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FAST REACTOR SHIELDING MONTHLY PROGRESS REPORT FOR MAY, 1972

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Neutron Physics Division

FAST REACTOR SHIELDING MONTHLY PROGRESS REPORT FOR MAY, 1972

189a No. 10028, Activity No. 40 01 61

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JULY 1972

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Monthly Progress Report

May 1972

FAST REACTOR SHIELDING

Activity No. 40 01 61
189a No. 10028

Approved by:

C. E. Clifford
Associate Director
Reactor and Weapons Radiation Shielding
Neutron Physics Division

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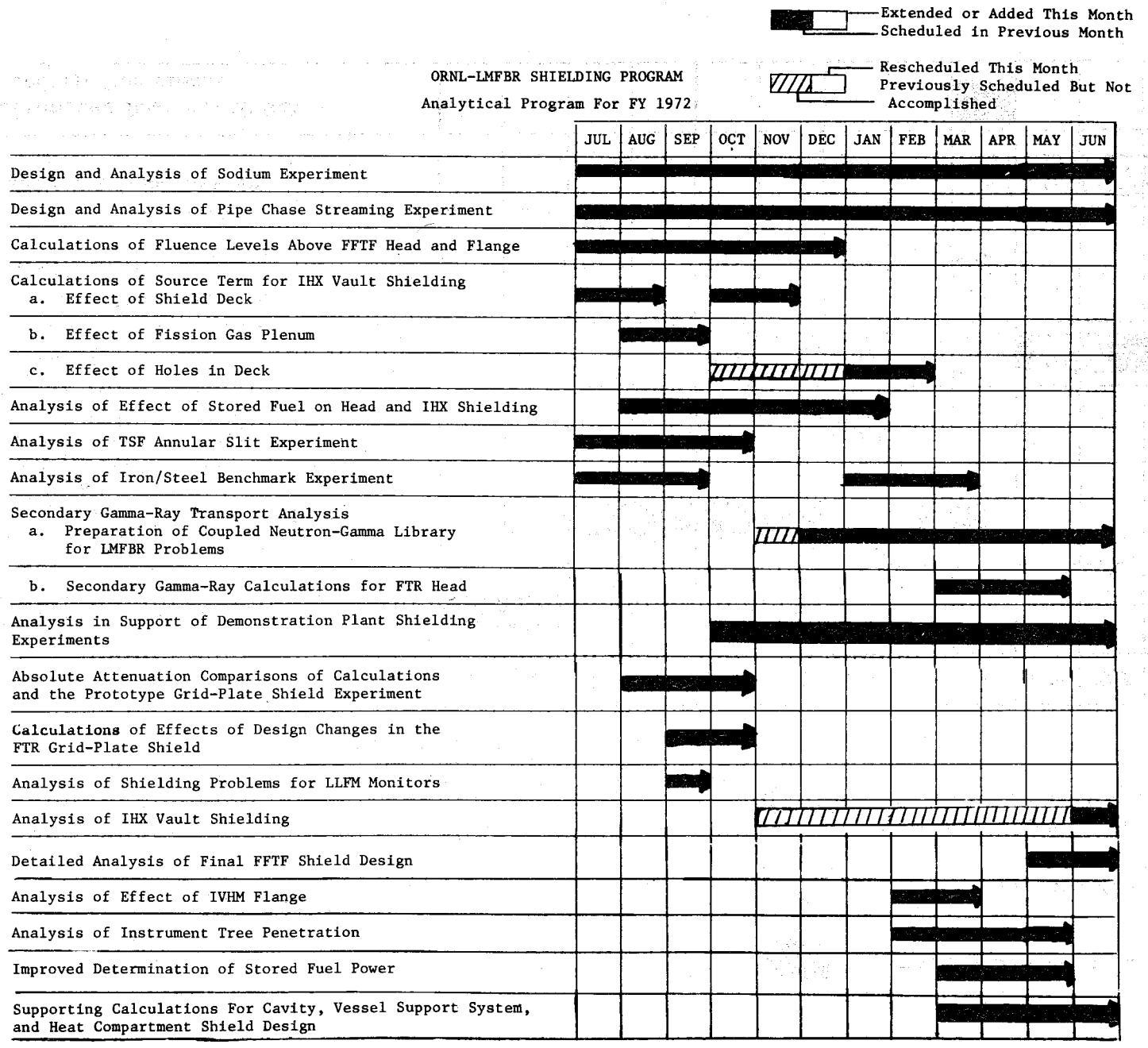
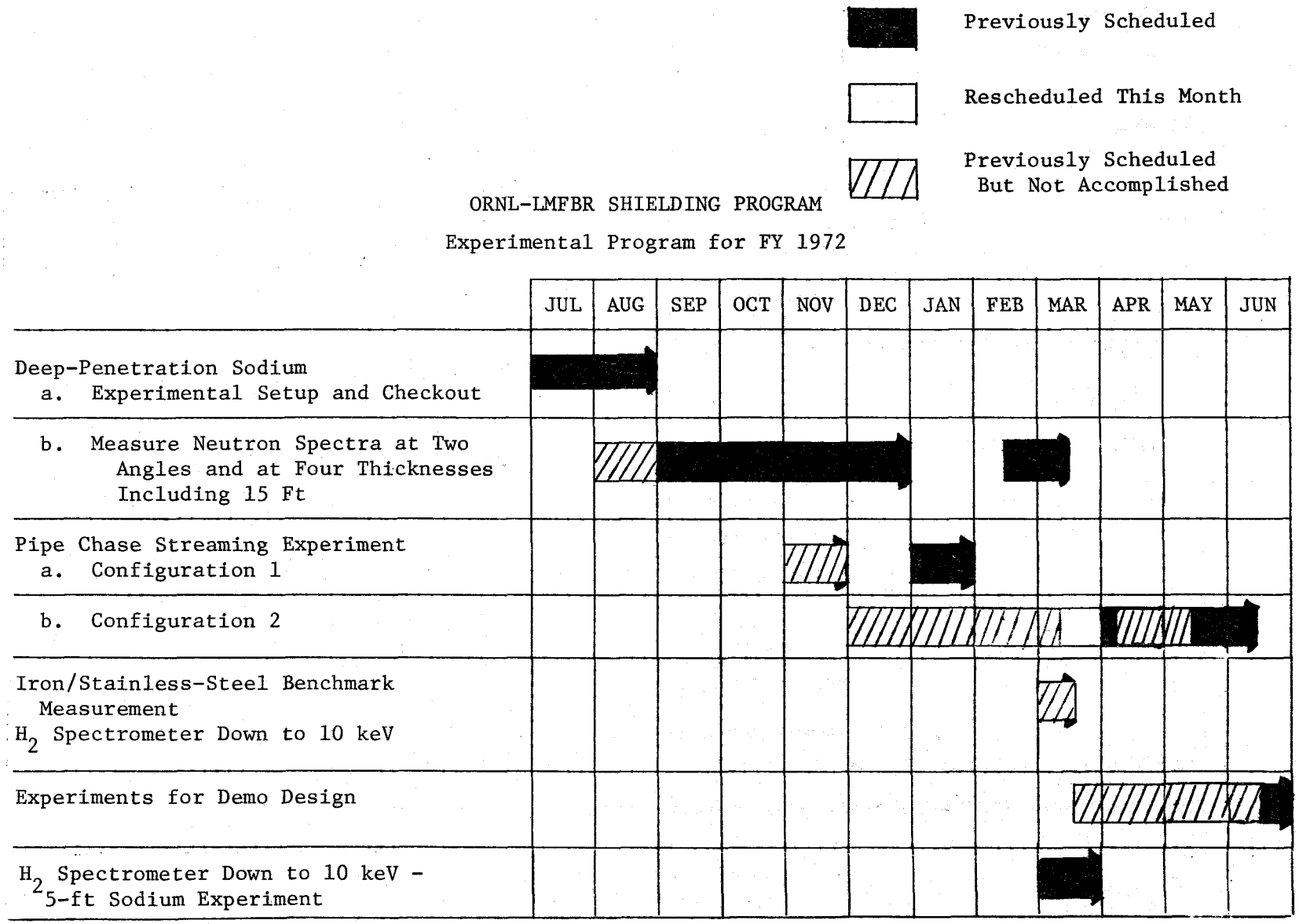
Abstract

