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FFTF RELATED STUDIES IN THE ORNL-LMFBR SHIELDING PROGRAM[†]

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Abstract

This paper documents an oral presentation given in the special session on Fast Reactor Shielding Sponsored by the Shielding and Dosimetry Division at the October 1971 meeting of the American Nuclear Society in Miami, Florida. The progress to date on neutron transport calculational studies of FFTF shielding design is presented, including the analysis of supporting integral experiments. The major FFTF shielding problems considered were the predictions of (1) neutron dose rates above the reactor head and (2) neutron fluence intensities at the core support grid plate.

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I. INTRODUCTION

The basic goal of the ORNL-LMFBR shielding program is to develop and verify in a timely fashion an analysis technology that will be capable of providing results suitable for accurate and efficient LMFBR design. This technology includes development of computer codes for radiation transport, optimum shield design, and cross-section processing, design and execution of integral experiments for evaluation of analytical methods and data, and development of a methodology for applying this capability to LMFBR design. Since the spring of 1970, the ORNL shielding program has been providing direct shield analysis support for the FFTF design effort. This work has been invaluable in providing the incentive for "now" capability, giving clear focus for short-range advancement in the state of the art and providing real-time design experience for those who are responsible for long-range planning and development.

The work reported here is preliminary, in the sense that more analysis is planned in each of the problem areas. This presentation gives a fairly comprehensive overview of the progress in the project in late 1971. The integral experiments and the analyses of design problems will be reported more completely by the principal investigators for each task.

The ORNL support effort for FTR shielding has been concentrated thus far on two major problems. These are the shield design and evaluation necessary to meet a criterion of 1-mrem/hr neutron dose rate in the reactor head compartment, and a criterion for a total flux limit of 4×10^{13} neutrons/cm²/sec on the core support grid plate. A third major problem area, that of limiting the neutron flux levels in the heat exchanger vaults, will be studied at ORNL early in 1972.

II. THE HEAD COMPARTMENT PROBLEM

The problem of calculating the neutron dose rates above the reactor cover is by far the more difficult of the two problems which have been considered in this design. An idea of its complexity can be obtained from Fig. 1, which shows an artist's drawing of the FFTF reactor. There are multiple paths through which radiation reaches the lower surface of the reactor head and multiple paths for streaming and transmission through the head and cover. The design objective is to provide for the total dose level of 2 mrem/hr directly above the reactor vessel head. Approximately 1 mrem/hr is allocated for gamma-ray dose from the sodium coolant in the branch arm piping; therefore, the desired neutron dose level is only 1 mrem/hr.

Although the most important part of the problem is the determination of the source distribution incident on the head, the construction schedule required a design evaluation for the clearance gaps around the major penetrations of the head at an early date. These evaluations were performed using a crude estimate of the source from one-dimensional transport calculations and a detailed two-dimensional transport calculation for the head penetration.

Figure 2 shows the DOT R-Z geometry for a model of a main rotating plug. A doubly offset clearance gap is shown having widths of 1/2 in., 3/8 in., and 1/4 in., the larger gaps being required in locations having more severe thermal gradients. In order to accurately calculate the streaming in the void gaps, a highly biased 315-angle quadrature was used in the 50-group, P_1 DOT calculations. Several DOT calculations were performed for various slit configurations using the one-dimensional source

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FFTF REACTOR

Fig. 1. Artist's Drawing of a Preliminary FFTF Reactor Design.

Fig. 2. DOT R-Z Geometry for Calculation of Streaming in Clearance Gaps Surrounding the IVHM Rotating Plugs.

estimate. The major conclusion from this preliminary effort was that the back-of-the-envelope estimates were extremely conservative and more realistic clearances, such as those shown in Fig. 2, could be used.

Figure 3 shows a computer drawing of the DOT R-Z geometry for the FTR reactor from the core midplane to the cover, and also shown are total flux contours for this problem which contained a B_4C shield deck in the radial cavity. Although the intensity of the total flux does not change dramatically over the lower surface of the head, the spectrum changes radically due to the influence of the radial cavity. Several DOT calculations were performed both with and without the shield deck during the study leading to an improved source description for the rotating plug calculations.

