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FAST REACTOR SHIELDING MONTHLY PROGRESS REPORT FOR AUGUST, 1971,
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Abstract

This report summarizes the progress made by the ORNL Neutron Physics Division in Fast Reactor Shielding Research during the month of August, 1971.

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Monthly Progress Report

August 1971

FAST REACTOR SHIELDING

Activity No. 40 01 61

189A No. 10028

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ORNL-LMFBR SHIELDING PROGRAM
Analytical Program For FY 1972

	JUL	AUG	SEP	OCT	NOV	DEC	JAN	FEB	MAR	APR	MAY	JUN
Design and Analysis of Sodium Experiment	█	█	█	█	█	█						
Design and Analysis of Pipe Chase Streaming Experiment	█	█	█	█	█	█	█	█	█	█	█	
Calculations of Fluence Levels Above FFTF Head and Flange	█	█	█	█	█	█						
Calculations of Source Term for IHX Vault Shielding	█	█		█	█							
a. Effect of Shield Deck	█	█		█	█							
b. Effect of Fission Gas Plenum		█	█	█								
c. Effect of Holes in Deck				█	█	█						
Analysis of Effect of Stored Fuel on Head and IHX Shielding		█	█	█	█	█	█					
Analysis of TSF Annular Slit Experiment	█	█	█	█	█							
Analysis of Iron/Steel Benchmark Experiment	█	█	█				█	█	█			
Secondary Gamma-Ray Transport Analysis												
a. Preparation of Coupled Neutron-Gamma Library for LMFBR Problems					█	█	█	█	█	█	█	█
b. Analysis in Support of Demonstration Plant Shielding Experiments				█	█	█	█	█	█	█	█	█
Absolute Attenuation Comparisons of Calculations and the Prototype Grid-Plate Shield Experiment		█	█	█	█							
Calculations of Effects of Design Changes in the FTR Grid-Plate Shield			█	█	█							
Analysis of Shielding Problems for LLFM Monitors			█	█								
Analysis of IHX Vault Shielding					█	█	█	█	█	█		

ORNL-LMFBR SHIELDING PROGRAM
Experimental Program for FY 1972

	JUL	AUG	SEP	OCT	NOV	DEC	JAN	FEB	MAR	APR	MAY	JUN
Deep-Penetration Sodium												
a. Experimental Setup and Checkout												
b. Measure Neutron Spectra at One or More Angles and at Six Thicknesses Including 15 ft												
Pipe Chase Streaming Experiment												
a. Map Source and Measure Transmission												
b. Determine Effect of Addition of Steel Liner, Steel Pipe, Sodium, and Insulation												
Iron/Stainless Steel Benchmark Measurement H ₂ Spectrometer Down to 10 keV												
Experiments for Demo Design												

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A. Transport Method Development for All Metal Shields

1. Prototype Bottom Grid-Plate Shield Experiment - Discrete Ordinates Analysis

The discrete ordinates analysis of the bottom grid-plate shield experiment performed at the TSF in May, 1971, consists of several DOT-III calculations for the configurations shown in Figs. 1-4. For configurations 2 and 3 (Figs. 3 and 4), the homogenized insert #3 with no lucite detector collimator was calculated. All of the calculations were performed using 50-group cross sections and an S-100 angular quadrature. The iron cross sections were the ORNL (Penny) evaluation with a $1/E \sigma_T$ weighting and the sodium cross sections were from the preliminary ENDF/B-III evaluation. The cross-section expansion is as noted for each calculation.

The first calculation was for the SNAP reactor and the spectrum modifier (10-in. SS-304, 1-in. polyethylene, two 1/8-in. sheets of boral) with the leakage spectra from the bare SNAP reactor as an internal boundary source at the reactor-modifier interface. The 50-group source was obtained as described in the monthly progress report for July, 1971. A P_1 cross-section expansion was used. From this calculation absolute comparisons were made with hydrogen counter measurements and Bonner ball detector count rates on centerline at the respective detector positions shown in Fig. 1. The calculated vs. measured hydrogen counter spectra 2-1/2 in. below the boral are shown in Fig. 5. As may be seen, the agreement in shape and intensity is excellent. The integral of the calculated spectra from 68 keV to 1.5 MeV agrees with the integral of the measured spectra over the same energy region to within 1%. The calculated Bonner ball count rates were obtained with the SPACETRAN code and are compared with measured values in Table 1. The calculated count rates are low by a factor of 2 which is quite unexpected

Fig. 1. Experimental Configuration No. 1 For Spectral Modifier

Neutron Transmission Measurement.