This report summarizes the progress made by the ORNL Neutron Physics Division in Fast Reactor Shielding Research during the month of May 1972.

ORNL-LMFBR SHIELDING PROGRAM

Analytical Program For FY 1972




Previously Scheduled
Rescheduled This Month
Previously Scheduled But Not Accomplished
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Experimental Program for FY 1972
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AUG
SEP
OCT
NOV
DEC
JAN
FEB
MAR
APR
MAY
JUN
Deep-Penetration Sodium
a. Experimental Setup and Checkout
b. Measure Neutron Spectra at Two Angles and at Four Thicknesses Including 15 Ft
Pipe Chase Streaming Experiment
a. Configuration 1
b. Configuration 2
Iron/Stainless-Steel Benchmark Measurement
H_2 Spectrometer Down to 10 keV
Experiments for Demo Design
H_2 Spectrometer Down to 10 keV - 5-ft Sodium Experiment

A. Current FFTF Studies

1. Stored Fuel Power

The DOT-DOMINO-MORSE coupled three-dimensional calculation was completed for the improved determination of the stored fuel power. The DOT R-Z calculation in this sequence was for a modification of the configuration specified in the letter FRP-603, November 17, 1971, from Fox to Mynatt. The modifications included the removal of the stainless steel representing the instrument trees and the in-vessel fuel handling mechanisms, and the addition of an 8-in.-thick B_4C deck in the cavity. An internal boundary angular flux tape was produced by the DOT calculation for a vertical surface at a radius of 198.12 cm which is just outside the flange at the top of the core barrel.

The MORSE calculation was performed in the adjoint mode for the reactor geometry extending from 198.12-cm radius to 553.72-cm radius and from the axial midplane to a height of 648 cm. The adjoint fixed "source" was the neutron-production macroscopic mixture cross section, $\nu\Sigma^f_{fuel}$, distributed uniformly in the fuel sections of the stored fuel elements. The adjoint particles were followed in the random walk until they crossed the cylindrical surface at r = 198.12 cm. At this point, the adjoint particle weight was multiplied by the appropriate value from the forward boundary angular flux from the DOT calculation. The sum of such scores under proper normalization gives the stored fuel power. The stored fuel elements were represented as cylinders 4.724 in. in diameter having the same composition as the homogenized core-1 (control rods withdrawn). The stored fuel cylinders were 69.5 in. high, reflected at the bottom, and had four axial zones -- core, insulator, reflector, and fission gas plenum. The 66 stored fuel

cylinders were located in three stored fuel arrays, each having a loading pattern as shown in Fig. 1. The pattern consists of 7 columns, each aligned on a radius vector rotated 4° from adjacent columns. In alternate columns, the elements are located on constant radii from the set (208, 216, 224, 232, 240, 248, 256, and 264 cm). The steel stored fuel grid and support structure were not included in the model.

The objective of the stored fuel calculation was the determination of the total stored fuel power, which is expressed as the ratio of stored fuel power to core power. The DOT-DOMINO-MORSE calculation gave a value of 2.87×10^{-4} with a statistical uncertainty of $\pm 0.2 \times 10^{-4}$.

The stored fuel power was also determined by an approximation using a combination of DOT R-Z calculations. The first DOT calculation is the full reactor R-Z calculation described above from which the scalar flux is averaged from a radius of 200 cm to 270 cm enclosing the stored fuel area. This gives an energy-, group-, and Z-dependent scalar flux which was used as an isotropic flux boundary condition for a DOT calculation of a cylindricized stored fuel array. The DOT stored fuel calculation (including subcritical multiplication) gave a stored fuel/core power ratio of 3.3×10^{-4} in the forward mode and a value of 3.8×10^{-4} when an adjoint calculation was performed and coupled to the same boundary source.

