(-2)	2.2099(-2)	2.1672(-2)	1.03	1.02
3	7-6	6.2292(-3)	6.0155(-3)	1.0587(-2)	0.59	0.57
4	6-5	6.2593(-3)	6.0050(-3)	8.3277(-3)	0.75	0.72
5	5-4	7.6465(-3)	7.3208(-3)	9.1202(-3)	0.84	0.80
6	4-3.5	3.0090(-3)	2.8218(-3)	4.7015(-3)	0.64	0.60
7	3.5-3	4.6301(-3)	4.4239(-3)	5.0821(-3)	0.91	0.87
8	3-2.5	4.1817(-3)	3.9382(-3)	4.7304(-3)	0.88	0.83
9	2.5-2	6.2365(-3)	3.6238(-3)	6.6408(-3)	0.94	0.55
10	2-1.6	4.1442(-3)	3.5802(-3)	5.9186(-3)	0.70	0.60
11	1.6-1.2	3.3912(-3)	2.8275(-3)	4.5156(-3)	0.75	0.63
12	1.2-0.9	2.3429(-3)	1.9365(-3)	3.1095(-3)	0.75	0.62
13	0.9-0.6	2.6060(-3)	2.2004(-3)	3.0787(-3)	0.85	0.71
14	0.6-0.4	2.7923(-3)	2.4529(-3)	4.4012(-3)	0.63	0.56
15	0.4-0.21	1.6566(-3)	1.3642(-3)	2.5088(-3)	0.66	0.54
1-15	10-0.21	8.083(-2)	7.404(-2)	9.786(-2)	0.83	0.76

^aDOT Calculation with 21 neutron groups and 18 gamma-ray groups.

^bRead: 2.592×10^{-2}

Table 4.7. (Contd.)

Group No.	Energy Interval (MeV)	Gamma-Ray Flux (γ 's $\cdot$ cm ⁻² $\cdot$ kW ⁻¹ $\cdot$ sec ⁻¹)			Ratio:	<u>Calc. Flux</u>
		Calculated		Measured		<u>Meas. Flux</u>
		All γ 's	Configuration γ 's Only		All γ 's in Calc.	Config γ 's Only in Calc.
Configuration: 22 in. Fe + 4 in. Borated Polyethylene + 1 in. Fe						
1	10-8	6.8186(-3)	6.8029(-3)	9.6866(-3)	0.70	0.70
2	8-7	5.0874(-2)	5.0707(-2)	5.5862(-2)	0.91	0.91
3	7-6	1.4412(-2)	1.4154(-2)	2.0518(-2)	0.70	0.69
4	6-5	1.5000(-2)	1.4684(-2)	1.8576(-2)	0.81	0.79
5	5-4	1.7401(-2)	1.6996(-2)	1.9444(-2)	0.89	0.87
6	4-3.5	7.0348(-3)	6.7978(-3)	9.9076(-3)	0.71	0.69
7	3.5-3	1.0022(-2)	9.7562(-3)	1.2102(-2)	0.83	0.81
8	3-2.5	9.3119(-3)	8.9989(-3)	1.2521(-2)	0.74	0.72
9	2.5-2	1.7513(-2)	1.5604(-2)	2.1828(-2)	0.80	0.71
10	2-1.6	9.9402(-3)	9.3052(-3)	1.1210(-2)	0.89	0.83
11	1.6-1.2	8.7599(-3)	8.1065(-3)	1.0997(-2)	0.80	0.74
12	1.2-0.9	6.0889(-3)	5.6048(-3)	8.2182(-3)	0.74	0.68
1-12	10-0.9	1.732(-1)	1.675(-1)	2.109(-1)	0.82	0.79
Configuration: Spectrum Modifier (12 in. SS + 12 in. Na + 6 in. SS + 12 in. Na)						
1	10-8	1.184(-2)	1.180(-2)	1.394(-2)	0.85	0.85
2	8-7	3.077(-2)	3.010(-2)	2.753(-2)	1.12	1.09
3	7-6	2.059(-2)	1.651(-2)	2.124(-2)	0.97	0.78
4	6-5	1.370(-2)	1.212(-2)	1.620(-2)	0.85	0.75
5	5-4	1.512(-2)	1.253(-2)	1.760(-2)	0.86	0.71
6	4-3.5	1.185(-2)	1.025(-2)	1.258(-2)	0.94	0.82
7	3.5-3	8.423(-3)	7.735(-3)	1.265(-2)	0.67	0.61
8	3-2.5	1.232(-2)	1.131(-2)	1.495(-2)	0.82	0.76
9	2.5-2	2.858(-2)	9.237(-3)	1.725(-2)	1.66	0.54
10	2-1.6	1.257(-2)	7.615(-3)	1.101(-2)	1.14	0.69
11	1.6-1.2	1.070(-2)	6.878(-3)	9.150(-3)	1.17	0.75
12	1.2-0.9	6.786(-3)	4.874(-3)	7.393(-3)	0.92	0.66
1-12	10-0.9	1.8325(-1)	1.4096(-1)	1.8149(-1)	1.01	0.78
Configuration: Spectrum Modifier + 22 in. Fe						
1	10-8	9.7144(-5)	9.5522(-5)	5.9252(-5)	1.64	1.61
2	8-7	6.8210(-4)	6.6988(-4)	4.3006(-4)	1.59	1.56
3	7-6	1.9417(-4)	1.7279(-4)	1.9294(-4)	1.01	0.90
4	6-5	2.0536(-4)	1.7788(-4)	1.6183(-4)	1.27	1.10
5	5-4	2.4352(-4)	2.0645(-4)	1.7620(-4)	1.38	1.17
6	4-3.5	1.0002(-4)	7.7141(-5)	9.0566(-5)	1.10	0.85
7	3.5-3	1.4690(-4)	1.2005(-4)	1.0024(-4)	1.47	1.20
8	3-2.5	1.3842(-4)	1.0579(-4)	9.0071(-5)	1.54	1.17
9	2.5-2	2.2290(-4)	9.7173(-5)	1.0673(-4)	2.09	0.91
10	2-1.6	1.5582(-4)	9.7954(-5)	9.5908(-5)	1.62	1.02
11	1.6-1.2	1.4155(-4)	7.6947(-5)	6.4015(-5)	2.21	1.20
1-11	10-1.2	2.328(-3)	1.897(-3)	1.568(-3)	1.48	1.21

Table 4.7. (Contd.)

