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# THE ORNL LMFBR EXPERIMENTAL SHIELDING PROGRAM

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## ABSTRACT

An ORNL experimental program supporting the development of LMFBR shield design methods has over the past five years included investigations of the fast neutron fluxes incident on the FFTF grid plate, the penetration of neutrons through very large thicknesses of sodium and iron, the streaming of neutrons in sodium gaps between iron-pin arrays, the production of secondary gamma rays in an iron slab simulating the FFTF top head, the streaming of neutrons around sodium coolant pipes, and the attenuating characteristics of various shields designed by General Electric, Westinghouse, and Atomics International. These experiments and their accompanying analyses not only have directly influenced some current shield designs, but have also prompted specific improvements in the radiation-transport calculational methods, as well as re-evaluations of some cross-section sets. The result is an increasing confidence in the combined experimental and analytical approach for developing accurate shield design techniques.

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An experimental program supporting the development of shielding design methods for the Liquid Metal Fast Brèeder Reactor (LMFBR) has been in progress at ORNL since 1969. The major purpose of the program has been to provide accurate results from integral experiments against which methods for calculating radiation transport could be tested. At the time the experiments were initiated, the methods being developed were primarily the discrete ordinates method and a multigroup Monte Carlo technique. In both of these approaches to solving the Boltzmann equation, approximations are required in representing the cross sections and in treating the angular and spatial mesh of a given LMFBR problem. These approximations are tested by using them in the analyses of integral shielding experiments which simulate the particular problems inherent in the LMFBR reactor shield design.

The LMFBR shield experiments have been carried out at the ORNL Tower Shielding Facility (TSF) by F. J. Muckenthaler, who directs the experimental program, and L. B. Holland, who operates the TSF. The



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radiation source is a spherical 1-MW, water-cooled reactor (1) having an aluminum-uranium core approximately 3 ft in diameter. The control rods for the reactor are also spherically symmetrical and are contained within a 2-ft-diam, water-filled region in the center of the core. This reactor configuration simplifies calculations of the radiation leakage from the source, which is an integral part of the analyses of the experiments. The analyses have been carried out principally by R. E. Maerker and R. L. Childs.

Figure 1 shows the operating area at the TSF where the experiments are conducted. The facility is designed to allow very high radiation levels in the operating area so that experiments can be carried out with very thin shields as well as thick shields. This flexibility in the operation is accomplished by arranging for the personnel to remain in a heavily shielded underground building.

A typical shielding configuration is shown slightly to the right of center in Figure 1. It is placed adjacent to a large concrete interface that provides a flat surface as the mating point for the shield configuration and fits around a spherical lead-water reactor shield on the opposite side. A 15-in.-diam radiation beam port extending from the reactor core through 4 ft of the lead-water shield emerges from the front of the concrete interface to provide the source for the experiments. Shielding samples are usually constructed of slabs of materials approximately 5 ft square and varying in thickness from 2 in. to 1 ft.

Other equipment shown in Figure 1 are a 12-ft-diam shield for a 5-in. sodium iodide gamma-ray spectrometer (lower right) and a new beam shield for the reactor (upper left of center). The new shield beam port has a larger diameter (3 ft) and a shorter length (1 ft) than the old shield. It is anticipated that the use of this shield will extend the range of attenuation that can be measured by a factor of 50 to 100.

Figure 2 presents a schematic of the earliest experiment which was conducted for the LMFBR shielding program at ORNL. This experiment was undertaken because of a serious concern with regard to radiation damage to the grid plate of the Fast Flux Test Facility (FFTF). The grid plate shielding at that point in the design was to be an array of 3/8-in.-diam stainless steel rods contained in the lower portion of a fuel bundle. The shield design had been based on calculations in which homogeneous materials were assumed, which raised a concern regarding possible streaming of high-energy neutrons in the rather large sodium gaps between the steel rods.

To test the calculational methods, a shield configuration 24 in. thick and 18 in. in diameter was constructed of iron rods immersed in sodium and the neutron transport in this shield was measured (2). The volume fraction of iron in the shield was approximately 38%. Neutron transport in shielding samples of pure sodium and pure iron were also studied.