The improved evaluation of the stored fuel power indicates a stored fuel/core ratio of 2.9×10^{-4} with a 50% uncertainty. The uncertainty is an estimate based on previous integral experiments for neutron transport through stainless steel and sodium. The uncertainties associated with homogenization of the fuel pins in each assembly and the fuel cross

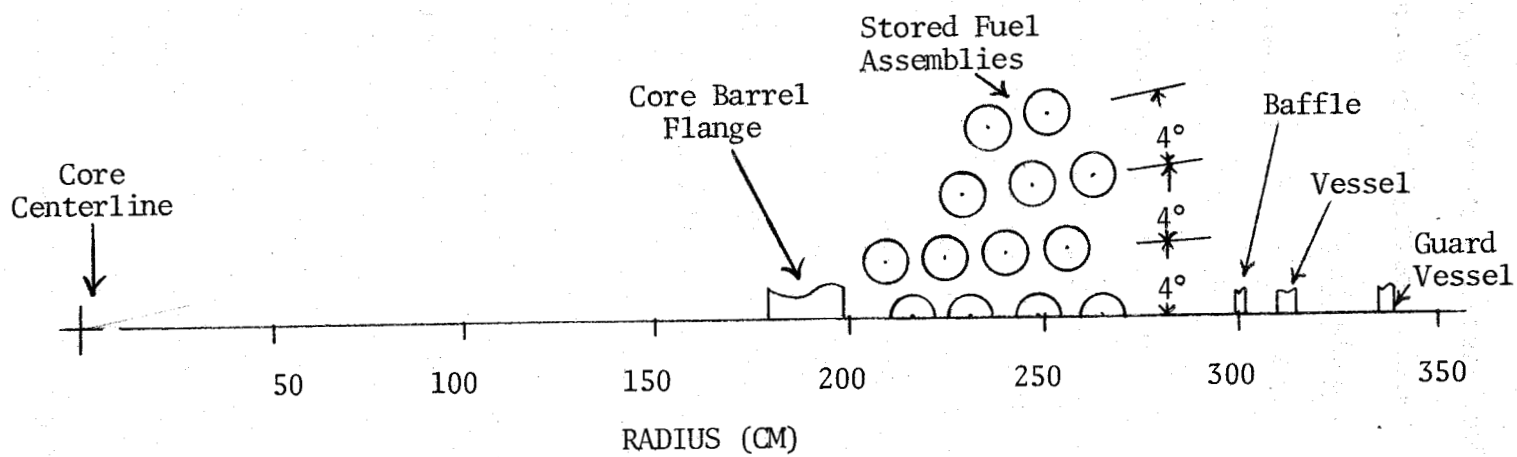

Fig. 1. Illustration of the Location of Stored Fuel Assemblies in MORSE Calculation Showing One-Half of One of the Three Stored Fuel Arrays.

sections are undetermined at the present time. All previous DOT R-Z calculations including stored fuel assumed a stored fuel/core power ratio of 1.0×10^{-4}, and the results should be multiplied by a factor of 2.9.

2. Head Compartment Shield Calculations

DOT calculations were performed to aid the evaluation of a shield concept, including a massive head compartment shield and the omission of the cavity shield and vessel support shielding. The head compartment shield consisted of 6 in. of borated (5 wt%) polyethylene followed by 9 in. of SA-508 steel. The objectives of the DOT calculations were to determine the neutron and secondary gamma-ray dose distribution for the head compartment shield and to determine the maximum activation rates for ^{59}Co and ^{55}Mn at the top of the cover.

The first step in the calculation consisted of restarting the R-Z calculation of the 11-71 model[1] to obtain a boundary angular flux at an elevation of 648 cm above the reactor midplane. This angular flux tape was converted from S-6 quadrature to the biased S-166 quadrature and from the 50-group structure to the 21-neutron group 18 gamma-ray group structure.

The modified tape was used as an external bottom boundary source for a 21-18 coupled DOT calculation of a horizontal section of the reactor extending from an elevation of 648 cm to the top of the cover. Figures 2-4 show the neutron total flux, neutron dose, and secondary gamma-ray dose contours from this calculation. These contours compare most nearly with the vessel support calculation without shielding, shown in Figs. 6 and 7 of the December monthly progress report, which show a maximum neutron dose rate of 9 rem/hr at a radius of 430 cm on the top of the cover. The new calculation, shown in Fig. 3, indicates a maximum dose rate of 1.5 rem/hr