Fig. 2. Modified Spectral Modifier Configuration No. 1 for the NE-213 Measurements.

Fig. 3. Configuration No. 2, Spectral Modifier and Grid-Plate Shield Mockup Arrangement.

Fig. 4. Configuration No. 3, Spectral Modifier and Grid-Plate Shield Arrangement With the Reactor Decoupled.

Fig. 5. Comparison of Unfolded Hydrogen Counter Spectra With DOT-III Calculation. $\square$ - Unfolded hydrogen counter spectra; $\bigcirc$ - DOT-III calculation.

Table 1. (DOT-SPACETRAN) Calculated and Measured Bonner Ball
 Detector Count Rates for Configuration #1 Without
 Scattering Effects Below Shield

Detector	Measured (cpm/W)	Calculated (cpm/W)	Calculated/ Measured
3-in. ball*	7.645(+5) [†]	3.598(+5)	0.47
4-in. ball	1.846(+6)	9.121(+5)	0.49
5-in. ball*	2.251(+6)	1.178(+6)	0.52
6-in. ball	2.144(+6)	1.164(+6)	0.54
8-in. ball	1.395(+6)	7.870(+5)	0.56
10-in. ball	7.035(+5)	4.013(+5)	0.57
12-in. ball	3.155(+5)	1.855(+5)	0.59

* Average of two measurements.

[†] Read as 7.645×10^5 .

in view of the hydrogen counter spectra comparisons of Fig. 5. Since the 8-in. detector response peaks in the energy region of the hydrogen counter measurement, there appears to be a discrepancy between the calculated Bonner ball count rates and the calculated hydrogen counter spectra. The major source of discrepancy in the comparison of these calculated count rates was found to be due to scattering from the lithiated-paraffin and concrete side shield which was neglected in this first calculation. Since the hydrogen counter is only 2-1/2 in. below the modifier, a position at which such scattering is negligible, the calculation of the hydrogen counter spectra is correct but the position of the Bonner balls was such that they were very sensitive to scattering. The coupled SNAP reactor-spectrum modifier was then recalculated using a P_3 cross-section expansion and a full model as shown in configurations #1 and #1-modified (Figs. 1 and 2).

The geometry in the calculation for configuration #1 consisted of the 8-in. SS-304, 1-in. polyethylene, and 1/4-in. boral, together with the lithiated-paraffin and concrete side shield down to and including the cover over the pool. The Bonner ball count rates were obtained at a position corresponding to the center of each detector and the calculated results are presented in Table 2, together with the measured values. As may be seen from the comparisons, the agreement between the measured and calculated values has improved markedly and appear to be very good. Thus, approximately one-half the total count rate in this configuration for each detector is due to the combined scattering from the side shield and the pool cover.

A calculation was made of the modified configuration #1 including the 1-15/16 in. zirconium slab gamma filter. This calculated result and the

Table 2. DOT Calculated and Measured Bonner Ball Detector
Count Rates for Configuration #1 With Scattering
Effects Below the Shield

Detector	Measured (cpm/W)	Calculated (cpm/W)	Calculated/ Measured
3-in. ball*	7.645(+5) [†]	9.00(+5)	1.18
4-in. ball	1.846(+6)	1.910(+6)	1.03
5-in. ball*	2.251(+6)	2.273(+6)	1.00
6-in. ball	2.144(+6)	2.140(+6)	1.00
8-in. ball	1.395(+6)	1.317(+6)	0.94
10-in. ball	7.035(+5)	6.456(+5)	0.92
12-in. ball	3.155(+5)	2.841(+5)	0.90

* Average of two measurements.

[†] Read as 7.645×10^5 .

measured NE-213 spectra in the energy range 1-15 MeV is shown in Fig. 6. The agreement in shape and intensity appears to be fairly good with the calculation approximately 50% low above 2 MeV and from 0-10% high below 2 MeV.