Group No.	Energy Interval (MeV)	Gamma-Ray Flux (γ 's $\cdot$ cm ⁻² $\cdot$ kW ⁻¹ $\cdot$ sec ⁻¹)			Ratio:	$\frac{\text{Calc. Flux}}{\text{Meas. Flux}}$
		Calculated		Measured		
		All γ 's	Configuration γ 's Only		All γ 's in Calc.	Config γ 's Only in Calc.
Configuration: Spectrum Modifier + 22 in. Fe + 4 in. Borated Polyethylene						
1	10-8	1.0458(-4)	1.0440(-4)	7.3105(-5)	1.43	1.43
2	8-7	8.1116(-4)	8.0827(-4)	6.1540(-4)	1.32	1.31
3	7-6	2.0592(-4)	2.0127(-4)	2.2531(-4)	0.91	0.89
4	6-5	2.1599(-4)	2.0952(-4)	2.0572(-4)	1.05	1.02
5	5-4	2.4849(-4)	2.3893(-4)	1.9909(-4)	1.25	1.20
6	4-3.5	9.3975(-5)	8.7692(-5)	9.6237(-5)	0.98	0.91
7	3.5-3	1.4543(-4)	1.3765(-4)	1.1344(-4)	1.28	1.21
8	3-2.5	1.3182(-4)	1.2181(-4)	9.1325(-5)	1.44	1.33
9	2.5-2	2.6392(-4)	2.2314(-4)	1.8182(-4)	1.45	1.23
10	2-1.6	1.4213(-4)	1.2239(-4)	9.0861(-5)	1.56	1.35
11	1.6-1.2	1.2132(-4)	9.7655(-5)	5.2977(-5)	2.29	1.84
12	1.2-0.9	8.7458(-5)	6.7440(-5)	2.3160(-5) ^c	3.78	2.91
13	0.9-0.6	9.3652(-5)	7.1867(-5)	4.8844(-6) ^c	19.17	14.71
14	0.6-0.4	1.5788(-3)	1.5541(-3)	9.9771(-4)	1.58	1.56
15	0.4-0.21	5.6304(-4)	5.3988(-4)	3.3302(-4)	1.69	1.62
1-15	10-0.21	4.808(-3)	4.586(-3)	3.304(-3)	1.46	1.39
Configuration: Spectrum Modifier + 22 in. Fe + 4 in. Borated Polyethylene + 1 in. Fe						
1	10-8	4.5129(-5)	4.4914(-5)	3.9218(-5)	1.15	1.15
2	8-7	3.5160(-4)	3.4910(-4)	3.2597(-4)	1.08	1.07
3	7-6	1.0102(-4)	9.6700(-5)	1.2618(-4)	0.80	0.77
4	6-5	1.0752(-4)	1.0152(-4)	1.1562(-4)	0.93	0.88
5	5-4	1.2412(-4)	1.1541(-4)	1.1618(-4)	1.07	0.99
6	4-3.5	5.1923(-5)	4.6249(-5)	6.1322(-5)	0.85	0.75
7	3.5-3	7.2092(-5)	6.5112(-5)	7.1192(-5)	1.01	0.91
8	3-2.5	6.9115(-5)	6.0260(-5)	6.0384(-5)	1.14	1.00
9	2.5-2	1.1978(-4)	9.1619(-5)	9.9771(-5)	1.20	0.92
10	2-1.6	7.6401(-5)	6.0018(-5)	6.0039(-5)	1.27	1.00
11	1.6-1.2	7.2704(-5)	5.3354(-5)	5.5851(-5)	1.30	0.96
12	1.2-0.9	5.3655(-5)	3.7556(-5)	3.2636(-5) ^c	1.64	1.15
13	0.9-0.6	5.4609(-5)	3.7401(-5)	7.9127(-6) ^c	6.90	4.73
14	0.6-0.4	2.5604(-4)	2.3539(-4)	1.5597(-4) ^c	1.64	1.51
15	0.4-0.21	1.6772(-4)	1.4842(-4)	1.1929(-4) ^c	1.41	1.24
1-15	10-0.21	1.723(-3)	1.543(-3)	1.448(-3)	1.19	1.07

^c Large uncertainty in the measurement.

For the spectrum modifier only, the agreement between the calculation and measurement is fairly good. For the three configurations that include the spectrum modifier and 22 in. of iron, the calculation is higher than the measurement in each case. It is difficult to say how much the calculational results would change if various improvements such as a finer energy group structure were made. Two measurements were made for the spectrum modifier plus 22 in. of iron and also for the spectrum modifier plus 22 in. of iron and 4 in. of borated polyethylene. A comparison of these measurements is shown in Table 4.8. What is considered to be an improved background measurement was subtracted from the later measurement; however, the older measurements agree with the calculation somewhat better than the remeasurement. The configuration which most closely resembles the actual design problem for the FFTF is the spectrum modifier plus 22 in. of iron, 4 in. of borated polyethylene, and 1 in. of iron. No remeasurement was made for this configuration, and it must be assumed that this case would be consistent with the other spectrum-modified cases if a remeasurement were made.