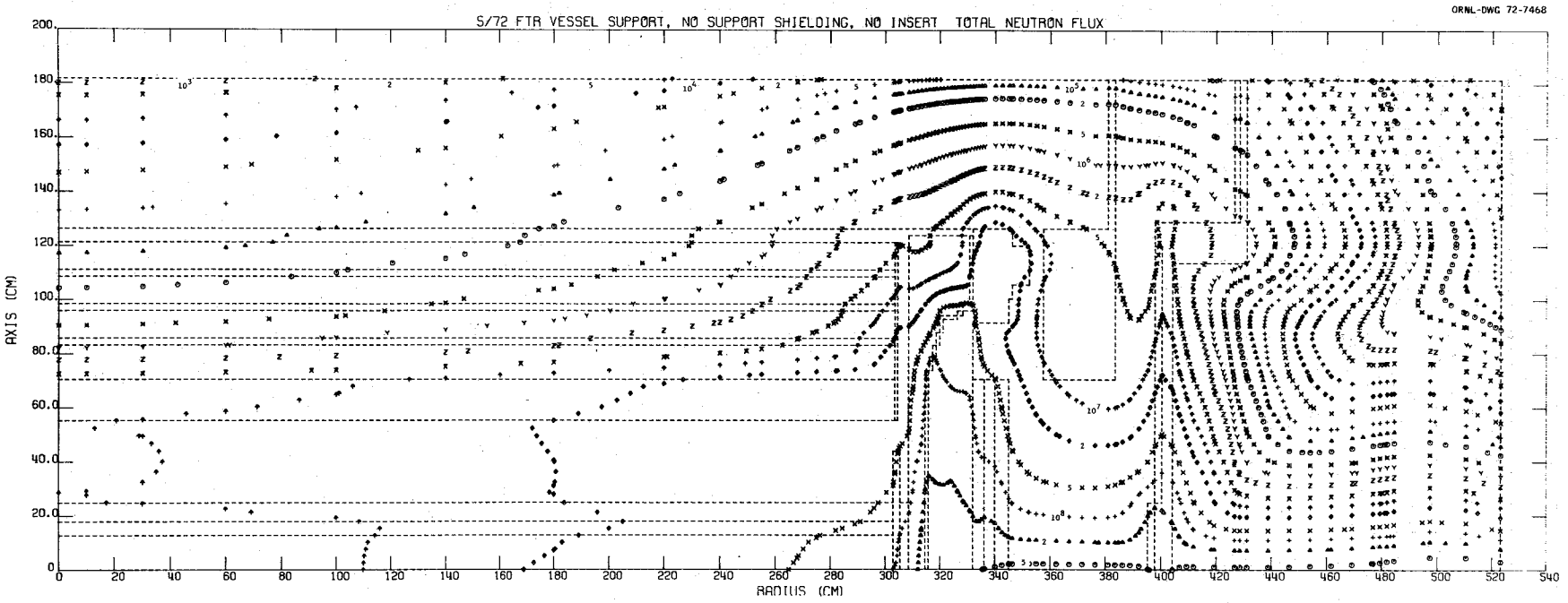

Fig. 2. Neutron Flux Contours From 5/72 DOT Vessel Support System Calculation.

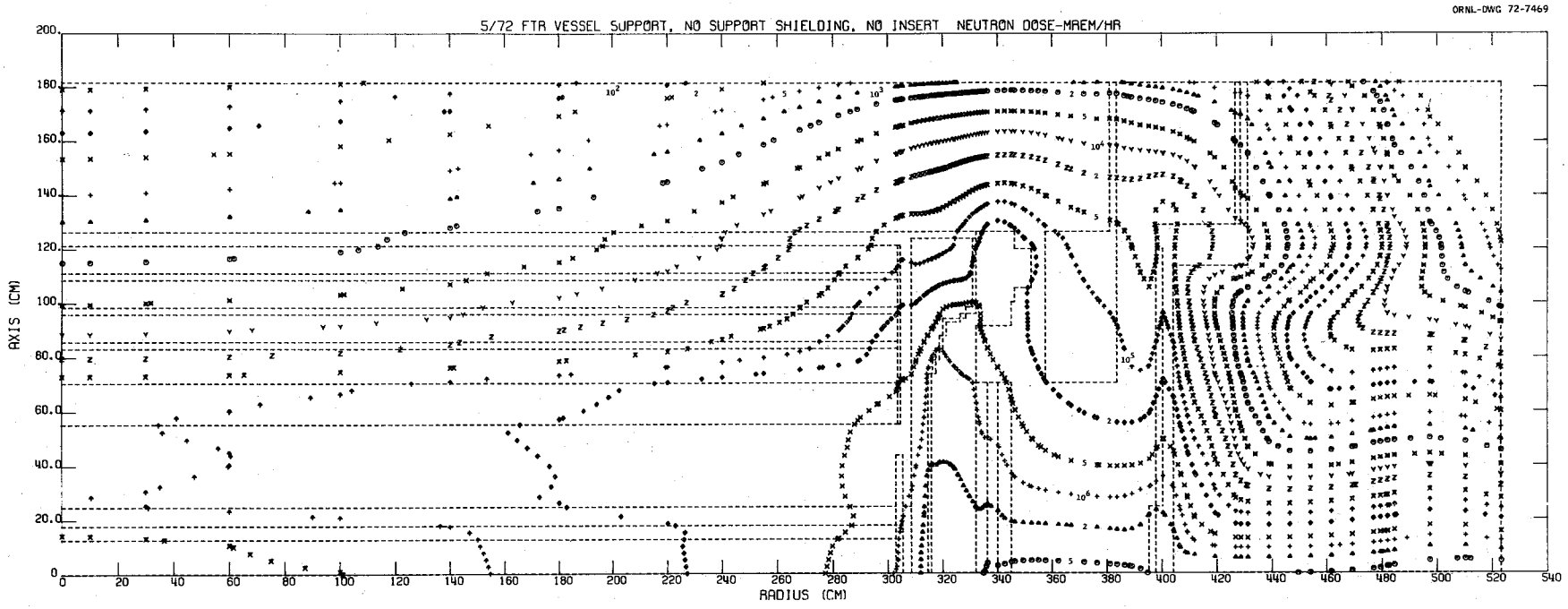

Fig. 3. Neutron Dose Contours From 5/72 DOT Vessel Support System Calculation.

