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A COMPARISON OF ROD-VERSUS SLAB-TYPE CORE NEUTRON SHIELD FOR LMFBR APPLICATIONS

FINAL REPORT

W. H. HARLESS
M. L. WEISS

U. S. ATOMIC ENERGY COMMISSION
CONTRACT AT(04-3)-893 TASK 4

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**A COMPARISON OF ROD- VERSUS SLAB-TYPE
CORE NEUTRON SHIELDS FOR LMFBR APPLICATIONS**

**FINAL REPORT
APRIL 1973**

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Prepared for the
United States Atomic Energy Commission
Under Contract AT(04-3)-893
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ABSTRACT

This report summarizes the results of experimental and analytical studies on LMFBR in-vessel axial and radial shields. The shielding measurements were performed at the Oak Ridge National Laboratory (ORNL) Tower Shielding Facility during the period June to September 1972. The test configurations were designed and fabricated by General Electric's Nuclear Energy Division based upon types of neutron shields planned for use in an LMFBR demonstration plant. The configurations tested, experiments performed, and the analytical methods are described; and comparisons between measured and calculated results are presented and interpreted.

1. INTRODUCTION AND SUMMARY

The objective of this LMFBR shielding experimental program was to reduce design uncertainties for in-vessel neutron shields, especially for streaming effects due to composition heterogeneities.¹⁻⁴

In the PA-62 program, shielding experiments and their analyses were focused on reducing calculational uncertainties for the following neutron shields (Figure 1-1):

- Reactor Closure lower neutron shield (this is the principal bulk shield for reducing the neutron flux level above the reactor closure head).
- Radial neutron shield (this shield limits neutron fluences at the lateral core support and at the reactor vessel).

An important calculational uncertainty for these two types of neutron shields arises from their actual heterogeneous form as opposed to their approximated homogeneous form in calculations. Configurations simulating the actual shields were designed and fabricated at General Electric's Nuclear Energy Division in San Jose, California. Testing was completed at the ORNL Tower Shielding Reactor. Measurements were made of neutron transmission through heterogeneous near-core neutron shields. Comparisons of measured data were made to computed results in order to assess calculational uncertainties for these types of neutron shields.

The following 2 types of experimental configurations were designed and fabricated for testing in this program:

- Axial: 5-1/4-inch o.d. solid Type-304 stainless steel rods, 2 feet in length, with sodium-filled interstitial spaces, or "triflutes" (Figure 1-2).
- Radial: 1 to 4 rows of 1-inch, 2-inch and 5-1/4-inch o.d. B₄C-filled tubes and their slab equivalents (Figures 1-3, 1-4 and 1-5).

As a result of FY-73 mid-year program scope changes, the analytical work originally planned for this program was reduced. Due to competing demands of the FFTF program on the ORNL Tower Shielding Facility, and associated schedule delays, part of the experimental work has not yet been completed, specifically the axial shield and radial stainless steel versus B₄C trade-off experiments. Consequently, no two-dimensional or three-dimensional analysis has been done for these items. It is expected, however, that ORNL measurements for these parts of the program will be completed during 1973 and reported separately by ORNL. The reader is referred to four quarterly reports (references 1-4) for detailed information on the scope and on all other phases of this program.

Figure 1-1. Near-Core Upper Axial and Radial Neutron Shields

Figure 1-2. Axial 5-1/4-inch-o.d. Solid Type-304 Stainless Steel Rods

**HOMOGENEOUS EQUIVALENT OF
5-1/4-in.-O.D. B₄C-FILLED TUBES**

Figure 1-3. Radial Shielding Experiment: 5-1/4-inch-o.d. B₄C-Filled Tubes and Slab Equivalents

Figure 1-4. Radial Shielding Experiment: 2-inch-o.d. B_4C -Filled Tubes

1-in.-O.D. B_4C -FILLED TUBES

**HOMOGENEOUS SLAB EQUIVALENTS OF
1 in.-O.D. B_4C -FILLED TUBES**

Figure 1-5. Radial Shielding Experiment: 1-inch-o.d. B_4C -Filled Tubes and Slab Equivalents

2. CONCLUSIONS

The principal general conclusions which have been derived from the neutron transmission measurements and comparisons with calculated results are:

- Radial shields consisting of cylindrical rods (2 or more rows) are equally effective as homogeneous slab shields containing the same material composition and mass.
- Good agreement was obtained between measured and calculated neutron attenuation for both rodDED and homogeneous slab shield configurations, with moderate sophistication in calculational methods.

These principal conclusions are based on the following specific observations:

- The neutron attenuation through two rows of close-packed B₄C-filled tubes, up to 5-1/4 inches in diameter, is approximately the same (less than 20 percent difference) as through slabs of equivalent mass.
- The MORSE⁵ three-dimensional Monte Carlo computer program provides a reliable means of accurately predicting neutron attenuation through shielding configurations consisting of one or more rows of B₄C-filled tubes.
- The DOT⁶ two-dimensional and ANISN⁷ one-dimensional discrete ordinates (transport theory) computer programs provide reliable means of accurately predicting neutron attenuation through shielding configurations consisting of two or more rows of B₄C-filled tubes. These programs provide satisfactory results for one row of tubes — adequate for design purposes.
- The major significance of these findings is that in-vessel radial shield designs fabricated with two or more rows of rods provide neutron shielding properties equivalent to those of solid slabs of the same material and mass. The rodDED shields are more easily fabricated, assembled, and are structurally more sound. The effects of unequal heat generation and neutron irradiation damage are less significant in a rod-type shield. It is expected that very significant cost reductions result from the use of rod type in-vessel radial shields as opposed to slab shields. Finally, the knowledge that neutron attenuation properties of rod-type shields may be analyzed by one- and two-dimensional computer programs means that the cost of such analyses does not exceed that for slab type shields.

3. DESCRIPTION OF TECHNICAL EFFORT

The planned experimental and analytical studies programs are illustrated by Table 3-1 and Figure 3-1. The parts completed as of the date of writing this final quarterly report (April 1, 1973) are indicated. Referring to Table 3-1, it is planned to complete the following measurements prior to the end of FY-73 (June 30, 1973):

Configuration No.	Description
IC	24-in. axial shield (5-1/4-in. o.d. stainless steel rods) using the IV C modifier
ID	IV C modifier plus axial shield plus 30 in. sodium
V (A, B, C and D)	Radial shield — stainless steel versus B ₄ C slabs
VII D (b)	Bonner ball radial traverses for 1, 2 and 3 3.8 in. B ₄ C slabs

Measured results will be reported by ORNL in their fast reactor experimental shielding monthly progress reports.