Table 4.9 shows the effect of varying the amount of boron in the borated polyethylene for the configuration consisting of 22 in. of iron, 4 in. of borated polyethylene, and 1 in. of iron. A decrease from 6% to 2% boron gives essentially the same neutron dose and a 20-25% increase in the gamma-ray dose.

The analysis for the iron/polyethylene experiment is considered complete for the first phase. Follow-on effort will consist of sensitivity analyses to answer the questions raised here and provide additional insight into the consequences of the experiment.

Table 4.8, Comparison of Latest Gamma-Ray Measurements
Behind Spectrum Modifier and 22 in. Fe with
Previously Reported Values

Group	Energy Interval (MeV)	Ratio: New Measurement/Old Measurement	
		Spectrum Modifier + 22 in. Fe	Spectrum Modifier + 22 in. Fe + 4 in. Borated Polyethylene
1	10-8	0.820	0.944
2	8-7	0.779	0.879
3	7-6	0.784	1.037
4	6-5	0.730	0.853
5	5-4	0.763	0.901
6	4-3.5	0.753	0.961
7	3.5-3	0.739	0.842
8	3-2.5	0.707	0.861
9	2.5-2	0.596	0.923
10	2-1.6	0.595	0.884
11	1.6-1.2		
12	1.2-0.9		
13	0.9-0.6		
14	0.6-0.4		0.839
15	0.4-0.21		0.678
Avg		0.727	0.867

^aSee January and May 1973, experimental monthlies for old and new measurements, respectively (ORNL-TM-4170 and ORNL-TM-4284).

Table 4.9. ANISN Calculation of the Effect of the Borated Polyethylene Composition on the Neutron and Gamma-Ray Responses Beyond 22 in. Fe + 4 in. Borated Polyethylene + 1 in. Fe

Boron Content (wt%)	Mixture Density (g/cm ³)	Relative Response (Arbitrary Units)		
		Neutron Dose	Gamma-Ray Dose	3-in. Bonner Ball Count Rate
1.00	0.9245	3.501(-2) ^a	4.344(-2)	5.355(-1)
2.00	0.9302	3.509(-2)	3.476(-2)	5.168(-1)
3.00	0.9358	3.526(-2)	3.118(-2)	5.009(-1)
3.96	0.9414	3.546(-2)	2.915(-2)	4.874(-1)
4.93	0.9470	3.568(-2)	2.774(-2)	4.755(-1)
5.89	0.9527	3.593(-2)	2.670(-2)	4.649(-1)

^aRead 3.501×10^{-2} .

Section 4 References

1. C. E. Clifford, F. J. Muckenthaler, and P. N. Stevens, "Fast Reactor Experimental Shielding Monthly Progress Report for January, 1973," ORNL-TM-4170 (March 1973).
2. F. R. Mynatt et al., "Fast Reactor Experimental Shielding Monthly Progress Report for January and February, 1973," ORNL-TM-4231 (May 1973).
3. C. E. Clifford et al., "Fast Reactor Experimental Shielding Monthly Progress Report for May, 1973," ORNL-TM-4284 (July 1973).
4. F. R. Mynatt et al., "Fast Reactor Analytical Shielding Monthly Progress Report for November and December 1972," ORNL-TM-4164 (March 1973).
5. F. R. Mynatt and R. E. Maerker, "Fast Reactor Analytical Shielding Monthly Progress Report for January and February 1973," ORNL-TM-4231 (May 1973).

5. ANALYSIS OF TSF WESTINGHOUSE DEMONSTRATION PLANT EXPERIMENTS

The analysis of the TSF Westinghouse Demonstration Plant Experiments was begun last year^{1,2} but was suspended, along with the experiments themselves, to allow other work to be performed. During this reporting period the analysis has been resumed and comparisons between calculated values and the measured data are now available for the spectrum modifier and for the configuration that has the first 5-in. radial blanket following the spectrum modifier (identified as Items II-A and IV-A in the outline of the experiment³). The experimental arrangement and composition of the spectrum modifier is shown in Fig. 5.1. The composition of the radial blanket, which consists of a region of closely packed UO_2 fuel pins in a matrix of sodium, is given in detail in an experimental monthly report.³

The earlier calculations covered the spectrum modifier only and were performed with the Monte Carlo code MORSE. The calculations presented here have been performed with the one-dimensional discrete ordinates code ANISN and with the two-dimensional discrete ordinates code DOT in combination with the GRTUNCL and FALSTF programs. As is pointed out in Section 4, GRTUNCL calculates the uncollided flux and the first-collision source to be input into DOT. In these calculations, the first-collision source includes fission contributions as well as scattering contributions since the depleted uranium used in the radial blanket contains ^{235}U and ^{238}U . FALSTF is used to calculate the flux at points outside the shield.

The cross-section set used in the calculations was the same coupled set described in Section 4. It contains 21 neutron groups and 18 gamma-ray groups, with the neutron and gamma-ray-production cross sections weighted $1/E\sigma_T$.

5.1. Spectrum Modifier

Neutron Transport Calculations

Bonner ball responses calculated with DOT for various distances behind the spectrum modifier are compared with the measured responses in Table 5.1.