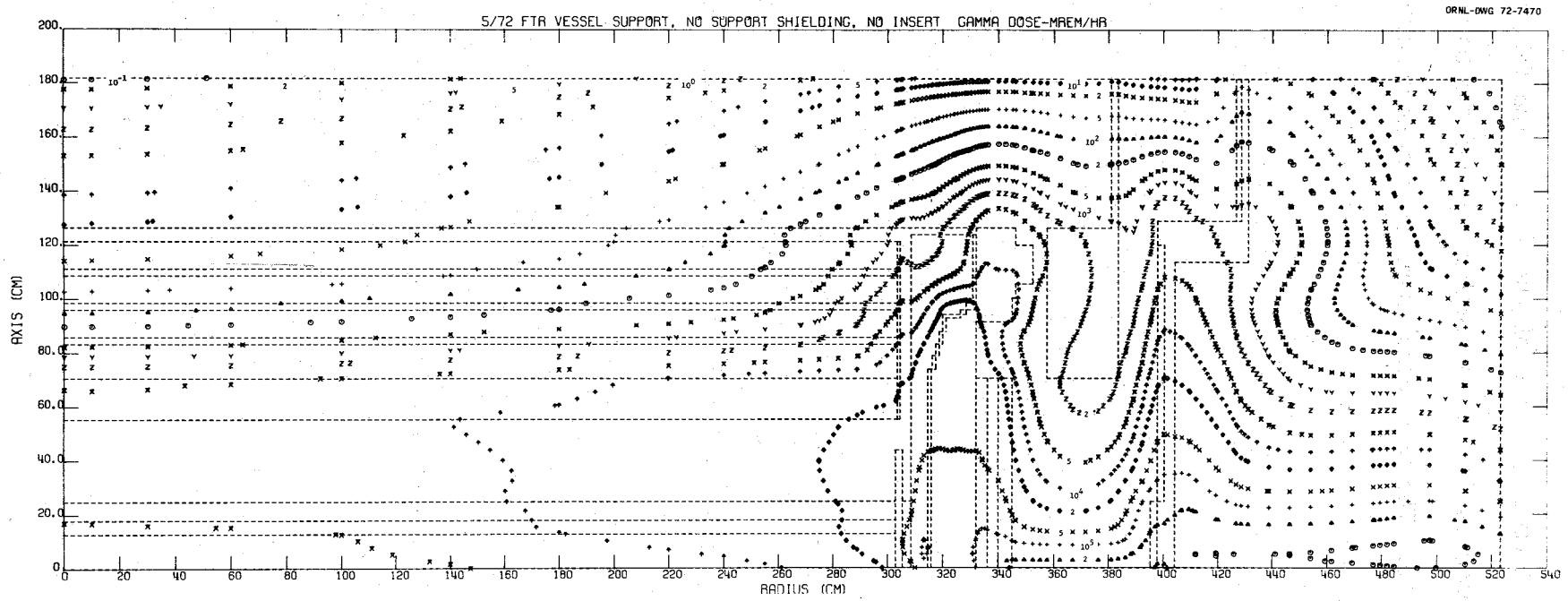

Fig. 4. Secondary Gamma-Ray Dose Contours From 5/72 DOT Vessel Support System Calculations.

at a radius of 350 cm on the top of the cover. The factor of 6 decrease in the neutron dose is due to the replacement of the empty box girder with magnetite concrete. The new vessel support system calculations contain the entire cover and thermal shield, and the neutron contours shown in Figs. 3 and 4 indicate that inflow of higher energy neutrons from the cavity influence the flux levels even to the center of the cover. The secondary gamma-ray contours, shown in Fig. 5, follow essentially the same pattern as the neutron contours. The maximum gamma dose rate on top of the cover is 9 mrem/hr, a phenomenal factor of 167 below the neutron dose.

The top boundary angular flux from the second step calculation was converted from S-166 quadrature to S-6 quadrature and used as an internal boundary source for the third step calculation. The third calculation was for a horizontal section of the reactor extending from 52.5 cm below the top of the cover to the top of the head compartment shield which is 91.44 cm above the cover. Figures 5-7 show the neutron total flux, neutron dose, and secondary gamma-ray contours from this calculation which used the 21 neutron group 18 gamma-ray group coupled cross-section library. The contours plotted in the zones below the boundary source are for the reflected component only and must be added to the values in corresponding positions on Figs. 2-4 to get the total flux or dose levels. The maximum neutron dose rate above the shield is 2×10^{-2} mrem/hr, and the maximum gamma-ray dose rate is 3×10^{-2} mrem/hr which is below the 0.1 mrem/hr objective for the nominal calculations. Activation rates for Co and Mn were calculated, and Table 1 shows the spatial dependence of the activation along the top of the cover.

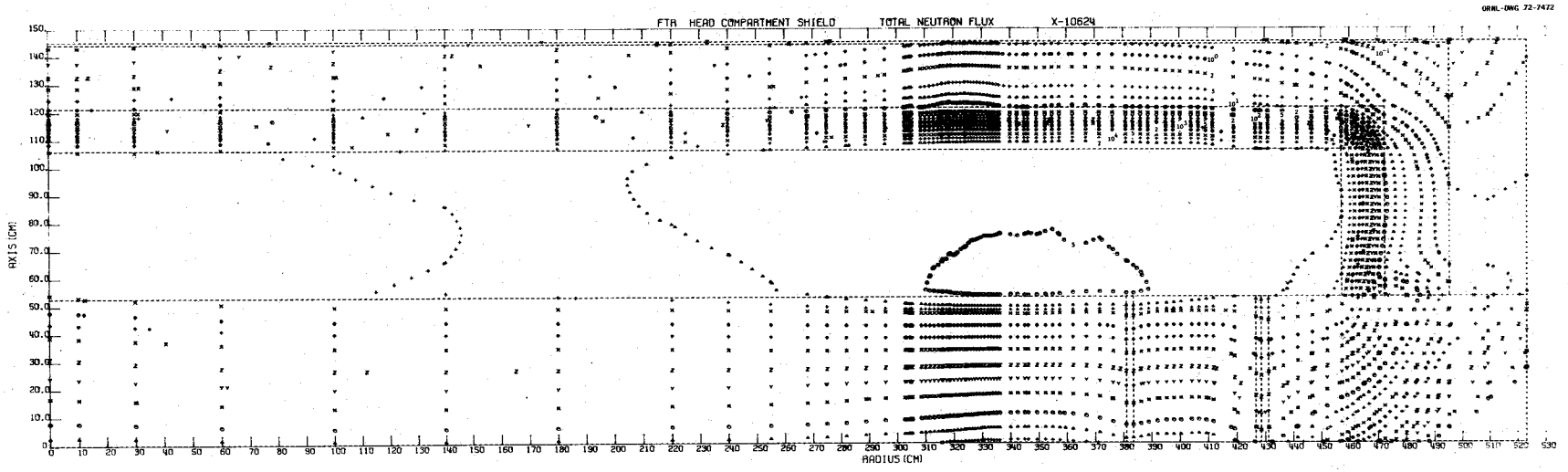

Fig. 5. Neutron Flux Contours From 5/72 DOT Calculation of a 6-in. Polyethylene 9-in. Steel Head Compartment Shield.