Table 3-1
EXPERIMENTAL PROGRAM FOR GENERAL ELECTRIC
AXIAL AND RADIAL SHIELD MEASUREMENTS AT TSF*

Configuration Number And Description	Neutron Measurements ^(a)					
			Bonner Balls			
			Axial Traverse ^(b)		Radial Traverse ^(c)	
	NE-213	H ₂ Counter	All 9 Balls	3, 6, and 10 in.	All 9 Balls	3, 6, and 10 in.
I. Axial Shield						
A. Partial Modifier (18 in. SS only)	☑, Close ^(d)	☑, 12 in.	—	☑, 12 to 144 in.	—	—
B. Full Modifier (18 in. SS + 48 in. Na + 1/8-in. Boral)	☑, Close ^(d)	☑, 12 in.	☑, 12 to 144 in.	—	—	12 in.; 0 to 36 in. on each side of ☑. 144 in.; 0 to 144 in. on one side of ☑.
C. Full Modifier + 24 in. Axial Shield (5-1/4 in. OD SS rods)	—	☑, 12 in.	—	☑, 12 to 144 in.	—	12 in.; 0 to 36 in. on each side of ☑. 144 in.; 0 to 144 in. on one side of ☑.
D. Full Modifier + Axial Shield + 30 in. Na	—	—	—	☑, 12 to 144 in.	—	—
II. Radial Spectrum Modifier ^(f)						
A. Radial Modifier ^(e)	☑, Close ^(d) 45°, Close	☑, 12 in.	—	☑, 12 to 144 in.	—	12 in.; 0 to 36 in. on each side of ☑. 144 in.; 0 to 144 in. on one side of ☑.
III. Axial Shield With Radial Spectrum Modifier						
A. Axial Shield with Radial Spectrum Modifier	☑, Close ^(d)	☑, 12 in.	—	☑, 12 in.	—	—

Experiments
Completed

Experimental
Data Received

Experiments
Partially
Completed

GEAP-13822-5

Table 3-1 Continued

Configuration Number And Description	Neutron Measurements ^(a)					
	NE-213	H ₂ Counter	Axial Traverse ^(b)		Radial Traverse ^(c)	
			All 9 Balls	3, 6, and 10 in.	All 9 Balls	3, 6, and 10 in.
IV. Radial Blanket Measurements for Westinghouse						
A. Radial Modifier + 5 in. Radial Blanket ^(e)	ϕ, Close ^(d) 45°, Close	ϕ, 12 in.	—	ϕ, 12 to 144 in.	—	12 in.; 0 to 36 in. on each side of ϕ. 144 in.; 0 to 144 in. on one side of ϕ
B. Radial Modifier + 10 in. Radial Blanket ^(e)	ϕ, Close ^(d) 45°, Close	ϕ, 12 in. 45°, 144 in.	—	ϕ, 12 to 144 in.	—	12 in.; 0 to 36 in. on each side of ϕ. 144 in.; 0 to 144 in. on one side of ϕ
C. Radial Modifier + 10 in. Radial Blanket + 5 in. Reflector ^(e)	ϕ, Close ^(d) 45°, Close	ϕ, 12 in.	—	ϕ, 12 to 144 in.	—	12 in.; 0 to 36 in. on each side of ϕ. 144 in.; 0 to 144 in. on one side of ϕ
V. Radial Shield—Stainless Steel versus B ₄ C Slabs						
A. IV.C. Modifier + 10 in. B ₄ C Slabs	ϕ, Close ^(d)	ϕ, 12 in.	—	ϕ, 12 to 144 in.	—	12 in.; 0 to 36 in. on each side of ϕ. 144 in.; 0 to 144 in. on one side of ϕ
B. IV.C. Modifier + 10 in. B ₄ C Slabs + Axial Shield	—	—	—	ϕ, 12 to 144 in.	—	12 in.; 0 to 36 in. on each side of ϕ. 144 in.; 0 to 60 in. on one side of ϕ
C. IV.C. Modifier + 10 in. SS + Axial Shield	—	—	—	ϕ, 12 to 144 in.	—	12 in.; 0 to 36 in. on each side of ϕ. 144 in.; 0 to 60 in. on one side of ϕ

Table 3-1 Continued

Configuration Number And Description		Neutron Measurements ^(a)					
		NE-213	H ₂ Counter	Bonner Balls		All 9 Balls	3, 6, and 10 in.
				Axial Traverse ^(b)			
				Radial Traverse ^(c)			
				All 9 Balls		3, 6, and 10 in.	
D. IV.C. Modifier + 10 in. SS		☒, Close ^(d)	☒, 12 in.	—	☒, 12 to 144 in.	—	12 in.; 0 to 36 in. on each side of ☒. 144 in.; 0 to 144 in. on one side of ☒.
VI. Radial Shield Measurements for Westinghouse							
h {	A. IV.C. + 5 in. Inconel + 12 in. Na						
	B. VI.A. + 5 in. SS						
	C. Radial Modifier + 10 in. Radial Blanket + 2-1/2 in. Inconel						
	D. Radial Modifier + 10 in. Radial Blanket + 5 in. Inconel						
	E. IV.C. + 5 in. Inconel + 5 in. Stainless Steel						
VII. Radial Shield—B ₄ C Rods and Slabs							
A. Radial Modifier + 5-1/4 in. OD B ₄ C Rods							
a. 1 and 3 rows**		☒, Close ^(d)	☒, 12 in.	—	☒, 12 to 144 in.; repeat 72 to 144 in. foreground; measure background at 72 and 144 in.	—	12 in.; 0 to 36 in. on each side of ☒. 72 in.; 0 to 72 in. on one side of ☒. 144 in.; 0 to 144 in. on one side of ☒.
b. 2 rows**		—	—	—	☒, 12 to 144 in.; repeat 72 to 144 in. foreground; measure background at 72 and 144 in.	—	12 in.; 0 to 36 in. on each side of ☒. 72 in.; 0 to 72 in. on one side of ☒. 144 in.; 0 to 144 in. on one side of ☒.
B. Radial Modifier + 2 in. OD B ₄ C Rods							
a. 1 row		☒, Close ^(d)	☒, 12 in.	—	☒, 12 to 144 in.	—	12 in.; 0 to 36 in. on each side of ☒.

**Also, radial traverses were measured with Hornyak buttons 2.6 inches behind the test configurations—1 and 2 rows of rods vs 1 and 2 slabs. Experimental data have been received for the latter.