It was originally assumed that streaming in the heterogeneous fission gas plenum and steel structure in the sodium pool might be significant. Since a rigorous calculation of such streaming is beyond the present capability, the approach taken was to model these zones with an annular ring configuration which exaggerates the streaming and gives an upper bound to the effect. If such streaming is found to be important, the model is then improved by reducing the widths of the rings to determine the lowest upper bound that can reasonably be obtained. If the streaming effect obtained with the first model is insignificant, one can, for the situation considered, justifiably use a homogenous model for that zone.

Table 1 shows results for several calculations involving the fission gas plenum and pool steel. The annular ring calculations were performed with a complex procedure involving overlapped or bootstrapped DOT calculations in which an S-315 biased angular quadrature was used to provide

Fig. 3. Computer Drawing of DOT R-Z Geometry and Resulting Total Flux
Contours for FTR Core-to-Cover Model 9-4-70.

Table 1

Effect of Fission Gas Plenum and Pool Steel
Models on Total Flux Levels at Top of
Pool and Top of Head

<u>Models</u>		<u>Total Flux (neut/cm²/sec)</u>	
<u>Fission Gas Plenum</u>	<u>Pool Steel</u>	<u>Above Pool</u>	<u>Above Head (Center)</u>
homogenized	none	2.8×10^8	3.7×10^0
void	none	6.8×10^9	5.4×10^3
1-in. annular rings	none	5.9×10^8	3.4×10^1
2-in. annular rings	none	8.8×10^8	6.5×10^1
homogenized	homogenized	3.0×10^7	1.3×10^0
1-in. annular rings	homogenized	6.8×10^7	6.2×10^0
1-in. annular rings	annular rings	8.9×10^7	6.2×10^0
0.325-in. annular rings	homogenized	-	-

accurate calculations of the streaming. For the fission gas plenum it is seen that an assumption of a pure void leads to a great enhancement of the flux above the head. However, for a 1-in. annular ring model, which is extremely conservative, the flux above the head is increased only by a factor of 5 when pool steel is included. For the heterogeneous pool steel, the calculations show that the conservative annular ring model gives no increase in the flux above the head. Therefore, the pool steel may be homogenized and the effect of the fission gas plenum is at most a factor of 5. Hand calculations estimate that the effect of the fission gas plenum is less than a factor of 2,¹ and a lower upper bound using a 0.325-in. annular ring model will be obtained during the next round of calculations.

Figure 4 shows a comparison of three spectra at the top of the pool obtained from one- and two-dimensional calculations. The spectra at the center of the pool from the two-dimensional calculation is harder than the spectra from the one-dimensional result primarily due to the omission of the pool steel in the one-dimensional calculation. The two-dimensional spectrum at the outer edge of the pool is much harder due to the inflow of fast neutrons from the cavity surrounding the reactor vessel. Many calculations have been made to evaluate the transmission and streaming through the rotating plug configuration for the various source spectra which have been obtained.

The neutron transmission through the bulk steel is very sensitive to the input spectrum and the transmission is greater for harder spectra. The streaming component, on the other hand, is less sensitive to spectral changes; therefore, the relative importance of streaming is spectral

Fig. 4. Comparison of Neutron Spectra at the Top of the Sodium Pool
From One- and Two-Dimensional Calculations.

dependent and is expected to vary at different positions on the FTR head.

Figure 5 shows the total flux distribution across the top of the rotating plug for three calculations. For the triple slit plug and the neutron spectra (I1) near the center of the pool the streaming is fairly significant but the maximum flux level is only about $12 \text{ neutrons/cm}^2/\text{sec}$. For the same plug geometry and the neutron spectrum near the edge of the pool (I74) streaming appears much less significant and the maximum flux level is approximately 1.5×10^2 . Since streaming appeared less significant in the dominant case, a plug configuration with only one offset was calculated, and this shows a slightly increased streaming effect with a maximum flux level of 2×10^2 . These flux levels correspond to dose rates on the order of 1 to 2 mrem/hr slightly exceeding the design objective. More importantly, these are nominal values and, while the final evaluation of the uncertainty has not yet been performed, present estimates indicate that the nominal calculated dose value at this stage in the design should be a factor of 10 lower than the design objective. Elimination of the streaming component would provide only a factor of 2 reduction, and it therefore appears that a single offset gap could be used in the plug design. A reduction in the dose level above the plug would best be achieved by reducing the inflow of fast neutrons from the cavity.