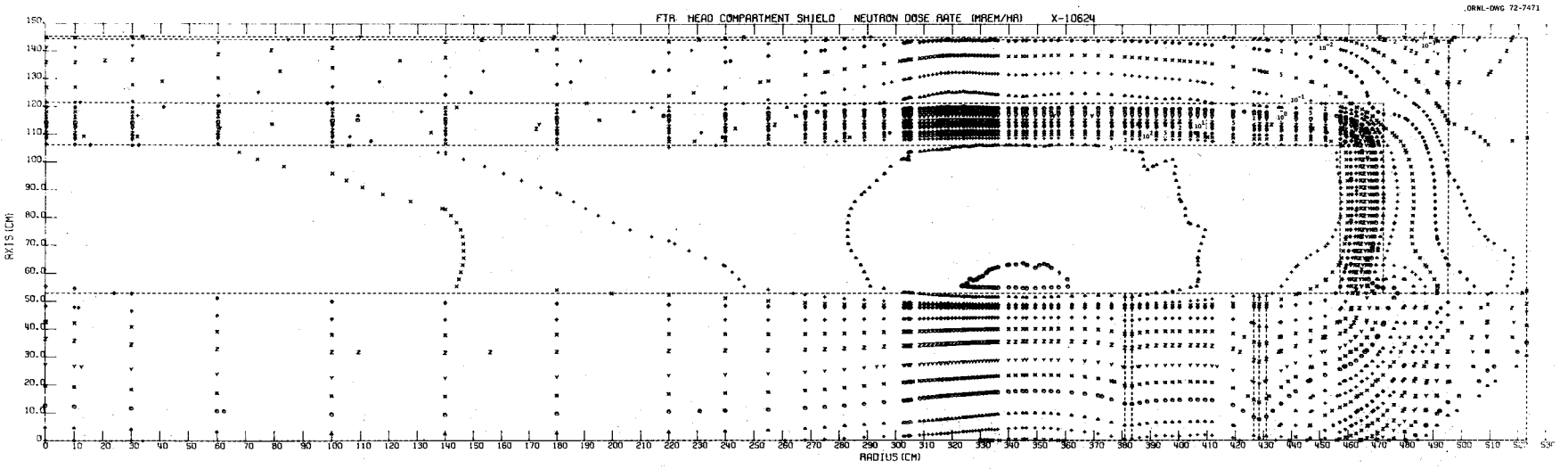

Fig. 6. Neutron Dose Contours From 5/72 DOT Calculation of a 6-in. Polyethylene 9-in. Steel Head Compartment Shield.

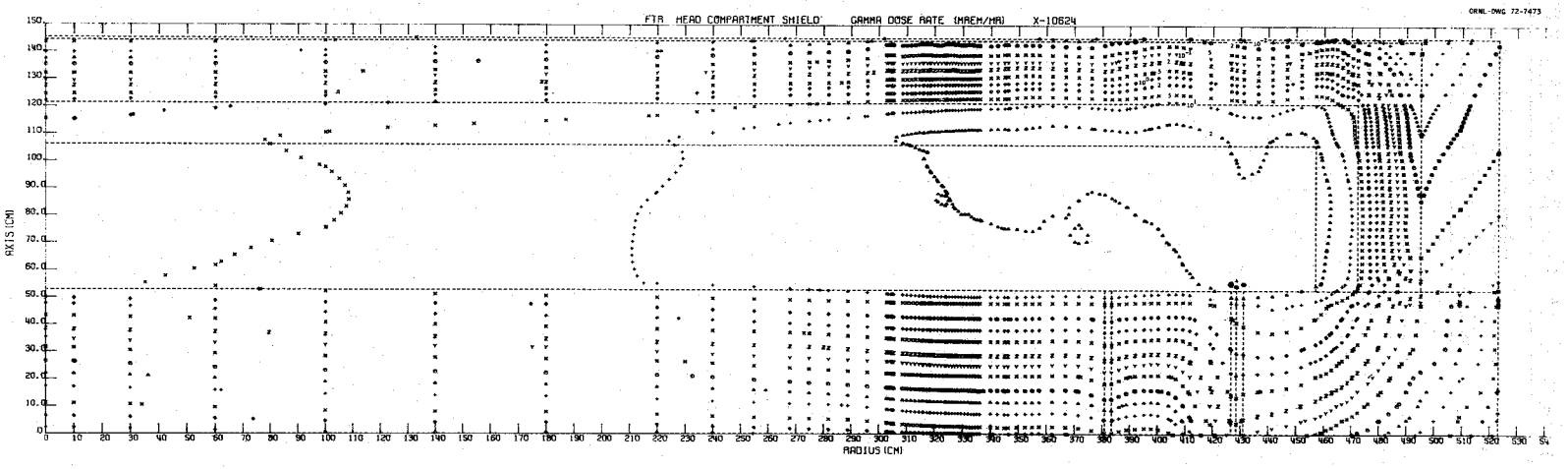

Fig. 7. Secondary Gamma-Ray Contours From 5/72 DOT Calculation of a 6-in. Polyethylene 9-in. Steel Head Compartment Shield.

Table 1. Activation Rates at the Top of the FTR Cover

Radius (cm)	Activation Rates, $\sigma\phi$, Activations/Sec/Nuclei x 10^{24}	
	Mn	Co
5	2.7×10^3	2.5×10^4
120	5.0×10^3	5.0×10^4
200	6.8×10^3	7.0×10^4
299	2.3×10^4	2.3×10^5
338	3.9×10^4	3.9×10^5
350	3.7×10^4	3.5×10^5
400	2.3×10^4	2.3×10^5

The results of the DOT calculations of this head compartment shield model indicate that the design objective neutron dose level could be achieved if the penetrations of the shield are designed such that streaming is not possible. The streaming problem could be quite serious. If, for example, it were not possible to add hydrogenous shielding in and above the IVHM rotating plugs, the neutron dose level directly above the IVHM penetrations would be approximately 10^3 mrem/hr, and, at an elevation 20 ft above the head the dose level would be approximately 15 mrem/hr. In these estimates, the dose levels in the present calculations are multiplied by a safety factor of 10.

B. Analysis of Pipe Chase Experiment

During the preanalysis of the TSF experiment simulating the FFTF HTS Pipeway, several DOT calculations and an AMC Monte Carlo calculation were performed to check the overall accuracy of DOT calculations of scattering in a cavity. The AMC calculation was for a cavity 11 ft by 11 ft by 4.8 ft with the source beam incident at the center of one of the large faces. Previously generated concrete albedos[2] were used in the AMC calculation. The DOT calculation was for a cylindrical 11-ft-diam. by 4.8-ft-long cavity surrounded by a 2-ft thickness of concrete. The beam entered the cavity on the centerline.