Table 3-1 Continued

Configuration Number And Description	Neutron Measurements ^(a)					
	NE-213	H ₂ Counter	Axial Traverse ^(b)		Radial Traverse ^(c)	
			All 9 Balls	3, 6, and 10 in.	All 9 Balls	3, 6, and 10 in.
C. Radial Modifier + 1 in. OD B ₄ C Rods a. 4 rows	—	—	—	☐ ☐, 12 to 144 in.	—	☐ 12 in.; 0 to 36 in. on each side of ☐. 144 in.; 0 to 144 in. on one side of ☐.
D. Radial Modifier + 0.6 in. and 3.8 in. B ₄ C Slabs a. 1, 2, 3, and 4 0.6 in. slabs	—	—	—	☐ ☐, 12 to 144 in. Completed for 3 and 4 slabs only	—	☐ Completed for 3 slabs only 12 in.; 0 to 36 in. on each side of ☐. 144 in.; 0 to 144 in. on one side of ☐.
b. 1, 2, and 3 3.8 in. slabs***	☐ ☐, Close ^(d) ****	☐, 12 in.	—	☐ ☐, 12 to 144 in. Completed for 1 slab only	—	12 in.; 0 to 36 in. on each side of ☐ 72 in.; 0 to 72 in. on one side of ☐
E. Radial Modifier + 5-1/4 in. OD B ₄ C Rods (3 rows) a. 6 in. SS reflector ~4 in. behind B ₄ C b. Axial shield ~4 in. behind B ₄ C	} Traverse with 3 in. Bonner ball adjacent to back of B ₄ C rods to locate peaks and valleys that may exist.					

*** Also, radial traverses were measured with Hornyak buttons 2.6 inches behind the test configurations—1 and 2 rows of rods vs 1 and 2 slabs. Experimental data have been received for the latter.

**** Measured at 10 feet behind and 10 feet off center for three 3.8-inch slabs.

- Measurements unless otherwise specified will be in the horizontal plane.
- Axially out beam centerline from back of configuration.
- Radially from beam centerline.
- As close to the back of the sample as the upper limit of the gamma ray sensitivity of the scintillator will permit.
- Gamma ray spectral measurements will be made (as part of the Westinghouse LMFBR shielding program) at one angle between 45° to 60° from centerline at a distance from the back of the slab to be determined by background considerations.
- This radial spectrum modifier was specified by Westinghouse for use with their radial shield measurements of neutron and gamma fluxes. Its use is specified in the GE program (III, V, and VII) since it also provides a satisfactory spectrum for the GE experiments and reduces overall TSF experimental time and costs.
- IV is a part of the AEC-funded Westinghouse LMFBR shielding base program. A description of the measurements is included here since the radial modifier + 10 in. radial blanket + 5 in. SS reflector will be used as a spectrum modifier for Part V of the GE program.
- VI is also a part of the AEC-funded Westinghouse LMFBR shielding base program. A description of the configurations to be measured is included for general information and schedule interaction.

C COMPLETION OF ANALYSIS
 --- COMPARISON & EVALUATION
 — 1D CALCULATIONS
 == 2D CALCULATIONS
 === 3D CALCULATIONS
 ■ COUPLED 2D + 3D CALCULATIONS

	J	A	S	O	N	D	J	F	M	A	M	J
I. AXIAL SHIELD												
A. PARTIAL AXIAL MODIFIER = 18 in. STAINLESS STEEL												
B. AXIAL MODIFIER = 18 in. STAINLESS STEEL + 48 in. Na + 1/8 in. BORAL												
C. AXIAL MODIFIER + AXIAL SHIELD												
D. AXIAL MODIFIER + AXIAL SHIELD + 30 in. Na												
II. RADIAL MODIFIER												
A. 4 in. STAINLESS STEEL + 6 in. Na + 1 in. BORAL												
III. RADIAL MODIFIER + AXIAL SHIELD												
A. 4 in. STAINLESS STEEL + 6 in. Na + 1 in. BORAL + AXIAL SHIELD												
V. RADIAL SHIELD — STAINLESS STEEL vs B₄C SLABS												
A. W ASSEMBLY** + 10 in. B ₄ C SLAB												
B. AS ABOVE + AXIAL SHIELD												
C. AS ABOVE EXCEPT 10 in. STAINLESS STEEL REPLACES 10 in. B ₄ C SLAB												
D. W ASSEMBLY** + 10 in. STAINLESS STEEL SLAB												
VII. RADIAL MODIFIER + B₄C SLABS OR RODS												
A a. 1 ROW 5—1/4 in. RODS												
A a. 3 ROWS 5—1/4 in. RODS												
A b. 2 ROWS 5—1/4 in. RODS												
C a. 1 ROW 1 in. RODS												
C a. 4 ROWS 1 in. RODS												
D a. 1 ROW 0.6 in. SLABS												
D a. 2 ROWS 0.6 in. SLABS												
D a. 3 ROWS 0.6 in. SLABS												
D a. 4 ROWS OF 0.6 in. SLABS												
D b. 1 ROW 3.8 in. SLABS												
D b. 2 ROWS 3.8 in. SLABS												
D b. 3 ROWS 3.8 in. SLABS												
E a. 3 ROWS 5—1/4 in. RODS + 6 in. STAINLESS STEEL SLAB												
E b. 3 ROWS 5—1/4 in. RODS + AXIAL SHIELD												

*ORNL, WITHIN FY-73 FUNDING LIMITATIONS, PLANS TO ANALYZE A NUMBER OF THE CONFIGURATIONS TESTED.

**W ASSEMBLY = 4 in. STAINLESS STEEL + 6 in. Na + 1 in. BORAL + 10 in. BLANKET + 5 in. REFLECTOR.

Figure 3-1. Schedule for Completing the GE-BRD Portion* of the PA-62 Analytical Program

3.1 EXPERIMENTAL MEASUREMENTS

3.1.1 Experimental Hardware

3.1.1.1 Axial Shielding Test Assembly (Figure 3-2)

The axial test hardware was delivered to ORNL in parts for assembly and sodium filling. Sodium filling was completed by ORNL personnel on May 23, 1972. The axial shielding test assembly was ready for testing on June 1, 1972.

3.1.1.2 Radial Shielding Tubular and Homogeneous Slab Test Assemblies (Figure 3-3, Figure 3-4, and Figure 3-5)

Test assemblies were fabricated at GE's Nuclear Energy Division facilities in San Jose, California. The full shipment of radial assemblies was delivered to ORNL on June 30, 1972.

Material compositions and densities for all test assemblies were carefully compiled during the procurement of materials and the fabrication and assembly of the components. Detailed tabulations of this data are provided in Section 3.2 of reference 2.

3.1.1.3 Quality Assurance

Quality assurance requirements for the project are defined in BR QCI 900, "Quality Assurance Verification Plan for PA-62 Axial and Radial Experiments." The concept behind this planning effort was "direct verification" of quality assurance requirements. A total of nine Quality Control Instructions (QCI's) were prepared to specify detailed inspection requirements which defined criteria for accepting or rejecting materials, hardware, and assemblies used in the experiments. Detailed descriptions of the QA activities carried out in support of this program are provided in Section 3.4 of references 1 and 2.