ORNL-DWG 72-12053

SLABS:

- a* STAINLESS STEEL 5 ft x 5 ft (2.012 in. + 2.021 in.)
- b* Na 5 ft x 5 ft
- c* BORAL 5 ft x 5 ft
- d* B₄C 50 in. W x 52 in. H

Fig. 5.1. Horizontal Cross Section Through Westinghouse Spectrum Modifier (Item II.A). Lithium-paraffin shield was on sides and top of configuration; entire assembly set on concrete blocks.

The calculated values are in somewhat poorer agreement with the measurements than were those obtained from the 100-group MORSE calculations last year.¹ In particular, the DOT calculations are higher for the 3-in. Bonner ball. This is thought not to be due to the coarser group structure, however, but rather to a too-coarse spatial mesh in the boral, resulting in an overprediction of the low-energy fluxes.

Comparisons of the calculated and measured fluxes for the hydrogen counter are shown in Table 5.2 and for the NE-213 spectrometer in Table 5.3. For either detector, the agreement between the measurements and calculations is very similar to that obtained between the measurements and the 100-group MORSE calculations.

Gamma-Ray Transport Calculations

In the experiment, gamma-ray measurements beyond the spectrum modifier were made with an NaI spectrometer positioned at a point 28 ft 8 in. behind the configuration and 30 ft 8 in. off the center line. A 12-in.-thick slab of LiH was located ahead of the spectrometer shield in line with the collimator, and an additional 2 in. of LiH was located in the spectrometer collimator itself. In the analysis, the gamma-ray attenuation in the LiH slabs, as well as in the intervening air, was considered.

Preliminary calculations of the gamma-ray fluxes at the detector position showed that the contribution from the lithiated paraffin around the modifier was negligible. Therefore the gamma-ray contributions calculated were limited to the secondary gamma rays produced within the modifier and subsequently transported through it and the core gamma rays transported through the spectrum modifier. These calculations differ from the earlier calculations² in that nonthermal production cross sections used in the first calculations were weighted $1/E$ rather than $1/E\sigma_T$ and the core gamma rays were not considered. In the absence of an absolute measured core gamma-ray spectrum, the spectrum used in these calculations was taken from an ANISN calculation.⁴

The analysis performed to date has consisted of both ANISN and DOT calculations of the total gamma-ray flux at the detector location in each energy interval and ANISN calculations of the contributions of the secondary gamma rays alone. To deduce the contributions by the secondary gamma rays in the DOT calculations, it was assumed that the fractional contributions calculated by ANISN were applicable to DOT, the difference between the secondary fluxes and the total fluxes then being attributed to the core gamma rays. The ANISN-calculated secondary contributions normalized to DOT are shown in Table 5.4.

It was expected that with the relatively thin spectrum modifier the core gamma rays would contribute significantly to the flux at the detector location but that the contributions would be limited to energies below about 2 MeV, that being the approximate upper energy limit for gamma rays singly scattered in the slabs at the detector angle of about 45 deg. However, as is shown in the second and third columns of Table 5.5, subtracting the secondary fluxes from the total fluxes left fluxes for the core gamma rays that were nonnegligible above 2 MeV. This indicates that the delta function in energy and angle that is important for single and low-order multiple scattering is poorly represented by the P_3 expansion used in the calculations and leads to finite contributions at energies that are physically impossible. An adequate way of handling this anomaly, without using higher P_n expansion of the Klein-Nishina formula, is to ignore the singly scattered core gamma-ray contribution calculated by S_n and substitute a correct analytical calculation. In addition, using a "P₃-corrected P₂" expansion of the scattering matrices in FALSTF reduces the core gamma-ray contribution in physically impossible energy regions to much lower levels, even though the fluxes in the slab are still calculated by DOT with the unaltered P_3 expansion. The effect of these various approximations on the scattered core gamma-ray spectrum at the detector is presented in Table 5.5. It is to be observed that in none of the cases is the calculated core gamma-ray contribution negligible above ~ 3 MeV, as it probably should be.

Table 5.1. Comparison of DOT-Calculated and Measured Bonner
Ball Count Rates on Center Line Behind the
Westinghouse Spectrum Modifier

	Count Rate (cpm/W)		
	3-in. Ball	6-in. Ball	10-in. Ball
<u>12 in. Behind</u>			
Calculated, scattered	1491	11010	6589
Calculated, uncollided	30	742	799
Calculated, total	1521	11752	7388
Measured	1062	9344	5887
Calculated/Measured	1.43	1.26	1.26
<u>18 in. Behind</u>			
Calculated, scattered	939	6681	3892
Calculated, uncollided	26	643	691
Calculated, total	965	7324	4583
Measured	708	6307	4076
Calculated/Measured	1.36	1.16	1.12
<u>24 in. Behind</u>			
Calculated, scattered	639	4437	2538
Calculated, uncollided	23	563	604
Calculated, total	662	5000	3142
Measured	527	4591	2949
Calculated/Measured	1.26	1.09	1.07

Table 5.2. Comparison of DOT-Calculated and Measured Neutron
Fluxes for Hydrogen Counter on Center Line 12 in.
Behind Westinghouse Spectrum Modifier

Energy Interval (MeV)	Neutron Flux (neutrons·cm ⁻² ·min ⁻¹ ·W ⁻¹)				Ratio: Calc. Flux Meas. Flux
	Calculated			Measured	
	Scattered	Uncollided	Total		
0.907-1.35	642	78	720	679	1.06
0.550-0.907	812	41	854	773	1.10
0.334-0.550	959	26	984	861	1.14
0.202-0.334	774	24	799	776	1.03
0.123-0.202	693	9	702	947	0.74
0.0409-0.123	754	7	761	1185*	0.64*
0.0409-1.35	4634	185	4819	5221	0.92

*Gamma-ray contamination is probably present.