The source of neutrons for this experiment was a SNAP reactor which had been in use at the TSF during an earlier SNAP shielding program.

This reactor was used as a source because its emergent neutron energy spectrum had already been carefully measured. Measurements were made of the transmitted neutron spectra through the samples of sodium and iron and also through an iron-pin array. The detectors were an NE-213 fast-neutron spectrometer and a Benjamin-spectrometer system utilizing hydrogen-filled proportional counters. Table I (2) compares the calculated and measured streaming factors, obtained by dividing a cell calculated or measured neutron flux by a calculated neutron flux for the homogeneous shield. As can be seen, the DOT cell calculated fluences (with streaming) were a factor of about 10 higher than the ANISN homogeneous shield calculation for energies above 3 MeV, whereas the measured fast-neutron fluences were roughly a factor of 20 higher than the DOT homogeneous calculation. This indicates that the DOT cell calculation underestimates the high-energy flux by almost a factor of 2 for all neutron energies, primarily because there was a strong uncollided flux component which was not adequately treated in the calculations. This experiment demonstrated that the use of simplified calculations had led to an underprediction of the radiation flux levels at the FFTF grid plate. Therefore, this shielding configuration was abandoned in favor of a more homogeneous design having a higher volume fraction of stainless steel.

Figure 3 (3) shows the arrangement for another important class of experiments in which the gamma-ray spectra transmitted through various shielding configurations were measured. For the top head shielding configuration of the FFTF reactor, the major radiation component contributing to the dose results from gamma rays which are produced by neutron interactions in the shielding materials themselves, rather than gamma radiation which originates in the core or in the sodium above the core. A series of experiments were undertaken at the TSF to study the secondary gamma rays produced both in mild steel shields and in shields typical of the FFTF inner radial shield, which consists of stainless steel and sodium. Figure 3 shows the general arrangement for utilizing the sodium iodide spectrometer system to measure the transmitted gamma-ray spectra from the shield configurations. The 5-in. sodium iodide crystal is contained in its large spherical shield, and because of the detector's extreme sensitivity, it was located approximately 40 ft from the reactor. The detector views the shield at an angle of 45° in order to minimize the effect of core gamma rays on the measurement.

Figure 4 (3) shows the arrangement for studying the gamma rays produced in a solid iron shield approximately 22 in. in thickness, with and without additional layers of borated polyethylene and iron. The configurations are surrounded by paraffin bricks which contain lithium carbonate to suppress capture gamma rays in the paraffin. A lead and concrete shield is also provided to eliminate gamma rays which emerge from the side of the slabs. A typical unfolded spectrum for the iron configuration alone is shown in Figure 5 (3), in which the primary iron capture lines, especially that at 7.64 MeV, can clearly be seen, as well as the hydrogen (in paraffin) line at 2.23 MeV. Table II (4) shows a comparison of calculated and measured gamma-ray spectra and indicates that the gamma-ray spectra can be calculated with accuracy from about 20 to 30 percent over the energy range measured. This discrepancy is due primarily to difficulties in calculating the absolute neutron intensity