Configuration #2, with the spectrum modifier separated from the insert and collar by 18 in. and including all of the side shielding from the modifier down to the pool cover, as shown in Fig. 3, was simulated in another DOT calculation. The calculated 5-in. Bonner ball detector count rate on centerline is 1.183×10^5 cpm/W compared to a measured count rate of 1.373×10^5 cpm/W. This gives a calculated/measured ratio of 0.86.

A similar calculation was made for configuration #3, shown in Fig. 4. The spectrum modifier on top of the insert and collar and all of the side shielding was simulated in the geometry as before. The calculated 5-in. Bonner ball count rate for this configuration is 7.38×10^4 cpm/W as compared to a measured count rate of 9.73×10^4 cpm/W, giving a calculated/measured ratio of 0.76. A previous calculation of this configuration, which did not include the lithiated-paraffin-concrete shielding to the side and below the detector gave a count rate of 6.7×10^4 cpm/W for a calculated/measured ratio of 0.69. Thus, it appears that approximately 10% of the calculated count rate is due to scattering below the detector. It is not clear at present as to the reason for the difference in the ratios of calculated to measured count rates for configurations 2 and 3. The calculation for configuration #3 assumed that the reactor and modifier were completely decoupled by the separation distance of 18 in. This assumption may not be correct and is being investigated.

Fig. 6. Comparison of NE-213 Spectra With DOT-III Calculation.

--- - Unfolded NE-213 spectra; — - DOT-III calculation.

B. Current FFTF Studies

1. Grid-Plate Shield Analysis

Analysis of the core support grid-plate shielding problem is continuing. An additional DOT calculation was performed for the purpose of testing the sensitivity to cross-section sets. The new calculation used the same model as previous ones, except that the stainless-steel cross sections used ORNL (Penny) iron, ENDF/B-II Cr and Ni, and ENDF/B-I Mn in which the $1/E \sigma_T$ weighting function was calculated from the point values of the combined Fe, Cr, and Ni cross sections. In order to do this, the smooth cross sections for Cr and Ni had to be constructed from the resonance parameters. This new DOT calculation gave a peak total flux at the grid of 3.66×10^{13} neutrons/cm²/sec, which is the lowest value yet predicted. This does not seem reasonable since the ENDF/B-II Cr and Ni cross sections appear to have s-wave resonance interference valleys which are too "low." It is suspected that there is an error in the preparation of this last set of cross sections. The three calculations in this series are summarized in Table 3. Comparisons of DOT calculations with the TSF grid-plate shield experiment, which are given in another section of this report, indicate that the true flux level should be higher by a factor between 1.16 and 1.32 than the highest value (4.78×10^{13}) given in Table 3. Consideration of the experimental corrections gives a total fluence of 1.4×10^{22} neutrons/cm² ($\pm 7\%$) in ten years at 75% availability for a 304 stainless-steel shield of this design. Having recently been informed of a design change to 316 stainless steel in the grid-plate shield, the DOT below-core calculations will be revised accordingly.

Table 3. Effect of Cross-Section Sets on Maximum Total Flux on the Core Support Grid-Plate Shield

Cross-Section Set	Maximum Flux (neutrons/cm ² /sec)
ORNL(Irving) Fe 1/E σ_T weighting + ENDF/B-I Cr, Ni, Mo, Mn	4.24×10^{13}
ORNL(Penny) Fe 1/E σ_T weighting + ENDF/B-I Cr, Ni, Mo, Mn	4.78×10^{13}
ORNL(P) Fe, ENDF/B-II Cr, Ni (1/E σ_T weighting) + ENDF/B-I Mo, Mn	3.66×10^{13}

2. Effects of Stored Fuel

The fissions in the stored fuel produce fast neutrons which greatly enhance the fast flux in the cavity. Calculations are being performed in order to evaluate the effect of the stored fuel and to indicate suitable design changes. DOT R- θ calculations have been performed with the same model that was used for the study of californium source enhancement of ex-vessel shutdown detector performance. This model had 22 core-1 composition fuel elements per 120° sector and did not include the guard vessel. Calculations were performed with and without stored fuel for the full power condition. Table 4 shows the neutron spectra in the center of the radial reactor cavity with and without the stored fuel. Table 5 shows the ratio for every fifth group of the results with stored fuel to that without stored fuel. The enhancement is very large at high energies and decreases to a negligible effect near thermal. The impact of the stored fuel is therefore very sensitive to the spectral dependence of the neutron importance in the cavity. This problem is very difficult because of the three-dimensional nature of the stored fuel and the pipe penetrations in the proposed cavity shield deck and the relative importance of streaming and bulk transmission in the head and vessel support system.