Table 5.3. Comparison of DOT-Calculated and Measured Neutron Fluxes for NE-213 Spectrometer at Various Distances Behind Westinghouse Spectrum Modifier

Energy Interval (MeV)	Neutron Flux (neutrons·cm ⁻² ·min ⁻¹ ·W ⁻¹)		Ratio:
	Calculated	Measured	Calc. Flux Meas. Flux
<u>10 ft Behind Configuration; 76 in. South of Center Line</u>			
6.7-10	0.779	0.629	1.24
4.49-6.7	2.21	2.21	1.00
3.01-4.49	3.92	4.05	0.97
2.02-3.01	6.57	6.49	1.01
1.35-2.02	8.89	8.14	1.09
0.907-1.35	9.45	8.56	1.10
0.907-10	31.82	30.08	1.06
<u>10 ft Behind Configuration; 10 ft South of Center Line</u>			
6.7-10	0.286	0.187	1.53
4.49-6.7	0.889	0.789	1.13
3.01-4.49	1.78	1.72	1.03
2.02-3.01	3.27	3.07	1.07
1.35-2.02	4.77	4.15	1.15
0.907-1.35	5.20	4.43	1.17
0.907-10	16.20	14.35	1.13
<u>30 ft Behind Configuration; 30 ft South of Center Line</u>			
6.7-10	0.030	0.023	1.31
4.49-6.7	0.095	0.101	0.94
3.01-4.49	0.193	0.243	0.80
2.02-3.01	0.359	0.442	0.81
1.35-2.02	0.526	0.604	0.87
0.907-1.35	0.574	0.670	0.86
0.907-10	1.777	2.083	0.85
<u>30 ft Behind Configuration; 9 ft South of Centerline</u>			
6.7-10	0.190	0.204	0.93
4.49-6.7	0.515	0.568	0.91
3.01-4.49	0.841	0.864	0.97
2.02-3.01	1.30	1.23	1.05
1.35-2.02	1.64	1.42	1.16
0.907-1.35	1.69	1.40	1.21
0.907-10	6.18	5.69	1.09

Table 5.4. Secondary Gamma-Ray Contributions at the Detector
 Location Behind the Westinghouse Spectrum Modifier
 (Calculated with ANISN and Normalized to DOT)

Energy Interval (MeV)	Gamma-Ray Flux (photons·cm ⁻² ·sec ⁻¹ ·kW ⁻¹)	Fraction of Secondary Gamma-Ray Flux Due to Thermal-Neutron Captures
8-10	0.291	0.74
7-8	0.665	0.68
6-7	0.270	0.64
5-6	0.265	0.63
4-5	0.297	0.57
3.5-4	0.171	0.44
3-3.5	0.193	0.46
2.5-3	0.258	0.33
2-2.5	0.292	0.28
1.6-2	0.276	0.26
1.2-1.6	0.272	0.24
0.9-1.2	0.173	0.27
0.6-0.9	0.227	0.20
0.4-0.6	0.470	0.11
0.21-0.4	0.239	0.14

Table 5.5. Approximate Core Gamma-Ray Contributions at the Detector Location Behind the Westinghouse Spectrum Modifier Calculated with Various Scattering Approximations^a

Energy Interval (MeV)	Gamma-Ray Flux (photons·cm ⁻² ·sec ⁻¹ ·kW ⁻¹)			
	DOT + FALSTF			
	ANISN P ₃ Throughout	P ₃ Throughout	P ₃ Through- out; Single- Scattering Correction	P ₃ in DOT; P ₃ -Corrected P ₂ in FALSTF; Single- Scattering Correction
8-10	0.0027	0.0412	0.0008	0.0140
7-8	0.0484	0.131	0.0133	0.0444
6-7	0.122	0.149	0.0318	0.0221
5-6	0.218	0.244	0.0579	0.0332
4-5	0.530	0.542	0.135	0.0578
3.5-4	0.497	0.493	0.128	0.0849
3-3.5	0.787	0.770	0.223	0.141
2.5-3	1.24	1.21	0.409	0.256
2-2.5	2.15	2.08	0.820	0.468
1.6-2	2.57	2.49	1.20	0.756
1.2-1.6	3.36	3.30	2.42	1.73
0.9-1.2	2.86	2.84	3.54	2.98
0.6-0.9	2.95	2.97	3.35	2.72
0.4-0.6	1.76	1.79	1.65	1.29
0.21-0.4	1.15	1.15	1.15	1.08

^aDeduced by assuming that the fraction of the total flux due to secondary gamma rays calculated in ANISN is applicable to DOT calculations (see text).

The comparison of the calculated gamma-ray fluxes with the NaI detector measurements is shown in Table 5.6. Even with the deficiencies in the calculation pointed out above, the agreement is good. Since it is planned to repeat these calculations when a new cross-section library becomes available, at which time a measurement of the core gamma-ray spectrum hopefully will have been completed, no further conclusions on the adequacy of the calculations will be drawn at this time.

5.2. Spectrum Modifier Plus 5-in. Radial Blanket

Neutron Transport Calculations

Calculations of the Bonner ball responses beyond the Westinghouse spectrum modifier plus the radial blanket are compared with the measurements in Table 5.7. The agreement is considered to be adequate, although the calculated 10-in. Bonner ball count rates appear to be overestimated by about 20%. The fluxes calculated for the hydrogen counter are also high by about the same percentage (see Table 5.8). On the other hand, the comparisons of the NE-213 spectrometer (Table 5.9) show excellent agreement between the calculations and the measurements.