3.1.2 Test Facility — ORNL's Tower Shielding Reactor

All measurements were carried out using the Tower Shielding Reactor at ORNL as a neutron source. This part of the program was under the operational control of the Neutron Physics Division at ORNL. Figures 3-6 and 3-7 schematically illustrate the test facility and typical experimental configuration for this program. The TSR-II neutron source was represented by collapsing the 100-group ORNL measured source spectrum to 24-energy groups as discussed in reference 1. Table 3-2 shows the GE-BRD 24-group representation of this source. Figure 3-8 shows schematically the TSR source geometry which was approximated by S_{16} quadrature for a comparison¹ neutron attenuation calculation through 2.5 feet of Na.

Measurements were made with the following three principal types of neutron detectors:

Bonner Balls — 3-, 6- and 10-inch diameters

NE-213 Neutron Spectrometer (0.8 MeV to 15 MeV)

Hydrogen Counter Neutron Spectrometer (10 keV to 1.5 MeV)

Both radial and axial traverses were made (Figure 3-6). Also, high energy neutron spectrum measurements were made with the NE-213 spectrometer.

3.1.3 Experimental Results

The first part of the PA-62 program required measuring the neutron attenuation through 18 inches of stainless steel originally planned for use as part of a spectrum modifier for the axial shield test.² Measured results were reported in August and September 1972.⁸ Additionally, by way of comparison to calculational results, the measured Bonner Ball responses and low energy spectral data are illustrated in Figures 3-21 through 3-30 of reference 3.

Figure 3-2. Axial Shielding Test Assembly

Figure 3-3. Radial Shielding Tubular Test Assembly

Figure 3-4. Homogeneous Slab Test Assembly

Figure 3-6. Schematic Diagram of Test Configuration and Radial Traverse Behind a Typical B₄C-Filled Tube Shield

Figure 3-7. Side View of Experimental Configuration for Radial Modifier Plus One 5.25-Inch B₄C Slab Unit

Figure 3-8. TSR-II Source Geometry

The second part of the experimental program required measuring neutron attenuation through boron carbide radial shields, consisting of either arrays of rods or simple slabs (Part VII, Table 3-1 and Figures 1-2 through 1-5). The results of these measurements are reported in reference 8. This series of experiments was shortened considerably after measurements taken on a single row of the largest (5-1/4-inch) diameter B₄C-filled tubes showed little streaming effect when the transmission for the single row of rods was compared to that for an equivalent homogeneous slab.

Radial traverses were taken at 12 inches behind the large diameter rods to see if variations in the flux due to the heterogeneities in the shield could be observed. The measured results gave little indication of ripple for any of the configurations whether the detector traverse was performed close behind the sample at 12 inches or as far away as 144 inches. The measurements at 144 inches should be the most sensitive to the uncollided flux and therefore should be the most likely position to indicate ripple.

Table 3-2
TSR-II 24-GROUP SOURCE SPECTRUM
(Revised October 4, 1972)

Group Number	Energy Band	Lethargy Increment (ΔU)	Percent of Total Flux	Neutron Flux* (n/cm ² -min-W)
1	10.0-16.5 MeV	0.5	0.12	3.1800(+2)
2	6.07-10.0 MeV	0.5	1.91	5.1480(+3)
3	3.68-6.07 MeV	0.5	6.93	1.8730(+4)
4	2.23-3.68 MeV	0.5	11.38	2.0740(+4)
5	1.35-2.23 MeV	0.5	9.30	2.5135(+4)
6	0.821-1.35 MeV	0.5	4.57	1.2355(+4)
7	498-821 KeV	0.5	3.62	9.7800(+3)
8	302-498 KeV	0.5	2.15	5.8150(+3)
9	183-302 KeV	0.5	1.68	4.5500(+3)
10	111-183 KeV	0.5	1.50	4.0540(+3)
11	67.4-111 KeV	0.5	1.36	3.6600(+3)
12	40.9-67.4 KeV	0.5	1.25	3.3800(+3)
13	24.8-40.9 KeV	0.5	1.15	3.1200(+3)
14	15.0-24.8 KeV	0.5	1.08	2.9300(+3)
15	9.12-15.0 KeV	0.5	1.00	2.7100(+3)
16	5.53-9.12 KeV	0.5	0.95	2.5700(+3)
17	3.35-5.53 KeV	0.5	0.90	2.4200(+3)
18	2.03-3.35 KeV	0.5	0.86	2.3300(+3)
19	1.01-2.03 KeV	0.7	1.14	3.0730(+3)
20	0.249-1.01 KeV	1.4	2.05	5.5380(+3)
21	55.6-249 eV	1.5	2.02	5.4450(+3)
22	12.4-55.6 eV	1.5	2.00	5.4160(+3)
23	0.625-12.4 eV	3.0	4.87	1.3148(+4)
24	Thermal	∞	36.22	9.7864(+4)
			Total 100.00	Total 2.7023(+5)

*As collapsed from 100-group structure given in Reference 2, and normalized to a power level of 1 watt. The values are given for a point on the centerline at the end of the collimator.

Measurements were also made with the 1/2-inch diameter Hornyak button behind one and two rows of the 5-1/4-inch B₄C rods. The Hornyak button response function is proportional to the "first collision dose," or KERMA — i.e., "kinetic energy released in matter." Therefore, it is primarily sensitive to fast neutrons.⁹ The Hornyak button data is illustrated in Figures 3-9 and 3-10. There is some indication of a ripple due to the heterogeneities. However, the effect is still rather small, being on the order of 10% even for fast neutrons.

Referring to Table 3-1, it is noted that measurements were not completed for the axial shield (Parts IC and ID) or for stainless steel versus B₄C slabs (Part V) as of the time of writing this report (4/73). However, ORNL has scheduled TSR operations to complete the experimental program prior to the end of FY-73 (June 30, 1973). The following measurements constitute the most significant and beneficial parts of the unfinished program:

Configuration No.	Description
IC	24-inch axial shield (5-1/4-inch o.d. stainless steel rods) using the IV C modifier.
ID	IV C modifier plus axial shield plus 30 inches of Na.
V (A, B, C and D)	Radial shield — stainless steel versus B ₄ C slabs.
VII D (b)	Bonner ball radial traverses for 1, 2 and 3 3.8-inch B ₄ C slabs.

ORNL will report the measured data for the balance of this experimental program in their fast reactor experimental shielding monthly progress reports.