An experiment was performed at the TSF to evaluate the accuracy of the DOT calculations for the rotating plug design. Figure 6 shows a drawing of the annular slit experiment which was constructed. Since the relative effects of streaming and transmission through slits in iron depend strongly on the incident spectrum, a spectrum modifier was included in

Fig. 5. Comparison of Calculated Total Flux Distributions Above the Rotating Plug.

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ALL DIMENSIONS IN INCHES

Fig. 6. Schematic Drawing of the Annular Slit Streaming Experiment.

the experiment. This modifier was designed to provide the most appropriate spectrum that could be obtained within an allowable overall attenuation of 10^{-6} for the modifier and the annular slit assembly. This attenuation criteria was based on previous experience and is related to the source strength and detector efficiency as well as the minimum neutron background that can be achieved.

The calculations of the experiment were to be made on an absolute basis from the known "free field" beam source. Errors incurred in calculation of the spectrum modifier would therefore impact on the conclusions drawn from comparison of the calculations with measurements at the slit exit. Figure 7 shows an absolute comparison of experiment and calculation for the 3-in. Bonner ball² traverse at the exit of the spectrum modifier alone. The agreement in intensity and shape is excellent. Comparisons of proton-recoil spectra and spectral indication with a complete Bonner ball set were also made, and indications are that the maximum errors are approximately 15-20%.

Figure 8 shows a more detailed drawing of the annular slit assembly. Several sleeves were used to vary the slit width and offset. Nickel plates were inserted in the first section of the assembly to increase the ratio of streaming to direct transmission. In the DOT calculation, the streaming in the slit provides the most stringent requirements on the quadrature. For a single annular slit in an absorbing media, the attenuation can be computed analytically and used to test the DOT calculation. Table 2 shows results for three slits and several quadratures. Slit 1 is representative of the first slit in the experimental assembly, slit 2 the second leg, and slit 3 the penetration through the lucite section.

Fig. 7. Comparison of Calculated and Measured 3-in. Bonner Ball Count Rates Along Exit Surface of the 8-in. Fe, 60-in. Na Spectrum Modifier.

DIMENSIONS IN INCHES

Fig. 8. Drawing of Annular Slit Assembly.

Table 2
 Analytic Vs. DOT Calculated Attenuation of Black Wall
 Annular Slits for Various Angular Quadratures

Slit	Quadrature	Analytic ϕ/ϕ_o	DOT ϕ/ϕ_o	DOT/ Analytic
1	S-100	3.179(-4)*	4.593(-5)	0.14
	S-315	3.179(-4)	2.670(-4)	0.84
	S-298	3.179(-4)	2.98(-4)	0.94
2	S-315	3.55(-2)	3.201(-2)	0.90
	S-298	3.55(-2)	3.298(-2)	0.93
3	S-315	1.467(-3)	1.239(-3)	0.84
	S-298	1.467(-3)	1.450(-3)	0.99

*Read as 3.179×10^{-4} .

It is shown that the quadrature identified as S-298 gives agreement within 7% for each of these slits.

Figure 9 shows the comparison of measurement and calculation for a 6-in. Bonner ball traverse for the 3/4-in. annular slit with no offset. The calculation assumes a point detector and thus gives a more sharply peaked distribution. However, the integrals under the measured and calculated shapes should be identical and are used to evaluate the comparison. Figure 10 shows a similar comparison for a 3/4-in. slit with 1-1/2 in. offset. Table 3 shows the measured/calculated ratios for the traverse integrals for several cases. For the 3-in. Bonner ball which measures the low-energy spectrum, the agreements are exceptionally good. For the 6-in. Bonner ball which measures the total flux the ratio gets larger as the streaming effect is reduced. The maximum value of 1.8 for the solid assembly is comparable to that obtained in previous iron comparisons,³ and the disagreement is assumed to be due to incorrect cross-section representation in the energy range from 50 to 500 keV. The major conclusion from these comparisons is that with the special quadratures, the uncertainty in the transport calculation is due primarily to the errors in the bulk transmission, and, with the cross sections presently being used, the maximum observed value for this error is the factor of 1.8 for a thickness of 44 in. of steel.