Gamma-Ray Transport Calculations

The gamma-ray measurements beyond the spectrum modifier and radial blanket were made with the NaI spectrometer at a point 28 ft 3 in. behind the configuration and 30 ft 8 in. off the centerline. As before, the calculations considered the effect of lithium hydride slabs located between the configuration and the detector.

Comparisons of the fluxes calculated with both ANISN and DOT are presented in Table 5.10. The calculated results must be regarded as preliminary because (1) they do not include the effect of the iron frame surrounding the blanket, and (2) they used a P_3 set for the gamma rays in both DOT and FALSTF, which was shown above to be inadequate for the core gamma rays. From the comparisons with the measurements shown in Table 5.10,

Table 5.6. Comparison of Calculated and Measured Gamma-Ray Fluxes at the Detector Location Behind the Westinghouse Spectrum Modifier

Energy Interval (MeV)	Gamma-Ray Flux (photons·cm ⁻² ·sec ⁻¹ ·kW ⁻¹)				Ratio: Calc. Flux Meas. Flux
	Calculated			Measured	
	Secondary Gamma Rays ^a	Core Gamma Rays ^b	Total		
8-10	0.291	0.014	0.305	0.405	0.75
7-8	0.665	0.044	0.710	0.715	0.99
6-7	0.270	0.022	0.293	0.392	0.75
5-6	0.265	0.033	0.299	0.364	0.82
4-5	0.297	0.058	0.354	0.383	0.92
3.5-4	0.171	0.085	0.256	0.249	1.03
3-3.5	0.193	0.141	0.333	0.319	1.04
2.5-3	0.258	0.256	0.513	0.399	1.29
2-2.5	0.292	0.468	0.760	0.680	1.12
1.6-2	0.276	0.756	1.032	1.146	0.90
1.2-1.6	0.272	1.73	1.998	2.385	0.84
0.9-1.2	0.173	2.98	3.150	2.683	1.17
0.6-0.9	0.227	2.72	2.949	2.587	1.14
0.4-0.6	0.470	1.29	1.760	1.747	1.01
0.21-0.4	0.239	1.08	1.317	0.933	1.41
0.21-10	4.36	11.68	16.03	15.37	1.04

^aFrom Table 5.4.

^bFrom Table 5.5.

Table 5.7. Comparison of DOT-Calculated and Measured Bonner Ball Count Rates on the Center Line Behind the Westinghouse Spectrum Modifier and 5-in. Radial Blanket

	Count Rate (cpm/W)		
	3-in. Ball	6-in. Ball	10-in. Ball
<u>12 in. Behind</u>			
Calculated, scattered	691	5465	2756
Calculated, uncollided	2	57	77
Calculated, total	693	5522	2833
Measured	656	4775	2145
Calculated/Measured	1.06	1.16	1.32
<u>18 in. Behind</u>			
Calculated, scattered	448	3432	1684
Calculated, uncollided	2	49	60
Calculated, total	450	3481	1745
Measured	449	3148	1426
Calculated/Measured	1.00	1.11	1.22
<u>24 in. Behind</u>			
Calculated, scattered	310	2321	1118
Calculated, uncollided	1	43	53
Calculated, total	311	2365	1171
Measured	324	2224	1008
Calculated/Measured	0.96	1.06	1.16

Table 5.8. Comparison of DOT-Calculated and Measured Neutron Fluxes for Hydrogen Counter on Center Line 12 in. Behind Westinghouse Spectrum Modifier and 5-in. Radial Blanket

Energy Interval (MeV)	Neutron Flux (neutrons·cm ⁻² ·min ⁻¹ ·W ⁻¹)				Ratio:
	Calculated			Measured	Calc. Flux
	Scattered	Uncollided	Total		Meas. Flux
0.55-0.907	435	2.4	437	359	1.22
0.334-0.55	449	0.41	450	313	1.44
0.202-0.334	490	0.89	491	421	1.17
0.123-0.202	448	0.27	448	380	1.18
0.0408-0.123	586	0.15	586	558	1.05
0.0408-0.907	2408	4.1	2412	2031	1.19

Table 5.9. Comparison of DOT-Calculated and Measured Neutron Fluxes
for NE-213 Spectrometer at Two Locations Behind
Westinghouse Spectrum Modifier and 5-in. Radial Blanket

Energy Interval (MeV)	Neutron Flux (neutrons·cm ⁻² ·min ⁻¹ ·W ⁻¹)				Ratio:
	Calculated			Measured	<u>Calc. Flux</u> <u>Meas. Flux</u>
	Scattered	Uncollided	Total		
<u>217 in. Behind Configuration; 217 in. Off Center Line</u>					
6.7-10	0.0644	-	0.0644	0.0449	1.43
4.49-6.7	0.188	-	0.188	0.203	0.93
3.01-4.49	0.310	-	0.310	0.326	0.95
2.02-3.01	0.665	-	0.665	0.613	1.09
1.35-2.02	0.815	-	0.815	0.692	1.18
0.907-1.35	0.748	-	0.748	0.658	1.14
0.907-10	2.790	-	2.790	2.537	1.10
<u>153.5 in. Behind Configuration; On Center Line</u>					
6.7-10	0.434	0.651	1.085	0.997	1.09
4.49-6.7	1.08	1.24	2.31	2.45	0.94
3.01-4.49	1.47	1.30	2.78	3.05	0.91
2.02-3.01	2.83	2.30	5.13	5.21	0.98
1.35-2.02	3.22	1.94	5.16	5.03	1.03
0.907-1.35	2.81	0.49	3.30	3.65	0.90
0.907-10	11.84	7.92	19.76	20.39	0.97