Figure I. Operating Area at Tower Shielding Facility

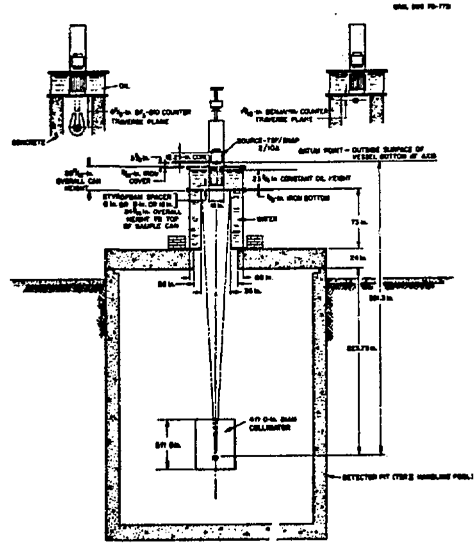


Figure 2. Configuration for FFTF Grid Plate Measurements

ORNL-DWG 74-9008

Table I. Neutron Streaming Factors as a Function of Energy for Pin Arrays

| Group[c] Upper Energy (MeV) | Cell Calculation[a] 24-in. | Exp./Calculation[b] 24-in. |
|---|---|---|
| 14.9 | 8.76 | 13.5 |
| 11.1 | 5.32 | 8.72 |
| 8.19 | 6.69 | 11.3 |
| 6.07 | 9.76 | 17.6 |
| 4.49 | 9.49 | 24.1 |
| 3.33 | 4.01 | 7.16 |
| 1.83 | 2.33 | 4.81 |
| 1.00 | 1.18 | 1.55 |

[a] Ratio of a DOT Fe-Na cell calculation to a homogeneous Fe and Na shield ANISN calculation.

[b] Ratio of measured transmitted neutron flux to a DOT calculated flux transmitted through a homogeneous sodium iron-pin array.

[c] $\frac{\Delta E}{E} = 0.10$

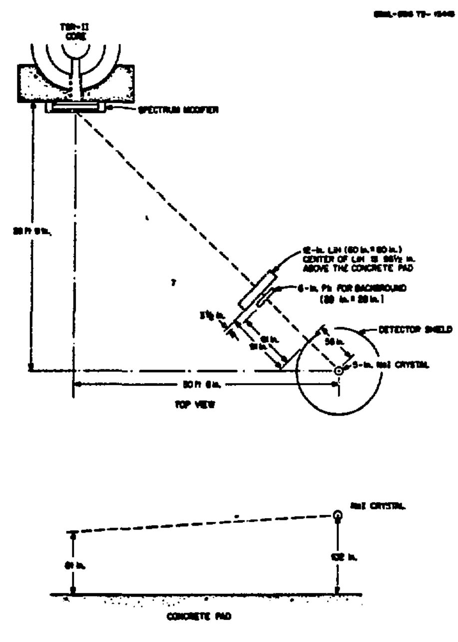



Figure 3. Reactor-Detector Geometry for Measurements of Secondary Gamma-Ray Production in Shields

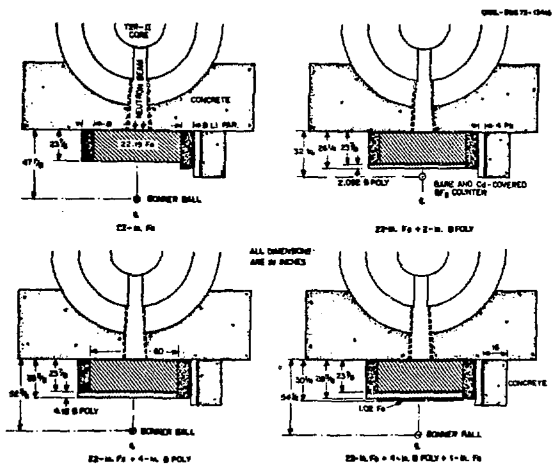



Figure 4. Experimental Configurations for Iron Gamma-Ray Production Studies

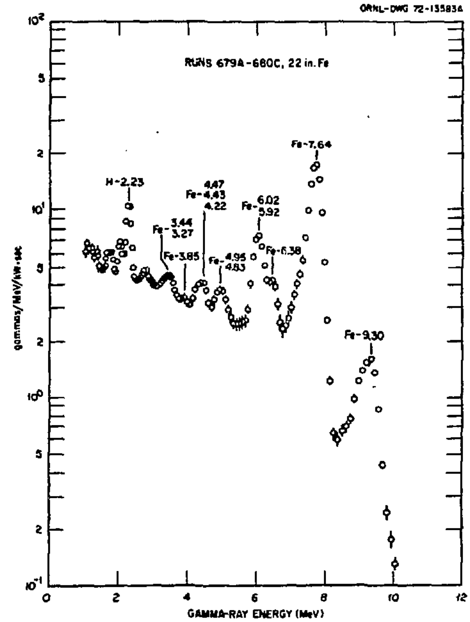
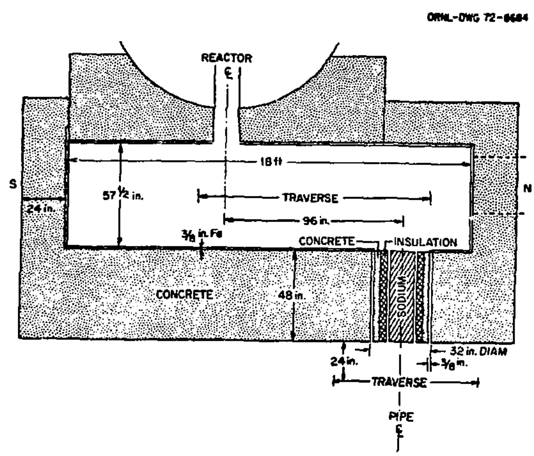