3.2 ANALYTICAL STUDIES

The analytical program was carried out to assess the calculational uncertainties of the GE-BRD shielding design methods through comparisons of calculated versus measured results. This effort also included comparisons of analytical results using alternate cross sections and up to 100 neutron energy groups. The GE-BRD shielding design methods utilize 24 neutron energy groups.

The following analytical studies were accomplished:

- a. Selection of neutron cross section and other material data.
- b. Determination and verification of approximations to be used in simulating the TSR-II source.
- c. Evaluation of geometry and background approximations for analytical mock-up of experiments.
- d. Determination of spectrum modifier/diffuser materials and thicknesses.
- e. Comparison of transmitted neutron spectrum shape between GE ID-S_n 24-group calculation and ORNL DOT-III 50-group calculation through 2.5 feet of sodium.
- f. Adaptation of the MORSE Monte Carlo computer program for local use on an IBM 370 Model 165 computer.
- g. Collapsing of Bonner ball response data for 3-, 6-, and 10-inch Bonner balls from 100, 104, and 123 groups to a 24-group structure for use with the BRD 24-group cross section library.
- h. ID-S_n diffusion calculation comparison of GE-BRD and ORNL neutron cross sections for sodium.

Figure 3-9. Measured Hornyak Button Response for Radial Traverse at 2.6 Inches Behind One 5.25-Inch Rod-Slab or Homogeneous Slab

Figure 3-10. Measured Hornyak Button Response for Radial Traverse
at 2.6 Inches Behind Two 5.25-Inch Rod-Slabs or Homogeneous Slabs

- i. ID-S_n diffusion and transport calculation comparisons of GE-BRD and ORNL neutron cross sections for Type-304 stainless steel.
- j. Completion of the following calculations (Figure 3-1):
 - ID-S_n transport calculations for the partial axial modifier, full axial modifier, and radial modifier.
 - MORSE Monte Carlo calculations for the partial axial modifier (18 inches of Type-304 stainless steel).
 - DOT III calculations for three B₄C homogeneous slab test assemblies. Also, generation of neutron isoflux plots for these cases.
 - MORSE calculations for one B₄C rod test assembly with Al triflutes, and with latter replaced by Na and void triflutes.

3.2.1 Neutron Cross Sections and Other Material Data

The calculations described in this report were primarily based on a 24-group shielding cross section library which was compiled and edited at GE-BRD from several external sources.¹⁰⁻¹⁶ The ID-S_n and DOT III calculations used this neutron cross section library exclusively. The MORSE Monte Carlo calculations also used this library with one addition. That is, GE-BRD 24-group P₃ scattering matrices were used for all materials and in addition, ORNL point total cross sections were used but only for Type-304 stainless steel and for neutron energies greater than 10 keV. The point total cross sections were included in the calculations through an option in MORSE which allows use of an 06R point total cross section tape in addition to multigroup cross sections. The point total cross sections are used in sampling collision sites and in next-flight estimation; they are not used in calculating the uncollided flux since this component is a very small part of the total flux.

Table 3-3 gives a complete list of densities and compositions for materials used in the DOT and MORSE calculations. These densities and compositions were determined from vendors specifications, chemical and spectrographic analysis done at the General Electric Nuclear Energy Division, Manufacturing procedures, and data provided by ORNL. It should be noted that the B₄C density of 2.463 gm/cm³ is a theoretical (maximum) density based on mixing 3 different mesh sizes of B₄C particles. For purposes of these calculations, the actual density resulting from compacting of each separate homogeneous slab configuration was used. These values turned out to be approximately 70 percent. For the MORSE calculations involving a single row of 5.25-inch diameter B₄C-filled tubes, an average B₄C density of 1.6833 gm/cc (68.36 percent of theoretical) was used.

In order to use ORNL Bonner ball response functions^{17,18} in 24-group discrete ordinates or Monte Carlo calculations, it was necessary to collapse the data for 3-, 6-, and 9.86-inch diameter Bonner spheres from 100-, 104-, and 123-energy groups to the GE-BRD 24-group structure. The determination of values for the reduced number of groups was done using log-log interpolation and extrapolation.

For the 3-inch diameter sphere, ORNL 194-group data were used for neutron energies between 1.855 eV and 3.679 MeV. For energies below or above this range, extrapolations were made using a shape derived from the ORNL 100-group data. For the 6-inch diameter sphere, ORNL 123-group data were used for energies above 101.3 eV with an extrapolation below this energy made using the shape of the 100-group data. Collapsed values for the 9.86-inch diameter sphere used ORNL 100-group data exclusively. Table 3-4 shows the results of this collapsing procedure.

3.2.2 Analytical Methods

The analysis performed during the execution of this program used both GE and ORNL computer programs. For the pre-analysis and spectrum modifier design, the GE version of the one-dimensional neutron transport code ANISN was used. Also, the general geometry version of the MORSE/SAMBO Monte Carlo code was adapted for

Table 3-3
DENSITIES AND COMPOSITIONS OF MATERIALS USED
IN DOT AND MORSE CALCULATIONS

Material	Density (gm/cc)	Atom Density
Sodium	0.945	Na = 2.4766(-2)atoms/barn-cm
Aluminum	2.702	Al = 6.0366(-2)atoms/barn-cm
B ₄ C	2.463	B-10 = 2.0902(-2)atoms/barn-cm C(a) = 1.1318(-1)atoms/barn-cm
Boral(b)	2.3	B-10 = 6.9765(-3)atoms/barn-cm C(a) = 3.6908(-2)atoms/barn-cm Al = 3.3370(-2)atoms/barn-cm
Type-304 SS(b)	7.88	C = 1.8578(-4)atoms/barn-cm Si(c) = 1.1405(-3)atoms/barn-cm Cr = 1.6719(-2)atoms/barn-cm Mn = 7.7777(-4)atoms/barn-cm Fe(d) = 6.0238(-2)atoms/barn-cm Ni = 6.9869(-3)atoms/barn-cm Cu = 1.9427(-4)atoms/barn-cm Mo = 1.4846(-4)atoms/barn-cm
Plain Polyethylene	0.900	H = 7.7292(-2)atoms/barn-cm C = 3.8646(-2)atoms/barn-cm
Borated Polyethylene	0.940	H = 7.6682(-2)atoms/barn-cm C(a) = 4.0441(-2)atoms/barn-cm B-10 = 5.1836(-4)atoms/barn-cm
Concrete	2.400	Si(e) = 4.3500(-3)atoms/barn-cm Ca = 9.7700(-3)atoms/barn-cm O = 4.4470(-2)atoms/barn-cm C = 9.9600(-3)atoms/barn-cm H = 6.4200(-3)atoms/barn-cm Mg = 2.2000(-3)atoms/barn-cm Al = 4.5000(-4)atoms/barn-cm K = 1.1000(-4)atoms/barn-cm Fe = 1.1000(-4)atoms/barn-cm

(a) Boron-11 is included in this density.