Table 5.10. Comparison of Preliminary ANISN- and DOT-Calculated and Measured Gamma-Ray Fluxes at the Detector Location Behind the Westinghouse Spectrum Modifier and 5-in. Radial Shield

Contributions by Secondary Gamma Rays (ANISN Calc.)								
Energy Interval (MeV)	Fraction of Total Flux	Fraction of Secondaries Produced in Spectrum Modifier	Fraction of Secondaries Produced in Spectrum Modifier Due to Thermal Captures	Fraction of Secondaries Produced in Blanket Due to Thermal Captures	Total Gamma-Ray Flux (photons·cm ⁻² ·sec ⁻¹ ·kW ⁻¹)			Ratio: DOT-Calc. Flux Meas. Flux
					Calculated		Measured	
					ANISN	DOT		
8-10	0.98	0.72	0.64	0.001	3.58(-3) ^a	6.24(-3)	1.22(-2)	0.52
7-8	0.86	0.69	0.64	0.005	1.16(-2)	1.81(-2)	6.78(-2)	0.27
6-7	0.59	0.64	0.62	0.002	1.11(-2)	1.43(-2)	3.46(-2)	0.43
5-6	0.55	0.45	0.62	0.001	2.04(-2)	2.20(-2)	3.37(-2)	0.67
4-5	0.70	0.12	0.58	0.000	7.95(-2)	8.41(-2)	5.45(-2)	1.56
3.5-4	0.66	0.10	0.52	0.000	6.74(-2)	7.07(-2)	4.73(-2)	1.51
3-3.5	0.64	0.08	0.52	0.000	1.04(-1)	1.08(-1)	7.38(-2)	1.47
2.5-3	0.66	0.06	0.46	0.000	1.74(-1)	1.81(-1)	1.12(-1)	1.62
2-2.5	0.73	0.03	0.43	0.000	3.64(-1)	3.88(-1)	2.11(-1)	1.88
1.6-2	0.45	0.07	0.41	0.001	1.98(-1)	1.98(-1)	2.19(-1)	0.92
1.2-1.6	0.44	0.07	0.40	0.001	2.19(-1)	2.21(-1)	2.36(-1)	0.94
0.9-1.2	0.44	0.06	0.41	0.001	1.38(-1)	1.40(-1)	1.71(-1)	0.83
0.6-0.9	0.56	0.04	0.41	0.000	1.13(-1)	1.14(-1)	1.41(-1)	0.82
0.4-0.6	0.59	0.05	0.52	0.000	5.92(-2)	5.60(-2)	7.12(-2)	0.82
0.21-0.4	0.70	0.03	0.49	0.000	1.45(-2)	1.12(-2)	2.01(-2)	0.68
0.21-10	0.59	0.08	0.51	0.0004	1.577	1.661	1.507	1.10

^aRead as: 3.58 x 10⁻³.

it is clear that the iron frame is contributing appreciably to the gamma-ray flux above 5 MeV and that the core gamma-ray contribution in the energy range above 2 MeV is overcalculated. Both these deficiencies in the calculations have been corrected in a new calculation, the results of which are presented in Table 5.11.

It is to be observed that the calculated core gamma-ray contribution is now 41% of the total, compared with 73% behind the spectral modifier alone (see Tables 5.4 and 5.6). There is now adequate agreement in the region 5-10 MeV, indicating the important effect in this energy region of the iron frame surrounding the blanket. The calculated core gamma-ray contribution is still probably accounting for most of the discrepancy in the region 2-5 MeV, although the agreement with measurement has been somewhat improved over that of the preliminary calculation in this energy region. Overall, the agreement shown in Table 5.11 is considered to be quite adequate and far superior to that shown in Table 5.10.

Table 5.11. Comparison of Corrected DOT-Calculated and Measured
Gamma-Ray Fluxes at the Detector Location Behind the
Spectrum Modifier and 5-in. Radial Blanket

Energy Interval (MeV)	Gamma-Ray Flux (photons·cm ⁻² ·sec ⁻¹ ·kW ⁻¹)		Ratio: <u>Calc. Flux</u> <u>Meas. Flux</u>
	Calculated	Measured	
8-10	1.35(-2) ^a	1.22(-2)	1.10
7-8	8.64(-2)	6.78(-2)	1.27
6-7	2.66(-2)	3.46(-2)	0.77
5-6	3.16(-2)	3.37(-2)	0.94
4-5	8.17(-2)	5.45(-2)	1.50
3.5-4	5.86(-2)	4.73(-2)	1.24
3-3.5	9.03(-2)	7.38(-2)	1.22
2.5-3	1.46(-1)	1.12(-1)	1.30
2-2.5	3.50(-1)	2.11(-1)	1.65
1.6-2	1.60(-1)	2.19(-1)	0.73
1.2-1.6	2.06(-1)	2.36(-1)	0.87
0.9-1.2	1.75(-1)	1.71(-1)	1.02
0.6-0.9	1.23(-1)	1.41(-1)	0.87
0.4-0.6	6.57(-2)	7.12(-2)	0.92
0.21-0.4	2.19(-2)	2.01(-2)	1.09
0.21-10	1.636	1.507	1.09

^aRead as: 1.35 x 10⁻².