A more serious problem associated with the head compartment is that of streaming in the vessel support region which is on the periphery of the reactor cover. The severe mechanical requirements in this area for support of the reactor and complications due to thermal gradients lead to complex geometry having many streaming paths. Unfortunately, this area is located

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Fig. 9. Comparison of Calculated and Measured Results for a 6-in. Bonner Ball Traverse Behind the 3/4-in. Annular Slit Without Offset.

Fig. 10. Comparison of Calculated and Measured Results for a 6-in. Bonner Ball Traverse Behind the 3/4-in. Annular Slit With 1-1/2 in. Offset.

Table 3
 Summary of Measured/Calculated Ratios of Traverse
 Integral For 3- and 6-in. Bonner Ball Detectors
 For Annular Slit Configurations

Configuration	3-in. Bonner Ball Integral Ratio	6-in. Bonner Ball Integral Ratio
3/4-in. straight slit with no collimator	0.95	1.36
3/4-in. straight slit with collimator*	0.94	1.26
3/4-in. slit with 3/4-in. offset with collimator	1.00	1.32
3/4-in. slit with 1-1/2-in. offset with collimator	1.12	1.52
solid annular assembly with no collimator	-	1.81
solid annular assembly with collimator	1.05	1.27

* Collimator is 4-in. lucite-polyethylene with
 1-1/8-in.-wide annular slit.

directly above the reactor cavity which has a harder neutron spectrum than that which exists over the center of the pool. Recently it has been shown that fissions in the stored fuel assemblies aggravate this problem.

Figure 11 shows a computer-drawn DOT R-Z geometry for an early model of the vessel support system which in this case includes a B_4C shield ring to reduce streaming. The flux contours show that the combination of streaming paths and high-energy spectra in the cavity leads to high flux levels above the support area. Table 4 shows the maximum flux levels calculated above the vessel support system for several shield configurations. These results do not include stored fuel effects. With no shielding the maximum total flux is 2.1×10^4 , which is approximately 200 mrem/hr. The minimum flux value of 5.08 is obtained with a B_4C cavity shield deck, as shown in Fig. 3, and a shield ring, as shown in Fig. 11. The calculations show that the shielding in the vessel support area is worth about a factor of 25, that additional shielding in the support area alone was not particularly effective, and that the shield deck in the reactor cavity is worth about a factor of 160. The support area and cavity shields appear to act independently so that with both shields the improvement compared with the unshielded case is a factor of $25 \times 160 = 4000$.

Figure 12 shows the effect of stored fuel on the reactor cavity spectrum as obtained from the DOT R-0 calculation at the core midplane. Preliminary adjoint calculations of the vessel support system geometry shown in Fig. 11 indicate that the spectral shift caused by the stored fuel produces a factor of 100 increase in the flux and dose levels above the head.⁴ Therefore, as an estimate, the fluxes in Table 4 can be multiplied by 100 to obtain comparable values with stored fuel effects.

Fig. 11. Computer Drawing of DOT R-Z Geometry and Resulting Total Flux
Contours for Vessel Support System Model 2-23-71.

Table 4

Summary of Vessel Support System Streaming Calculations

Configuration	Maximum Total Flux Above Vessel Support (neutrons/cm ² /sec)
No shielding in reactor cavity or in vessel support area	2.10×10^4
Shielding in vessel support area only	8.59×10^2
Two shields in vessel support area	4.60×10^2
Shield deck in reactor cavity; no shield in vessel support area	1.34×10^2
Shield deck in reactor cavity; one shield ring in vessel support area	5.08×10^0

Fig. 12. Comparison of Calculated Neutron Spectra in FTR Radial Cavity

With and Without Stored Fuel.

In order to estimate the overall uncertainty in the dose predictions above the head, experimental comparisons are needed in addition to the annular slit experiment previously discussed. The accuracy of transport calculations through solid iron has been studied extensively,³ and a definitive benchmark experiment is in progress at the TSF for neutron penetration through bulk sodium. Figure 13 shows a photograph of one of the sodium-filled tanks used in the experiment. This tank is 2-1/2 ft thick by 11 ft in diameter and is surrounded by concrete. Since verification of deep penetration in sodium is required, the experiment was designed to check transmission through up to 15 ft of sodium. There are two 5-ft-thick tanks and two 2-1/2 ft thick tanks.