Figure 5. Spectra of Gamma Rays Produced in Iron

ORNL-DWG 74-9009

Table II. Comparison of Calculated and Measured Gamma-Ray Fluxes at an Angle of 45 deg Beyond 22 in. of Iron

| Energy Interval (MeV) | Gamma-Ray Flux ($\gamma$'s$\cdot$cm$^{-2}\cdot$kW$^{-1}\cdot$sec$^{-1}$) | | Calc. Flux / Meas. Flux |
|---|---|---|---|
| | Calculated | Measured | |
| 10–8 | 9.5317(−3) | 9.0462(−3) | 1.05 |
| 8–7 | 5.4129(−2) | 5.4999(−2) | 0.98 |
| 7–6 | 1.6192(−2) | 2.7072(−2) | 0.60 |
| 6–5 | 1.6090(−2) | 2.1960(−2) | 0.73 |
| 5–4 | 1.9990(−2) | 2.3906(−2) | 0.84 |
| 4–3.5 | 8.0764(−3) | 1.2008(−2) | 0.67 |
| 3.5–3 | 1.2178(−2) | 1.2991(−2) | 0.94 |
| 3–2.5 | 1.1024(−2) | 1.2476(−2) | 0.88 |
| 2.5–2 | 1.5078(−2) | 1.7037(−2) | 0.89 |
| 2–1.6 | 1.0625(−2) | 1.6578(−2) | 0.64 |
| 1.6–1.2 | 8.8283(−3) | 1.2882(−2) | 0.69 |
| 1.2–0.9 | 6.0973(−3) | 8.8242(−3) | 0.69 |
| 0.9–0.6 | 8.3371(−3) | 1.0686(−2) | 0.78 |
| 0.6–0.4 | 7.8259(−3) | 1.2852(−2) | 0.61 |
| 0.4–0.21 | 4.7135(−3) | 7.6231(−3) | 0.62 |
| 10–0.21 | 2.087(−1) | 2.609(−1) | 0.80 |




Plan View of Geometry for Phase Two of TSF Pipe Chase Streaming Experiments.

Figure 6. Mockup of Pipeway and Pipe Chase for Neutron Streaming Measurements

using few-group discrete ordinate calculations. Similar gamma-ray measurements have been made for solid iron shields varying in thickness from 10 to 34 in. for typical FFTF radial shield configurations, and for shield configurations which are currently being considered for the Clinch River Breeder Reactor (CRBR).

Another problem which was studied early in the shielding program is that of neutron streaming through the large penetrations in the shield which must be allowed for the sodium coolant pipes. Figure 6 (5) shows the geometry for one of two experiments which were conducted to study neutron streaming in the sodium pipe chaseways. In this experiment the neutrons from the reactor beam were scattered in the large steel-lined pipeway whose inside dimensions were approximately 18 ft by 14 ft by 5 ft. Measurements were made of the neutron distributions within the pipeway and also of the neutron distributions along the center line of the sodium pipe penetration and along a horizontal traverse beyond the sodium pipe penetration. Typical results from this experiment are shown in Figure 7 (5), which presents the measurements with a 5-in. Bonner ball along the center line of the pipe chase in the outer wall. As can be seen, the measurements and calculations agree to within approximately 10-20 percent over the entire range of the measurements. This close agreement with calculations was typical for the measurements taken in this experiment. Analysis of this experiment indicated that the primary source of scattered neutrons into the pipe chase resulted from scattering near the lip of the reactor beam collimator. As a result of this study, particular care was taken in treating the scattering within the sodium pipes which extended into the plenum in the actual reactor configuration. Also, uncertainty factors for the neutron flux in the primary heat exchanger in the FFTF were estimated from the results of the analysis.