Count-rate distributions were determined from the calculation for the 5-in. and 12-in. Bonner balls for positions along the pipe-chase axis, which is located at an 8-ft radius from the centerline and midway between top and bottom walls in the three-dimensional calculation. Figures 8 and 9 show the comparisons of the calculated results, which include one AMC and four DOT calculations. For the S-12 (96 angles) calculation, 5-group set-1 cross sections were collapsed from the 50-group library using the TSR-II source for weighting, and the cross sections identified as 5-group set-3 and the 14-group set were collapsed using the 50-group leakage spectrum from the DOT calculation of the pipe penetration experiment.[3] The group structures are shown in Table 2.

Pronounced ray effects were evident in the S-12 calculations. A one-iteration restart with an S-380 quadrature effectively removed the ray effects. The 14-group S-380 calculation agrees well with the AMC results near the source entrance wall and is 10 to 25% higher near the pipe exit wall. The 5-group set-3 DOT results agree substantially with the 14-group

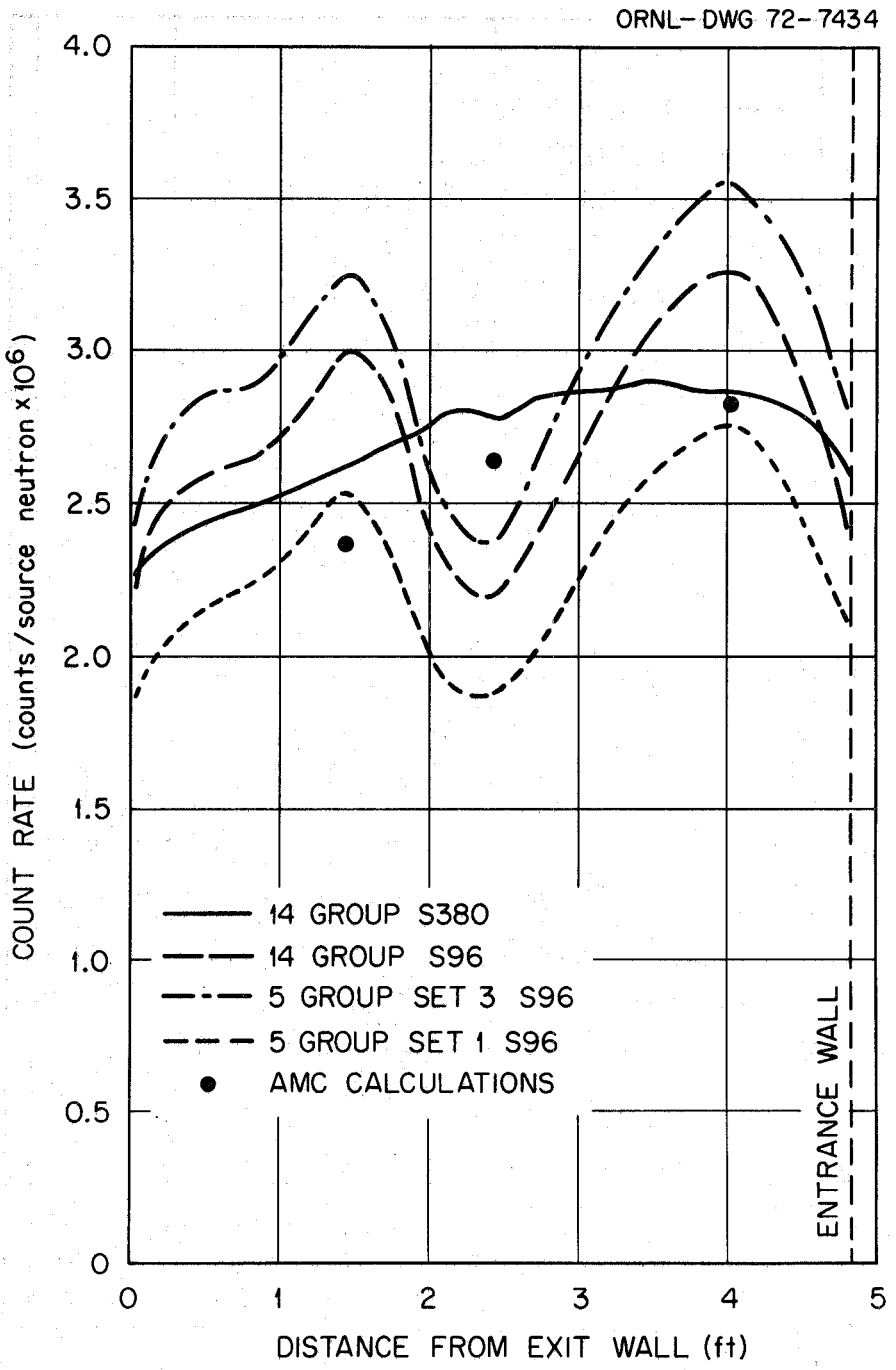

Fig. 8. Calculated 5-in. Bonner Ball Response Distribution Across Cavity at Pipe Chase Location.