(b) Modified slightly from composition given in Reference 1, page 3-37.

(c) Sulphur and phosphorus are included in this density.

(d) Cobalt is included in this density.

(e) Sulphur is included in this density.

Table 3-4
BONNER BALL NEUTRON RESPONSE FUNCTIONS
FOR 3-INCH, 6-INCH, AND 9.86-INCH DIAMETER SPHERES

Group	Energy Range		Bonner Ball Responses ^(a) (Counts/Incident Neutron-cm ⁻²)		
			3-In. Diameter	6-In. Diameter	9.86-In. Diameter
1	10.0	— 16.5 MeV	0.004	0.282	0.635
2	6.07	— 10.0	0.007	0.456	0.917
3	3.68	— 6.07	0.012	0.708	1.106
4	2.23	— 3.68	0.019	0.964	1.177
5	1.35	— 2.23	0.035	1.267	1.169
6	0.821	— 1.35	0.053	1.492	1.003
7	498	— 821 keV	0.077	1.570	0.782
8	302	— 498	0.116	1.573	0.596
9	183	— 302	0.142	1.496	0.462
10	111	— 183	0.188	1.502	0.371
11	67.4	— 111	0.238	1.318	0.314
12	40.8	— 67.4	0.259	1.340	0.276
13	24.8	— 40.8	0.294	1.295	0.252
14	15.0	— 24.8	0.333	1.219	0.234
15	9.12	— 15.0	0.367	1.195	0.220
16	5.53	— 9.12	0.414	1.225	0.209
17	3.35	— 5.53	0.501	1.139	0.192
18	2.03	— 3.35	0.445	1.270	0.200
19	1.01	— 2.03	0.538	1.221	0.183
20	0.249	— 1.01	0.670	1.138	0.167
21	55.6	— 249 eV	0.873	1.070	0.141
22	12.4	— 55.6	1.110	1.002	0.126
23	0.625	— 12.4	1.402	0.729	0.088
24	0.0	— 0.625	0.764	0.205	0.248

^(a) Values derived from ORNL 100, 104 and 123 Energy Group Data.

local use on an IBM 370 Model 165 computer to allow testing of GE-BRD cross-section sets, checkout of geometry input, and to gain familiarity with the program prior to using it at ORNL. The primary analysis of the homogeneous slab experiments was done using the current (November 1972) version of the ORNL two-dimensional discrete ordinates code, DOT, along with its auxiliary programs, UNCL, SPACETRAN, and ISOPLOT. The primary analysis of the heterogeneous rod-array experiments was done using the new version (combinatorial geometry) of MORSE with its auxiliary programs, XCHECKR, PICTURE, and COLDEN. In order to utilize the most up-to-date versions of DOT and MORSE, the primary analysis was carried out using the ORNL IBM 360 Model 91 computer and in consultation with personnel of the ORNL Neutron Physics Division. A brief summary of the major features of the various calculations is given below.

For the ANISN reactor shield design calculations, the following data and options were used.

- GE-BRD 24-group P_3 neutron cross sections
- Buckling correction factors determined from 2-D calculations using the GE version of DOT.
- Cylindrical geometry
- S_8 Angular quadrature

- Reflective interior and vacuum exterior boundary conditions
- Fixed boundary source

ANISN calculations for the neutron spectrum modifier/diffuser and for the homogeneous slab test configurations included the following data and options.

- GE-BRD 24-group P_3 neutron cross sections
- GE-BRD 24-group Bonner ball response functions
- Buckling correction factor for leakage (plane height and depth input as 152.4 cm)
- Void correction factor (half-height input as 82.5 cm)
- Plane geometry
- S_{16} angular quadrature
- Vacuum boundary conditions on both sides of the experimental test assembly.
- Shell source

The two-dimensional DOT homogeneous slab calculations included the following data and options.

- GE-BRD 24-group P_3 neutron cross sections.
- GE-BRD 24-group Bonner ball response functions
- Cylindrical geometry (R-Z) with symmetry about the Z-axis (beam ϕ)
- S_{100} angular quadrature
- First collision source calculated by UNCL with the source directed down the Z-axis through the collimator, and into the modifier.
- Approximately nine inches of concrete on all sides of the test assembly except behind the last slab and in the collimator.
- Air attenuation neglected
- Total flux, 24-group fluxes, 3-, 6-, and 10-inch diameter Bonner ball responses calculated by SPACETRAN for 30 detector locations.
- Total flux and few-group isoflux plots made using ISOPLOT.

The MORSE calculations used to test the adequacy of the GE-BRD 24-group P_3 Type-304 stainless steel cross section set for multigroup Monte Carlo Transport used the following data and options.

- GE-BRD 24-group P_3 cross sections for Type-304 stainless steel.
- ORNL point total cross sections for Type-304 stainless steel were used in sampling collision sites and in next flight estimation.

- GE-BRD 24-group Bonner ball response functions
- Finite (three-dimensional) slab geometry
- Surface crossing estimation
- Next-flight estimation
- Energy and region dependent path stretching
- Russian roulette
- 9-point detectors, 4-surface detectors
- 7-detector locations
- Effects of concrete shield blocks surrounding the Type-304 stainless steel were neglected.
- Air attenuation neglected
- 2500-particle histories/run (50 particles/batch, 50 batches)

The MORSE heterogeneous rod-array calculations included the following data and options.

- GE-BRD 24-group P_3 cross sections for all materials.
- ORNL point total cross sections for Type-304 stainless steel were used in sampling collision sites and in next flight estimation.
- GE-BRD 24-group Bonner ball response functions.
- Combinatorial (three-dimensional) geometry with 41 bodies.
- Collision density fluence estimation.
- Track length per unit volume estimation.
- Next flight estimation.
- Energy and region dependent path stretching.
- Russian roulette.
- Source enhancement effects due to multiple scattering in the concrete collimator were neglected.
- Effects of concrete shield blocks surrounding the test assembly were neglected.
- Air attenuation neglected.
- 25-point detectors (total flux, 24-group fluxes, 3-, 6-, and 10-inch diameter Bonner ball responses).
- 13-detector locations.
- 2000-particle histories/run (100 particles/batch, 20 batches).

A summary of the principal calculations completed under this program are given in the following paragraphs. Emphasis is directed to the significance of the results, omitting details of the calculational methods (model, procedure, geometric details, etc., which were previously described in references 1, 2, 3 and 4).

3.2.3 Spectrum Modifier Designs (References 1 and 2)

Calculations showed that a close approximation to an LMFBR axial shield reference neutron spectrum could be obtained using a modifier consisting of 18 inches of stainless steel plus 48 inches of sodium plus 1/8 inch of Boral (reference 2).