The ORNL effort on this problem will consist of DOT R- θ calculations to support similar diffusion theory calculations at WARD and the development and application of DOT and MORSE models to evaluate the effect of the fast flux enhancement in the cavity on the shielding problems in the head, vessel support, and pipe penetration areas.

Table 4. Effect of Stored Fuel on Neutron Flux in Cavity

Group	Upper Energy of Group (eV)	Flux Without Stored Fuel (neuts/cm ² /sec/U)	Flux With Stored Fuel (neuts/cm ² /sec/U)
1	1.49000E 07	2.11458E C1	6.10445E 05
2	1.22000E 07	1.71514E 01	2.63794E 06
3	1.00000E 07	1.75627E 01	9.33130E 06
4	8.19000E 06	1.91696E 01	2.43668E 07
5	6.70000E 06	1.92002E 01	4.31537E 07
6	5.49000E 06	2.30023E 01	8.33983E 07
7	4.49000E 06	2.81120E 01	1.24044E 08
8	3.68000E 06	7.07938E 01	2.54709E 08
9	3.01000E 06	1.55060E 02	4.10766E 08
10	2.47000E 06	2.29287E 02	4.92362E 08
11	2.02000E 06	1.59560E 03	8.62084E 08
12	1.65000E 06	7.62477E 03	1.31580E 09
13	1.35000E 06	1.70561E 04	1.66136E 09
14	1.11000E 06	4.26354E 04	1.68416E 09
15	9.07000E 05	7.17488E 04	1.96534E 09
16	7.43000E 05	8.46816E 04	2.08345E 09
17	6.09000E 05	1.70044E 06	3.25697E 09
18	4.98000E 05	3.54286E 06	2.32780E 09
19	4.08000E 05	1.05676E 07	4.57059E 09
20	3.34000E 05	3.73094E 07	5.99088E 09
21	2.73000E 05	4.11888E 07	5.08624E 09
22	2.24000E 05	4.54351E 07	4.33211E 09
23	1.83000E 05	9.38015E 07	5.79036E 09
24	1.50000E 05	1.53501E 08	6.38529E 09
25	1.23000E 05	2.52629E 08	6.04476E 09
26	9.65000E 04	2.57545E 08	4.56825E 09
27	5.25000E 04	2.87085E 08	4.31799E 09
28	4.09000E 04	2.63707E 08	3.45043E 09
29	3.18000E 04	4.37985E 08	4.65172E 09
30	2.48000E 04	6.97481E 08	5.98034E 09
31	1.93000E 04	3.52471E 08	2.95144E 09
32	1.50000E 04	4.73912E 08	3.28727E 09
33	7.10000E 03	3.39557E 08	2.34813E 09
34	4.31000E 03	3.76426E 08	2.40317E 09
35	3.35000E 03	3.22981E 08	2.13524E 09
36	2.61000E 03	2.99897E 08	2.04195E 09
37	2.03000E 03	3.93893E 08	2.45861E 09
38	1.58000E 03	5.20852E 08	2.88664E 09
39	1.23000E 03	4.89529E 08	2.57490E 09
40	9.61000E 02	1.24872E 09	3.60307E 09
41	4.54000E 02	1.12943E 09	2.95613E 09
42	2.14000E 02	2.39628E 09	4.02505E 09
43	1.01000E 02	3.32298E 09	4.51424E 09
44	4.79000E 01	3.86063E 09	4.57345E 09
45	2.26000E 01	4.14806E 09	4.53018E 09
46	1.07000E 01	4.13470E 09	4.26108E 09
47	5.04000E 00	3.87009E 09	3.91604E 09
48	2.38000E 00	3.44498E 09	3.53090E 09
49	1.13000E 00	3.13250E 09	3.52051E 09
50	4.14000E-01	8.92309E 09	1.10673E 10