Section 5 References

1. F. R. Mynatt et al., "Fast Reactor Analytical Shielding Monthly Progress Report for August and September 1972," ORNL-TM-4040 (November 1972).
2. F. R. Mynatt et al., "Fast Reactor Analytical Shielding Monthly Progress Report for October, 1972," ORNL-TM-4081 (January 1973).
3. C. E. Clifford et al., "Fast Reactor Experimental Shielding Monthly Progress Report for April 1973," ORNL-TM-4254 (June 1973).
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6. STUDY OF THE SENSITIVITY OF CALCULATIONS OF THE TSF INCONEL EXPERIMENT TO NICKEL CROSS SECTIONS: CORRELATION WITH CALCULATIONS OF NEUTRON DAMAGE TO FTR GRID PLATE

The July experimental progress report¹ presents results from a TSF experiment in which 2-1/2-in. and 5-in. slabs of inconel were positioned immediately behind the Westinghouse spectrum modifier (see Fig. 5.1) and neutron measurements were made at various locations beyond the configurations. The purpose of the experiment was twofold: (1) to investigate the attenuating properties of inconel, which is to be used as a reflector in the FTR, and (2) to provide experimental results to test our capabilities for performing an analysis of a configuration containing nickel, the cross sections for which are not well known. The analysis, currently underway, is being performed with the two-dimensional DOT code using 21-group cross sections and will be described in the September-October analytical progress report. In the meantime, the SWANLAKE sensitivity code² has been applied to results from a preliminary one-dimensional 50-group ANISN analysis of the experiment to study the effects of the uncertainties in the nickel total cross sections. The results should give some insight into any large discrepancies that might occur between the full analysis and experiment. In addition, the sensitivity study has provided a demonstration of a technique in which the data from an experiment and a corresponding analysis can be used to evaluate the accuracy of calculations for a real system — in this case our calculations of the neutron damage to the FTR grid plate.³ The paragraphs below summarize both the sensitivity study of the experiment and the technique relating the results to the FTR calculation.

In the FTR calculation³ a stainless-steel energy-dependent neutron damage function was used as the response function so that the results were obtained in terms of the component lifetime per source neutron per square centimeter. The calculation was extended to include a study of the sensitivity of the damage to the nickel cross sections. The results indicated that the sensitivity per unit lethargy of the calculated grid plate damage to the nickel cross sections is less than 1 in all energy regions, which on the basis of experience with earlier sensitivity studies is considered to be low.

It is desired to correlate the calculations for the inconel experiment with the FTR calculations. If neutron damage were measured and the spectrum properly identified, the experiment-to-calculated ratios would give directly the error in the FTR calculations. A sensitivity calculation was performed for the TSF configuration with a hypothetical "damage function" detector. These sensitivity functions, shown in Fig. 6.1, do not resemble the sensitivity plot obtained in the FTR calculation. The plot for the 5-in. slab comes closest, but it shows a much higher sensitivity to high-energy cross sections than the plot for the FTR configuration showed. The fact that spectrometer measurements were made in the experiment provides additional information content. In this exercise we show that through sensitivity analysis this additional information can be used to construct an experimental comparison adjusted to the FTR sensitivity function. In order to do this, the experiment must be "close" to the design configuration in the sense that the adjustment cannot create sensitivity in a particular energy range which has little or no sensitivity in the experiment. This technique, however, allows wider-range use of the experiment and provides a clear correlation between experiment comparison and design analysis uncertainty.

In this demonstration of the sensitivity approach which can correlate the FTR calculation with the experiment, more detailed sensitivity studies were performed for the experimental configuration — a sensitivity function being generated for each of several energy bins over the energy range covered by the neutron spectrometer. Six energy bins were selected, and a nickel sensitivity calculation was performed for each bin for each of two spatial regions in the 5-in. inconel slab configuration, as well as for the total configuration. The resulting total-configuration flux sensitivity curve in each energy bin was then normalized to the FTR damage sensitivity curve in the same energy bin (each bin covered five energy groups). The normalized flux sensitivity curve is compared with the FTR damage sensitivity curve in Fig. 6.2. Except for the two lowest energy bins, the sensitivity plots for the two cases are very similar.

Fig. 6.1. Comparison of Damage Sensitivities to Nickel Cross Sections for 2-1/2-in. and 5-in. Inconel Configurations.

ORNL-DWG 73-11239

Fig. 6.2. Comparison of FTR Grid-Plate Damage Sensitivity to Nickel Cross Sections with Scaled Flux Sensitivity for 5-in. Inconel Configuration.

With the sensitivity curves now being similar, a comparison between the calculated and experimental responses for the TSF configuration can be used to determine the uncertainty in the FTR calculation due to uncertainties in the nickel cross sections. The procedure is to multiply the ratio between the calculated and experimental spectra in each energy bin by the weighting factor that was required to normalize the TSF sensitivity curve in that region to the FTR sensitivity curve. Adding the results for the six regions gives the calculation/experiment ratio in the calculated FTR damage response with prototypic sensitivity to the nickel cross sections. The weighting factors in this case are as follows:

<u>Energy Bin No.</u>	<u>Energy Range (MeV)</u>	<u>Weighting Factor</u>
1	12.2-4.49	0.0
2	4.49-1.65	0.17
3	1.65-0.608	0.34
4	0.608-0.224	0.53
5	0.224-0.0525	0.29
6	0.0525-0.015	0.063

When the comparisons of the two-dimensional calculations and experimental results for the TSF inconel experiment are completed, the procedure described above will be applied to the FTR calculation.

Section 6. References

1. C. E. Clifford et al., "Fast Reactor Experimental Shielding Monthly Progress Report for July 1973," ORNL-TM-4364 (September 1973).
2. D. E. Bartine, F. R. Mynatt, and E. M. Oblow, "SWANLAKE, A Computer Code Utilizing ANISN Radiation Transport Calculations for Cross Section Sensitivity Analysis," ORNL-TM-3809 (May 1973).
3. F. R. Mynatt et al., "Fast Reactor Analytical Shielding Progress Reports for May and June, 1973," ORNL-TM-4401 (to be published).

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