The large diameter is required to minimize boundary effects in the deep-penetration measurements. The surrounding concrete serves to reduce background from neutrons which would leak from the side of the tanks and provides a controlled reflection of neutrons into the tank. In order to further control the environment around the sodium, a new concrete shield interface for the TSR-II reactor was fabricated and is shown in Fig. 14. As can be seen in the photograph, the beam emerges at the center of the shield and the flat concrete provides a well-defined interface with the sodium tank.

Figure 15 shows a photograph of a sodium experiment in progress. All four tanks are in position for a full 15-ft depth of sodium. The measurement being made utilizes a 3-in. Bonner ball surrounded by a 2.9-in.-thick shell of ^{10}B and provides an integral measurement of the flux between 10 and 100 keV. Comparisons with this experiment which will soon be completed will aid greatly in reducing the uncertainty in the flux incident on the FTR cover.

Fig. 13. Photograph 2.5-ft Thick by 11.0-ft Diameter Sodium Tank for ORNL Sodium Benchmark Experiment.

PHOTO 1760-71

Fig. 14. Modified TSR-II Reactor Shield and Beam Collimator.

Fig. 15. Experiment in Progress.

III. LOWER AXIAL SHIELD ANALYSIS

The second major effort which has been underway at ORNL is the analysis of the FTR grid-plate shield. Due to the compact design, the FTR core support grid plate is close to the core such that shielding must be provided to prevent excessive radiation damage. The current criteria is for 10% residual ductility at the end of ten years of operation at 75% availability assuming a grid-plate temperature of 600°F. The energy-independent fluence limit which has been used, corresponding with this criteria, is 9.5×10^{21} nvt for 7-1/2 full power years.

The shield has stringent requirements on configuration and materials. The present design utilizes 316-stainless-steel hexagonal-shaped blocks in each subassembly. Coolant flows through six cylindrical holes in each shield block. The shield zone in each subassembly contains at least one offset, where the subsequent section has the hole pattern rotated 90°, with a 1/2-in.-thick plenum space separating the sections with offset holes. Streaming may occur in these coolant holes or in the small space between subassemblies. For calculational purposes, the shield may be completely homogenized (74.75 volume % steel) or the central subassembly may be homogenized with a minimum envelope around the holes, an annular envelope containing the holes in the surrounding second row subassemblies, a zone in between these envelopes, and a completely homogenized zone outside the second row of subassemblies. This gives four homogenized zones with varying steel content.

Figure 16 shows a computer drawing of a DOT R-Z geometry for an FTR lower axial calculation with the relative locations of the shield and grid-plate identified. For this calculation, in which the shield zone was

Fig. 16. Computer Drawing of DOT R-Z Geometry and Resulting Total Flux Contours for FTR Below-Core Model 9-4-70.

completely homogenized, no strong irregularities are observed in the flux contours. Table 5 shows maximum flux levels at the grid plate, resulting from several S-6, P_1 , 50-group DOT calculations of lower axial FTR models. It is apparent that the peak flux is affected by the iron cross-section set denoted by ENDF material number, the type of stainless steel, and the method of homogenization. From the 304-stainless-steel results, it appears that the four-zone model gives a factor of 1.08 increase in the peak flux while the change to 316 stainless steel gives a flux reduction by a factor of 0.76 compared with 304 stainless steel. This effect is presumed to be due to the nickel content.