Table 5. Ratio of Neutron Flux With Stored Fuel
to That Without Stored Fuel

Group	Ratio
1	2.89×10^4
5	2.50×10^6
10	2.15×10^6
15	2.74×10^4
20	1.61×10^2
25	2.39×10^1
30	8.54
35	6.61
40	2.89
45	1.09
50	1.34

3. Fission Gas Plenum Studies

When an overlap zone on the side of the plenum was included in the latest DOT calculation for a 201 radial interval fission gas plenum model with 315 angles and P_1 scattering, it was determined that it could not fit in the available memory on the 360/91 computer. A 101 radial interval model was then selected as an appropriate alternative. A DOT problem was prepared with this model and the fission gas plenum portion of the calculation is expected to be completed early next month.

4. Evaluation of Material Changes for the Thermal Shield

Several DOT calculations were performed to determine the effect of various choices of thermal shield materials. These calculations included the head area only and are similar to those performed for evaluation of vessel support streaming except that the head and thermal shield extend completely across the assembly. The input boundary source was obtained from an r-z calculation with an homogenized fission gas plenum and was specified at an elevation of 647.4 cm. The old and new calculations are compared on the basis of the leakage from the top head. Since the ORNL (Irving) preliminary evaluation for iron has been used in all previous head calculations, it was first necessary to determine the effect of changing to the new ORNL (Penny) iron evaluation. Use of the Penny cross sections had been delayed because it would have been confusing to other comparisons in previous studies if the cross sections had also been changed. Calculations were then performed for two additional thermal shield compositions. In these calculations, the ORNL (Penny) iron with $1/E \sigma_T$ weighting was used and for the new configurations the thermal shield zones were at full density. The results are shown in Table 6. It is apparent that the change to the

Table 6. Comparison of Neutron Transmission For
Several Thermal Shield Compositions

Description of Head and Thermal Shield	Relative Leakage
25-in. (80% dense) SS-304, 21-in. iron (ORNL-Irving cross sections)	1.0
25-in. (80% dose) SS-304, 21-in. iron (ORNL-Penny cross sections)	3.24
1.3-in. SS-304, 8.0-in. Inconel, 8.0-in. SA516, 21.75-in. SA508	4.54
1.3-in. SS-304, 5.0-in. Inconel, 4.0-in. SS-410, 8.0-in. SA508, 21.75-in. SA516	8.18

ORNL (Penny) cross sections is a major but not unexpected effect. It is also apparent that there is a significant loss of shielding effectiveness in the newer thermal shield compositions. This is due to both a decrease in thickness and nickel content in the new compositions.

5. Shielding Considerations for the Low-Level Flux Monitor Guide Tubes

The present FFTF design includes three low-level flux monitors (LLFM). These are ^{235}U fission detectors which will measure the multiplication during refuelling operations and the early stages of start-up. However, present day technology monitors are unable to withstand the high burnup they would incur if left in their monitoring position, so it is proposed that they be withdrawn to a position where their lifetime from irradiation damage is reasonable.

In the present design, the monitors can be retracted while the reactor is at power and removed periodically to allow replacement through 6-in.-diam. pipes leading from the core midplane level to the top of the head. A brief study has been performed to explore the shielding requirements needed in these pipes and the effects of various engineering clearances which may be required. Several problems were considered that might result from partial or complete retraction of the detectors. While these calculations were not performed to evaluate a particular proposed design, the results should serve as a guide to illustrate shielding requirements for the LLFM design and operation.

a. Hole to Top of Pool

One possible option is that the monitors may need to be withdrawn to a position above the sodium pool and just below the thermal shield and head to reduce its temperature. This will leave a void from the monitoring

position to the bottom of the head. The hole in the thermal shield and head is assumed to be filled with removable shielding. One case assumes the adjacent materials in the thermal shield and head are used as filling while the other assumes a shield plug consisting of 80% theoretical density boron carbide fills the hole. Calculations assumed an isotropic source ($\phi = 1.8 \times 10^{14}$ n/(cm²sec) entering the pipe at the core midplane level at a position between the radial reflector and radial shield. The assumed spectrum and intensity were those calculated with DOT for this position. DOT calculations were then performed for the head and pool zones surrounding the upper end of the pipe void and the maximum doses (mrem/hr) on the top of the head were determined to be:

	<u>Dose (mrem/hr)</u>	<u>Position of Maximum Dose</u>	<u>% of Dose in Groups 30-33 (5-25 keV)</u>
(1) Boron carbide in hole	. 500	7 in. from center of hole	33
(2) Adj. materials in hole	4500	center of hole	32

The average dose from these holes over the area of the head (4.5×10^4 in.²) is 11 mrem/hr and 100 mrem/hr for the respective cases. Fluences [n/(cm²sec)] are approximately 80 times the doses for these spectra. These doses are much higher than the design objectives so some method such as flooding the holes with sodium after withdrawing the monitor must be used to meet the shielding design objectives with the monitor above the sodium pool at power.

b. Fission Reaction Rate

Assuming that the monitor can be designed to withstand the high temperatures, the integrated number of fissions per unit volume should

determine the monitor lifetime. A criteria suggested by Rathbun of WARD is that the monitors be replaced after one year's full power operation and be located during operation so that the maximum annual burnup is equivalent to that in a thermal flux giving 10^{19} nvt in one year. The equivalent flux is obtained as follows: $3.2 \times 10^{11} \sigma_{th}^{fiss} = \int \phi(E) \sigma^{fiss}(E) dE$ (ignoring self shielding in the counter which may be important).

v2U For a pipe starting at the boundary between the radial reflector and shield at the core midplane level, the criteria for the ^{235}U fission reaction rate is met at a distance of 3.5 ft from the monitoring position. This point is just above the level of the top of the core. At this position the fission reaction rate varies by a factor of 2-3 for each foot moved along the tube. This calculation considers the surrounding environment as dictating the position. No enhancement is included to account for streaming in the tube below the detector which may or may not be a problem.

c. Clearance Gaps

Some mechanical device will be required to retract the monitor during operation. Also, assuming the monitor can be positioned near the core during full power operation, the hole from the monitor to the top of the head must be filled with a removable shield such as stainless steel. Clearance gaps and coolant channels may be required in such a design. In order to get a feeling for the streaming problems associated with small gaps running 27 ft from the core centerline to the top of the head, three cases were evaluated.

1. 1/16-in. annular gap at inside edge of pipe (3-in. radius).
2. 1/16-in. annular gap surrounding a solid 2-in.-diam. shaft.
3. 1/16-in.-radius circular hole at the center of the pipe.

The following doses were calculated at the top of the holes and also averaged over the head area:

	<u>Peak Dose (mrem/hr)</u>	<u>Sum of Areas of Three Holes (in.²)</u>	<u>Dose Averaged Over Head (mrem/hr)</u>
(1)	3.0×10^5	3.5	23
(2)	1.8×10^5	1.2	5
(3)	0.5×10^5	0.04	0.04

The fluences are about 30 times the doses for these spectra. The doses due to annular slits can be decreased by stepping the pipes and central core or decreasing the slit widths. The peak doses for the annular slits decrease by a factor of about 3 when the slit width is halved.

Summary

Conclusions from this study are as follows:

(1) The low-level flux monitors cannot be withdrawn to the head during reactor operation unless the access tubes are flooded with sodium and the shielding in the tubes within the head is equivalent to that of the head materials. Optimum shielding in the head penetration combined with a gas-filled tube gives an unacceptable dose level above the head of 500 mrem/hr.

(2) Assuming that the monitor system is designed to withstand the high thermal environment, its position during reactor operation will be dictated by burnup. Calculations show that a one-year lifetime can be obtained by retraction to a position 3.5 ft above the midplane, and a doubling of the lifetime is obtained for each additional foot of retraction.

(3) For a detector location as described in (2), it may be assumed that possible designs would contain contiguous gas-filled clearance gaps extending from the detector to the top of the head for the purpose of detector movement, shield withdrawal, and coolant flow. Hand calculations of 1/16-in. annular clearance gaps show dose levels as high as 3.0×10^5 mrem/hr. Therefore, it appears that all such gaps must be stepped to provide suitable offsets.