A radial spectrum modifier option was adopted which reduced ORNL TSF time required for spectrum modifier measurements. This option consisted of selecting the Westinghouse ARD modifier (4 inches of Type-304 stainless steel plus 6 inches sodium plus 1 inch Boral) followed by 2 rows of W blankets plus reflector as the configuration (Part IV C, Table 3-1) modifying the spectrum for input to the B₄C versus stainless steel slab configurations (Part V, Table 3-1). This option takes advantage of the fact that the GE and Westinghouse neutron spectra at the blanket-reflector interface are very similar.¹⁹ For the B₄C rod versus slab measurements (Part VII, Table 3-1), the Westinghouse ARD modifier only was used. This was based on the logic that the neutron streaming effects would be enhanced by selecting a spectrum that was "harder" (higher energy) than the heterogeneous B₄C rod-type shields would nominally experience. The use of these modifier options, as opposed to using a modifier of different material and thicknesses, resulted in a savings in TSF time of about 1 month.

3.2.4 Calculational Comparison of GE-BRD and ORNL Neutron Cross Sections for Sodium and Type-304 Stainless Steel³

Since both the axial and radial modifiers contain sodium and stainless steel slabs, it was deemed advisable to test the adequacy of the GE-BRD 24-group P₃ scattering matrices before performing extensive analyses of the experiments. This was possible because ORNL had done sodium and Type-304 stainless steel benchmark experiments and calculations at the Tower Shielding Facility (TSF). ORNL provided 100-group P₃ sodium (MOD 0 ENDF/B-III) and Type-304 stainless steel cross section sets — both sets benchmark tested.

The ORNL 100-group set and the GE-BRD 24-group set were used to predict the neutron transmission through 30 inches of sodium by running the GE version of ANISN in the diffusion mode (thereby using P₀ and P₁ cross sections). The total flux values from the two diffusion calculations agreed quite closely with the values at the 30-inch thickness differing by only about 6% — the higher result being given by the 24-group calculation. This difference in calculated total flux is due not only to the number of groups used in the calculations but also due to the fact that the two libraries came from different data sets. Also, the 100-group sodium cross section set includes the effects of impurities in the sodium (400-ppm calcium, 200-ppm oxygen, 150-ppm potassium, and 60-ppm hydrogen) whereas the GE-BRD 24-group set represents pure sodium.

The GE 24-group Type-304 stainless steel cross sections were evaluated against ORNL's 100-group P₃ set by comparing the neutron transport through 18 inches of stainless steel calculated by each set (one-dimensional neutron diffusion and transport modes). As in the case of the sodium calculations, the 24-group and 100-group results agreed satisfactorily. There was a difference of only about 2% in the total neutron flux predicted by the two cross section sets at the exit face of the 18-inch thickness slab for the diffusion calculation. The difference in the total flux values at the exit face for the transport calculations was about 11%. In both diffusion and transport calculations the GE-BRD 24-group library gave the higher result. It should be emphasized that these differences in calculated total flux are due not only to the number of groups used in the calculations but also due to the fact that the GE-BRD and ORNL Type-304 stainless steel cross section libraries come from different data sets.

3.2.5 Calculated versus Measured Neutron Attenuation Data Through 18 Inches of Type-304 Stainless Steel³

Figures 3-11, 3-12, and 3-13 illustrate comparisons of measured and calculated neutron attenuation through 18 inches of Type-304 stainless steel. These figures illustrate comparisons of measured versus calculated responses for 3-, 6-, and 10-inch diameter Bonner balls. For the one-dimensional calculational results, transformation factors were

Figure 3-11. Calculated and Measured 3-inch Bonner Ball Response Behind 18 inches of Type-304 Stainless Steel

Figure 3-12. Calculated and Measured 6-inch Bonner Ball Response Behind 18 inches of Type-304 Stainless Steel

required and applied to the computed raw data.³ The agreement with experimental results is reasonable but not exceptionally good—especially so for the low energy spectrum comparison (Figure 3-14).

On the other hand, the data from the 24-group Monte Carlo calculation agrees reasonably well with the experimental data. The 24-group response values are seen to be consistently higher than the 100-group values by a factor of about 1.3 for the 3-inch diameter Bonner ball and a factor of about 1.6 for the 6- and 10-inch Bonner balls. The neutron energy spectrum comparisons (Figures 3-15 and 3-16) show good agreement for both the 24- and 100-group calculations.

The overall significance of these calculated versus measured results is that they show the methods and cross section data are reliable for predicting neutron attenuation through large thicknesses of stainless steel. Also, two-dimensional methods are much superior to one-dimensional methods due to the requirement of applying approximate transformation factor corrections to the 1D results.

3.2.6 Calculated versus Measured Neutron Attenuation Through B_4C Rods and Slabs⁴

The most significant finding from the experimental and calculational results obtained from this program to date is that the neutron streaming effect through the heterogeneous B_4C rod shields is essentially negligible. This result is true even for a single row of the largest diameter (5-1/4-inch) B_4C -filled tubes. This result was first observed from radial traverse measurements behind a single row of the large B_4C -filled tubes (Part 3.1.3, Figures 3-9 and 3-10). Calculational results have verified the measured data.

Figure 3-13. Calculated and Measured 10-inch Bonner Ball Response Behind 18 inches of Type-304 Stainless Steel

Figure 3-14. Comparison of Unfolded Hydrogen Counter Spectra with Transformed 1D-S_n Calculations for 18 inches of Type-304 Stainless Steel

Figure 3-15. Comparison of Unfolded Hydrogen Counter Spectra with MORSE 24-Group Calculations for 18 inches of Type-304 Stainless Steel

Figure 3-16. Comparison of Unfolded Hydrogen Counter Spectra with MORSE 100-Group Calculations for 18 inches of Type-304 Stainless Steel

3.3 MORSE CALCULATIONS FOR ONE ROW OF 5.25-INCH B₄C Rods

Figures 3-17, 3-18, and 3-19 show a comparison of MORSE Monte Carlo calculated and experimental Bonner ball results for radial traverses at 12, 72, and 144 inches behind one row of 5-1/4-inch B₄C rods. Inspection of these figures shows that the agreement between calculation and experiment is reasonably good. Calculated results are generally higher than experiment by factors ranging from 1.0 to 1.9. This comparison indicates that calculations somewhat under-predict the actual neutron attenuation. However, for both the measurements and experiments, there are no significant peaks in responses due to uncollided neutron streaming through the triflute regions (between the B₄C-filled tubes). This is true even at 144 inches behind the shield rods where the responses should be the most sensitive to the uncollided flux and therefore should be the most likely position to indicate ripple.