An experiment has been performed at the ORNL Tower Shielding Facility to measure the effect of streaming in a nearly prototypical model of the lower axial shield.⁵ Although not its primary function, the experiment has also served to test the accuracy of the homogenized shield calculation. Figure 17 shows a drawing of the experimental assembly, including a spectrum modifier which was used to provide the appropriate input spectrum to the shield zone. Most of the measurements were made with the 5-in. Bonner ball which is essentially a total flux detector.²

Figure 18 shows a photograph of the shield insert during fabrication.⁶ The assembly represents a cluster of seven FTR subassemblies, each containing the six coolant passages. Sodium was used to fill the plate assembly within the removable insert, and an aluminum mesh was used to represent sodium in the collar. Three removable inserts were fabricated: in one insert the coolant holes extended straight through; in the second insert, a single offset was used; and in the third insert the maximum number of offsets allowed by the plate construction was used. The third

Table 5

Results of Several Grid-Plate Shield Calculations

Grid-Plate Shield Model	Peak Flux at Grid Plate (neut/cm ² /sec)
73.6% SS-304 (1101 Fe)	4.11×10^{13}
4 zone (74.2%) SS-304 (1101 Fe)	4.24×10^{13}
4 zone (74.2%) SS-304 (1124 Fe)	4.78×10^{13}
74.75% SS-304 (1124 Fe)	4.41×10^{13}
74.75% SS-316 (1124 Fe)	3.33×10^{13}

Fig. 17. ORNL TSF Prototype Grid-Plate Shield Experiment Configuration.

Fig. 18. Photograph of Prototype Grid-Plate Shield Insert During Fabrication.

insert was intended to represent an homogenized assembly; however, some heterogeneity exists and the four-zone homogenization should be used for calculations for this insert.

The design of the spectrum modifier was an important part of the experiment preanalysis. Numerous ANISN calculations were used in a trial and error fashion to determine the configuration of available materials which would best represent the desired spectrum within a maximum allowable attenuation limited by background problems and detector sensitivity. The modifier that was chosen consisted of 10 in. type-304 stainless steel, 1 in. polyethylene, and 1/4 in. boral, as shown in Fig. 19. In order to verify the calculations of the spectrum modifier, fast and intermediate spectra were measured with the NE-213 and Benjamin spectrometers located in the positions indicated. Bonner ball measurements were made at a distance 43-1/2 in. below the modifier and a constant source detector separation was maintained in the subsequent shield measurements.

A DOT calculation was performed for the assembly shown in Fig. 19, including the reactor, the modifier support structure, and background shielding. This extremely complicated calculation was required because scattering in the support structure contributed approximately one half of the counts in the Bonner ball.

Figure 20 shows a comparison of the desired FTR spectrum incident on the lower axial shield shown as a histogram compared with the spectrum from the DOT calculation of the spectral modifier. While the agreement is certainly not perfect, it represents a major step forward in the use of spectrum modifiers. Figure 21 shows the comparison of a DOT calculation and an NE-213 measurement of the fast spectrum from the modifier, and

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Fig. 19. Experimental Configuration Used in Measurements for the Spectrum Modifier.

Fig. 20. Comparison of Objective Spectrum and That Calculated by DOT
for the Spectrum Modifier.

Fig. 21. Comparison of Measured and Calculated Neutron Spectra
($E > 0.8$ MeV) for the Spectrum Modifier.

Fig. 22 shows the corresponding comparison in the intermediate energy range. The agreement between calculation and experiment for the spectral modifier made on an absolute basis is extraordinarily good, presumably because some of the difficulty in calculating low-energy neutron transport through stainless steel is mitigated by the presence of the polyethylene. Table 6 shows a comparison of calculations with the Bonner ball measurements for the spectrum modifier assembly, as shown in Fig. 19. The calculations include effects of scattering around the detector, and the comparisons again are excellent indicating that no severe disagreements occur in the energy range not covered by the comparison with unfolded spectra.

The measurements of the streaming effect were performed in the configuration shown in Fig. 17 with and without a 4-in. lucite section having the same hole pattern as the lower part of the shield insert and placed directly below the insert so as to increase the streaming effect. The idea was to measure the maximum streaming effect which may occur at a forward angle. Since the streaming effect is expected to be small, several measurements were made for each insert.

For each measurement, the reactor and detector were repositioned, and from the three or four measurements that were performed for each configuration, one can judge the reproducibility of the experiment and estimate the experimental error. The measurements for the configuration having the maximum number of offsets were taken as the base, and each streaming measurement is referenced to this base as a relative streaming effect. Since, as mentioned previously, the maximum offset insert is not truly homogeneous, a calculation of the shield must be multiplied by a heterogeneous factor for this effect and one for the streaming effect as measured.