C. Monte Carlo Transport Through Sodium

MORSE trial calculations have been performed for penetration through the full thickness of sodium and aluminum using path-length stretching and source biasing parameters determined from an adjoint ANISN calculation. The source used in the adjoint calculation was constant in energy for all groups above thermal. This corresponds to a constant response for neutrons at any energy above thermal in the forward problem.

The statistics of the MORSE calculation were somewhat disappointing (10% statistics on the total episcadmium flux for an hour run on the 360/91). Efforts are now underway to incorporate angular biasing into the calculation, which in conjunction with the path-length stretching should improve matters considerably. Also, the effect of an adjoint run using a uniform source above ~ 3 keV on the forward flux above 3 keV will be investigated, in an attempt to improve the calculation of leaking neutron fluxes in the region above 20 keV. Better than 95% of the episcadmium neutrons leaking 15 ft of sodium have energies below 1 keV.

D. ORNL-LMFBR Experimental Program at the TSF

1. Sodium Benchmark Experiment

During the month, the concrete about the reactor collimator was poured, as shown in Fig. 7, the ~1-in.-thick borated polyethylene liner was inserted into the collimator, and the lead shutter installed. The upper concrete cradles surrounding the sodium tanks were installed and the photographs illustrate (Figs. 8 and 9) the resulting configurations for tank #1 (~2.5 ft of sodium) and tank #2 (~5 ft of sodium). The reactor collimator and sodium tanks are now ready for use.

The thickness variation of each of the four sodium tanks was measured along the vertical centerline at roughly one-foot intervals. Table presents the results. Thickness measurements were made for tanks 1 and 4 by plumbbobbing two vertical lines 42 in. apart and 72 in. apart for tanks 2 and 3. Subtracting the horizontal distances to the front and back of each tank at each vertical location from the basic dimension gave the thickness of the tank. From the results presented in Table 7, it is evident that tanks 1, 3, and 4 are slightly bowed inward at the center, and that tank 2 is slightly thicker below the centerline than above it.

Measurements were also made of the variation of the sodium density along the vertical centerline for each of the four tanks. These were made using a ^{24}Na source placed about 1 in. in front of each tank and an uncollimated 3×3 in. NaI detector placed about 60 in. behind the tank. The source and detector were always located in a horizontal plane and the measurement consisted of determining the ratio of counts under the 2.75-MeV total absorption peak from the uncollided flux with and without the tank present. Corrections were made for source decay, variable source-to-detector distance, attenuation due to the 1 in. of aluminum and 5 ft

Fig. 7. Concrete Interface for Reactor Beam Shield.

Fig. 8. Sodium Sample, 2-1/2 ft thick by 11-ft diam., and Concrete Collar.

Fig. 9. Sodium Sample, 5-ft thick by 11-ft diam., and Concrete Collar.

Table 7. Variation of Tank Thickness Along the Vertical Centerline

Approximate Location Above Horizontal CL	Tank #1	Tank #2	Tank #3	Tank #4
+5 ft	30 in.	60 in.	$60 + 1/32$ in.	$30 + 1/8$ in.
+4 ft	$29 + 31/32$	$59 + 25/32$	$59 + 3/4$	$29 + 13/16$
+3 ft	$29 + 27/32$	$59 + 9/16$	$59 + 19/32$	$29 + 1/2$
+2 ft	$29 + 3/4$	$59 + 23/32$	$59 + 17/32$	$29 + 1/4$
+1 ft	$29 + 11/16$	$59 + 7/8$	$59 + 7/16$	$29 + 1/16$
0	$29 + 25/32$	$60 + 1/4$	$59 + 21/32$	$29 + 3/32$
-1 ft	$29 + 27/32$	$60 + 9/16$	$59 + 31/32$	$29 + 5/16$
-2 ft	$29 + 31/32$	$60 + 19/32$	$60 + 7/32$	$29 + 1/2$
-3 ft	30	$60 + 11/32$	$60 + 7/32$	$29 + 11/16$
-4 ft	$30 + 1/32$	$60 + 5/32$	$60 + 1/16$	$29 + 31/32$
-5 ft	$30 + 1/16$	60	60	$30 + 1/8$
Max. Var.	$3/8$	$1 + 1/32$	$25/32$	$1 + 1/16$
Ave. Tank Thickness	$29 + 29/32$	$60 + 5/64$	$59 + 55/64$	$29 + 19/32$
Ave. Sodium Thickness	$28 + 29/32$	$59 + 5/64$	$58 + 55/64$	$28 + 19/32$