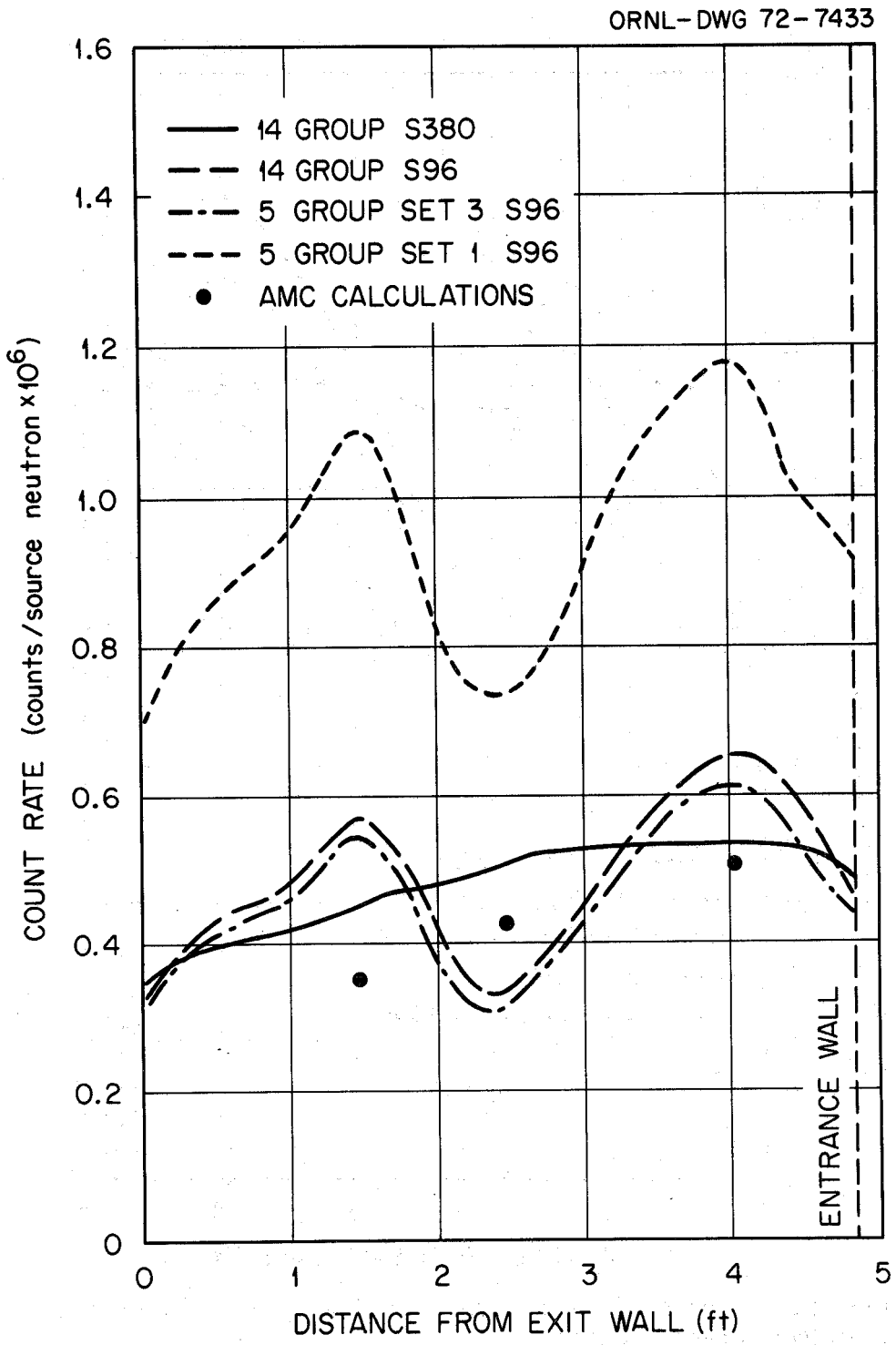

Fig. 9. Calculated 12-in. Bonner Ball Response Distribution Across Cavity at Pipe Chase Location.

Table 2. Group Structures Used in Analysis of Pipe Chase Experiment

Group	50 Group Set Upper Energy (eV)	14 Group Set Upper Energy (eV)	5 Group Set 1 Upper Energy (eV)	5 Group Set 3 Upper Energy (eV)
1	1.49 (+7)*	1.50 (+7)	1.5 (+7)	1.5 (+7)
2	1.22 (+7)	3.00 (+6)	1.8 (+5)	1.35 (+6)
3	1.00 (+7)	7.40 (+5)	4.3 (+3)	1.25 (+5)
4	8.19 (+6)	1.80 (+5)	22.6	2.03 (+3)
5	6.70 (+6)	5.20 (+4)	0.414	0.414
6	5.49 (+6)	1.50 (+4)		
7	4.49 (+6)	4.30 (+3)		
8	3.68 (+6)	1.23 (+3)		
9	3.01 (+6)	4.54 (+2)		
10	2.47 (+6)	1.01 (+2)		
11	2.02 (+6)	2.26 (+1)		
12	1.65 (+6)	5.04		
13	1.35 (+6)	2.38		
14	1.11 (+6)	0.414		
15	9.07 (+5)			
16	7.43 (+5)			
17	6.08 (+5)			
18	4.98 (+5)			
19	4.08 (+5)			
20	3.34 (+5)			
21	2.73 (+5)			
22	2.24 (+5)			
23	1.83 (+5)			
24	1.50 (+5)			
25	1.23 (+5)			
26	8.65 (+4)			
27	5.25 (+4)			
28	4.09 (+4)			
29	3.18 (+4)			
30	2.48 (+4)			

Table 2 (continued)

Group	50 Group Set Upper Energy (eV)	14 Group Set Upper Energy (eV)	5 Group Set 1 Upper Energy (eV)	5 Group Set 3 Upper Energy (eV)
31	1.93 (+4)			
32	1.50 (+4)			
33	7.10 (+3)			
34	4.31 (+3)			
35	3.35 (+3)			
36	2.61 (+3)			
37	2.03 (+3)			
38	1.58 (+3)			
39	1.23 (+3)			
40	9.61 (+2)			
41	4.54 (+2)			
42	2.14 (+2)			
43	1.01 (+2)			
44	47.9			
45	22.6			
46	10.7			
47	5.04			
48	2.38			
49	1.13			
50	0.414			

results while the 5-group set-1 calculation overpredicts the 12-in. Bonner ball count rate by a factor of two. The overall agreement between DOT and AMC is considered to be good in view of the differences in geometry and in the concrete cross sections.

C. TSF Experiments

The experiments performed during May for the pipe chase streaming experiment, configuration 2, will be reported in the June progress report.

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2. W. A. Coleman and R. E. Maerker, "Differential Neutron Current Albedos For Concrete in the Incident Energy Range 0.5 eV to 200 keV," ORNL-3967, Vols. I-VI (1966).

3. F. R. Mynatt et al., "Fast Reactor Shielding Monthly Progress Report for March, 1972," ORNL-TM-3800 (1972), p. 25.

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