Figure 3-20 shows a comparison of MORSE calculated and NE-213 spectrometer results for the neutron spectrum at a detector point located 10 feet off centerline and 10 feet back of the single row of 5-1/4-inch o.d. B₄C rods. The NE-213 data correspond to neutron energies greater than about 1 MeV (GE-BRD energy groups 1 to 6). The MORSE calculated value for the integral over this energy range appears to be higher than the measured value by a factor of about 1.7. This result tends to agree with the difference between calculated and measured Bonner ball responses.

One of the main purposes of these calculations was to determine the amount of neutron streaming through the triflute regions between adjacent B₄C rods. This information could then be used to evaluate the effectiveness of a rod-type core radial shield. In performing this analysis, there are some differences between experimental, Monte Carlo model, and actual core radial rod-type shields which should be considered. The triflutes used in the experiment were made of aluminum for convenience of construction and because aluminum has neutron total flux attenuation properties similar to sodium which is the material which would be present in the core radial shield. Also, the aluminum triflutes used in the experiment were equally concave on all three of their longitudinal surfaces and therefore constituted a volume fraction of 0.093 of the rod array rather than a volume fraction of 0.2146 which was actually used in the Monte Carlo calculations. In order to determine the effect on neutron attenuation introduced by these differences, the Monte Carlo calculations were done for the three following cases, i.e.,

1. with aluminum triflutes
2. with sodium triflutes, and
3. with void triflutes.

Figures 3-21, 3-22, and 3-23 show the results of these calculations for detector locations representative of radial traverses at 12 inches behind the rod array. Inspection of these figures reveals that a small but noticeable increase in streaming occurs when the aluminum triflutes are replaced with voids. The calculations therefore predict experimental results that are intermediate between these two cases (voids versus aluminum). These figures also show DOT calculated results for the corresponding homogeneous slab configuration (one 5-1/4-inch equivalent homogeneous B₄C slab) — a kind of "reference" comparison. These calculational comparisons show that there is a relatively minor increase in transmitted flux in going from a homogeneous slab to the heterogeneous rod-type shield. This calculational result confirms the experimental results obtained from the measurements made at ORNL. (Subsection 3.1.3, Figures 3-9 and 3-10).

Finally, Figure 3-24 shows the relatively good agreement of DOT calculated Bonner ball responses 12 inches behind a 5-1/4-inch equivalent B₄C slab (measurement not yet completed) to measured responses behind one row of 5-1/4-inch o.d. B₄C-filled tubes.

3.4 DOT III CALCULATIONS FOR THREE B₄C HOMOGENEOUS SLAB TEST ASSEMBLIES

Three large DOT III (two-dimensional transport theory) calculations were completed to predict neutron attenuation through and Bonner ball responses behind the following B₄C slab shield test configurations:

1. The radial modifier plus one 5-1/4-inch B₄C slab,

Figure 3-17. Measured and MORSE Calculated Bonner Ball Response for Radial Traverse at 12 Inches Behind One Row of 5.25-Inch o.d. B_4C -Filled Tubes

Figure 3-18. Measured and MORSE Calculated Bonner Ball Response for Radial Traverse at 72 Inches Behind One Row of 5.25-Inch o.d. B_4C -Filled Tubes

Figure 3-19. Measured and MORSE Calculated Bonner Ball Response for Radial Traverse at 144 Inches Behind One Row of 5.25-Inch o.d. B_4C -Filled Tubes

Figure 3-20. Comparison of NE-213 Measured and MORSE Calculated Spectra Behind One Row of 5.25-Inch Diameter B_4C Rods at a Distance of 10 Feet and 10 Feet Off Centerline

Figure 3-21. Comparison of MORSE Calculated Bonner Ball Response for Radial Traverse at 12 Inches Behind 5.25-Inch o.d. Rod-Slab with Aluminum Triflutes and DOT Calculated Bonner Ball Response at 12 Inches Behind 5.25-Inch Homogeneous B_4C Slab

Figure 3-22. Comparison of MORSE Calculated Bonner Ball Response for Radial Traverse at 12 Inches Behind 5.25-Inch o.d. Rod-Slab with Sodium Triflutes and DOT Calculated Bonner Ball Response at 12 Inches Behind 5.25-Inch Homogeneous B_4C Slab

Figure 3-23. Comparison of MORSE Calculated Bonner Ball Response for Radial Traverse at 12 Inches Behind 5.25-Inch o.d. Rod-Slab with Void Triflutes and DOT Calculated Bonner Ball Response at 12 Inches Behind 5.25-Inch Homogeneous B_4C Slab

Figure 3-24. Measured Bonner Ball Response for Radial Traverse at 12 Inches Behind One Row of 5.25-Inch o.d. B_4C Filled Tubes and DOT Calculated Bonner Ball Response at 12 Inches Behind 5.25-Inch Homogeneous B_4C Slab

2. The radial modifier plus three 1-inch B₄C slabs, and
3. The radial modifier plus one 1-inch B₄C slab.*

The three-dimensional experimental configuration was approximated by two-dimensional R-Z geometry in the calculations. Refer to Subsection 3.3.3 of reference 4 for additional details of the calculational models.

These calculations used the GE-BRD 24-group neutron cross section library with P_3 scattering matrices, and used a special 100-angle quadrature set developed at ORNL for TSR shielding calculations. Five isoflux plots were obtained for each of the three configurations and are reproduced in reference 4.

Tables and other figures in reference 4 show comparisons of the DOT III calculated and experimental Bonner ball results for axial and radial traverses behind these three experimental configurations (except where measurements were not yet completed). Agreement is excellent for the 3-inch Bonner ball and reasonably good for the 6- and 10-inch diameter balls — calculated values being slightly higher for the latter. A comparison of calculated versus measured high energy ($E_n > 1$ MeV) neutron spectrum results is also illustrated. The calculated value for the integral over this energy range appears to be higher than the measured value by a factor of about 1.8. This result tends to agree with the differences between calculated and measured Bonner ball responses.

The significance of these DOT calculated versus measured results is that the methods and cross section data are shown to be reliable for design applications in predicting neutron attenuation through relatively large thicknesses of stainless steel and B₄C. The DOT III two-dimensional program is clearly superior to one-dimensional methods since approximate transformation factor corrections are not required to be applied to the two-dimensional calculational output.

*Each "1-inch" B₄C slab unit has an actual B₄C thickness of 0.6 inch. This gives it the same mass as a single row of 1-inch o.d. B₄C-filled tubes. The "5-1/4-inch" B₄C slab has an actual B₄C thickness of 3.755 inches which gives it the same mass as a single row of 5-1/4-inch o.d. B₄C-filled tubes.

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