of air, as well as for a background measured with the source removed, which in some cases was important and added to the low energy side of the total absorption peak due to the uncollided flux. The calculated densities are tabulated in Table 8.

The measurements for tank 1 were obtained when the source was weakest and, hence, were the least accurate. This is reflected in the relatively large variation in the density in these measurements, which is thus probably due to statistics. Since there is no apparent correlation between the density and thickness at a **given** location, the conclusion was reached that the density of the sodium is sensibly independent over any one tank, and from tank to tank as well. Thus, the calculations will assume a density of 0.945 g/cm^3 , which should be accurate to better than 1%.

2. Bonner Ball Response Function Measurements

The upscattering correction to the response functions for the Bonner balls was found to be negligible above $\sim 10 \text{ keV}$ (see ORNL-TM-3451 and ORNL-TM-3465). For use in the sodium benchmark experiment, however, the responses should peak in the region around 10-100 eV for deep penetration, where upscattering corrections may not be negligible. A series of time-of-flight measurements utilizing ORELA in the incident energy range 0.5 eV to 100 eV were made to compare the relative responses of a 3-in. Bonner ball ($\sim 1/2$ -in. shell of polyethylene) to a 2-in. Bonner ball (i.e., no polyethylene shell). The ratios as a function of energy were then compared to the calculated ratios using no upscattering (ORNL-TM-3451). The preliminary results indicate that the effect of upscattering reduces the 3-in. Bonner ball response by $\sim 25\%$ at 0.5 eV, $\sim 20\%$ at 1.5 eV, $\sim 15\%$ at 10 eV, and $\sim 10\%$ at 30 eV. The effect disappears in the vicinity of 100 eV. The direction

Table 8. Variation of Sodium Density Along the
Vertical Centerline Using Source #2

Approximate Location Above Horizontal CL	Tank #1	Tank #2	Tank #3	Tank #4
+3 ft		0.941 g/cm ³	0.940	
+2 ft	0.933		0.948	0.949
+1 ft	0.934	0.941	0.950	
0		0.945		0.958
-1 ft	0.940*	0.946	0.953	
-2 ft			0.944	0.948
-3 ft	0.963		0.942	
-4 ft			0.945	0.954
Max. Var.	0.030	0.005	0.013	0.010
Ave. Sodium Density	0.942	0.943	0.946	0.952

* Average of two values.

of the effect is reasonable, since upscattering tends to elevate the average neutron energy leaking the polyethylene's inner surface which reduces the capture probability in the BF_3 . Further, as the source neutron energy increases, an increasing percentage of the neutrons leaving the inner polyethylene surface lie above the upscattering region (~ 2 eV), and the response of the BF_3 , which is dominated by the $1/V$ capture of these neutrons, is affected less and less by the thermal treatment. Similar reasoning indicates that as the thickness of the polyethylene increases (at least until spectral equilibrium is reached), the upscattering effect increases for a fixed source energy, since more leaking neutrons will have been degraded to energies lying below 2 eV. After spectral equilibrium is reached, the upscattering effect should saturate, i.e., for the bigger Bonner balls the upscattering correction for a particular source energy should approach a constant value. These general trends were observed in the ANISN response calculations performed much earlier in the program; however, the magnitude of the effect seemed too exaggerated, perhaps due to a "bug" in the calculation, and hence, these responses with upscattering were not used. However, by suitably correlating the measured upscattering effect for the 3-in. Bonner ball with these early calculations, a series of upscattering corrections could be generated for all the Bonner balls, subject only to the limitation that the corrections do not exist for source energies above ~ 10 keV. Corrections of this kind will be made prior to comparison of calculations with the measured Bonner ball responses behind 15 ft of sodium.

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