SNAP Plus 10-in. 304 Stainless Steel, 1-in. Polyethylene,
 1/4-in. Boral.

Fig. 22. Comparison of Measured and Calculated Neutron Spectra
 (70 < E < 1500 keV) for the Spectrum Modifier.

Table 6

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.

Table 7 shows a summary of the streaming measurements. Even for the case with straight-through holes the streaming effect is not large, and for the insert with an offset the streaming effect is almost within the statistics of the experiment. Similar effects are observed for the Y slots representing the space between the assemblies.

It was found during the course of the experiment that the disagreement in the absolute comparison between calculation in the experiment for the nearly homogenized insert was greater than the streaming effect. An additional measurement for the homogenized insert was made in the configuration shown in Fig. 23 where the reactor detector separation is maintained and the spectrum modifier is lowered onto the shield insert. Therefore, there are two assemblies (Figs. 17 and 23) for which comparisons with absolute calculations were made. Table 8 shows the results of these comparisons. The calculations were performed with DOT using a 100-angle biased quadrature, P_3 scattering, and the 50-group cross sections used at ORNL for FTR design support calculations. The agreement with experiment is really quite good considering the large amount of steel in the modifier and shield prototype and the complexity of the geometry.

The disagreement, however, is larger than the maximum streaming effect. Furthermore, the disagreement differs for the two configurations. A possible explanation for this difference is that the most significant deficiency in the cross-section set is in the definition of the deep iron windows between 20 and 500 keV. As has previously been observed in the iron benchmark comparisons,³ calculations influenced by this energy range agree better with experiment when the ratio of collided to uncollided flux is large. Configuration #2, which has the spectrum modifier midway

Table 7

Summary of Streaming Factor Measurements

<u>Configuration</u>	<u>Streaming Factor</u>
Maximum offsets	1.0 (base)
No offsets	1.10 ± 0.02
One offset	1.03 ± 0.02
200-mil Y slot	1.09 ± 0.02
100-mil Y slot	1.03 ± 0.02

Fig. 23. Modified Configuration for ORNL-TSF Prototype Grid-Plate Shield Experiment.

Table 8

Comparison of Experiment and Calculation For
TSF Grid-Plate Shield Experiment

Absolute Count Rate in 5-in. Bonner Ball	Configuration No. 1 (cpm/W)	Configuration No. 2 (cpm/W)
Experiment	1.373×10^5	9.73×10^4
Calculation	1.183×10^5	7.38×10^4
Ratio (exp/calc)	1.16	1.32

between the reactor and detector, tends to emphasize the uncollided component more than configuration #1. This possibly explains the increasing disagreement.

Although the lower axial TSF experiment was intended to be a prototype test, the experience obtained in this test argues against the direct use of prototypes and in favor of the use of integral experiments to gain understanding of basic transport effects. The test occurred late in the design schedule; however, the current shield design has departed significantly from this prototype. The major change is the use of 316 stainless steel rather than 304 stainless steel, and this will markedly affect the bias obtained from the comparisons with calculation, shown in Table 8.

Of lesser consequence, the coolant hole diameters and overall steel volume fractions are different in the current design. Also, the fact that the two configurations shown in Table 8 give different bias factors shows that even in a prototype where the incident spectrum and detector are appropriately chosen, differences in overall geometry and angular distribution may cause the bias factor to be inappropriate. It appeared to be more cost effective to use the integral experiments as tests, indicating need to improve the cross sections or transport calculation so that the indicated errors are eliminated rather than to attempt to obtain the appropriate bias factors from detailed prototype experiments.

This experiment has been most useful in showing that streaming in this geometry is not overly important, where the total flux is concerned, in providing a test of the stainless-steel cross sections, and in demonstrating the need for concern for the bias and uncertainty factors discussed above.

The ORNL calculational effort in support of FTR shield design is continuing with emphasis on the problem of streaming in the vessel support area, including the effects of stored fuel. In the experimental program, the sodium experiment will be followed by an experiment concerned with problems of neutron streaming around the coolant pipe shield penetrations into the heat exchanger vault. When the FTR design is finished, the shield calculations will be updated and uncertainty estimates will be re-examined using the final comparisons with the integral experiments.

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