A major part of the experimental program at the TSF was developed as a cooperative effort between ORNL and General Electric, Westinghouse, and Atomics International. Each company chose a particular LMFBR shielding problem, and designed and developed a mockup to be used at the TSF to study the problem. General Electric proposed to study the effect of using shielding in a rod configuration as a possible technique for reducing the cost of the shield fabrication. They provided a series of mockups for testing the neutron streaming through arrays of boron carbide-filled rods and stainless steel rods. Boron carbide rods were considered to be part of a radial shield design; therefore neutron streaming was studied in the direction perpendicular to the axis of the rods. Consideration of the possible use of large-diameter stainless steel rods in the upper axial shield led to a study of neutron streaming in the direction parallel to the axis of these rods.

Westinghouse provided a configuration which simulated the radial shield being considered for the first demonstration plant. This shield consisted of a radial blanket configuration followed by a thick layer of stainless steel and an additional layer of sodium. As part of the Westinghouse preanalysis of the experiment, a spectral modifier was designed which would alter the neutron leakage from the TSR-II so that it closely resembled the spectral leakage from an LMFBR fast reactor core.

Atomics International provided a mockup of the lower axial shield configuration with changeable plugs to study the effect of control rod penetrations. In addition to a plug which made the shield essentially homogeneous, plugs were provided to simulate a sodium-filled penetration and control rod configuration. Experimental studies were done to study the severity of streaming to be found with various configurations.

Figure 8 (6) shows the experimental configuration for one of the General Electric shielding configurations. This is the case for a single row of 5-1/4-in. boron carbide-filled shield rods. Not shown in the figure are aluminum triflutes which were used to simulate sodium coolant in the space between the 5-1/4-in. rods. The rods were preceded by the Westinghouse spectral modifier in order to provide a typical fast reactor neutron spectrum as the source for the experiment. The single row case was anticipated to have the most serious streaming problems of all of the General Electric samples. Typical results are shown in Figure 9 (6), which presents the comparison of Hornyak button radial traverses directly behind one row of 5-1/4-in. $B_4C$ rods and behind a homogeneous slab configuration. This figure indicates that the neutron flux penetrating the rod configuration is slightly less than that penetrating the homogeneous slab, which may be due to minor differences in the density of the two sets of boron carbide powder. The streaming, as indicated by the ripples in the radial traverse, is shown to be on the order of 10 percent. It was found for shield configurations containing four or more layers of rods 1 to 5 in. in diameter that the streaming could essentially be ignored in the shielding calculations. This of course greatly simplifies the radiation transport analysis problems in a shield constructed of layers of rods.

Figure 10 (7) shows the configuration for the Westinghouse radial shield configuration, which was studied intensively with respect to the spectra of neutrons and gamma rays penetrating the shield. This shield mockup was also utilized to study the accuracy of predicting the fission rate for fuel stored in the outer sodium region of the CRBR shield. The mockup was extended to study the counting rate of a low-level flux monitor located just external to the reactor pressure vessel. The mockup consists of the neutron spectral modifier which contains 4 in. of stainless steel, 6 in. of sodium, and 1 in. of boral. The spectral modifier is followed by a $UO_2$-filled blanket region, approximately 9 in. thick, and this is in turn followed by 5 in. of stainless steel, by 5 in. of Inconel, by 14-in. iron and stainless steel slabs, and by 1 ft of sodium. As part of the stored-fuel fission rate study, both uranium and plutonium fission chamber measurements were made behind this configuration and were compared with calculated results. This comparison is shown in Table III (8), which indicates that the counting rate predicted by the calculations is low by a factor of about two. The cause of this discrepancy is currently assigned to uncertainties in the nickel and chrome cross sections for the stainless steel of the radial shield.

One of the important categories of the LMFBR shielding experiments performed at the TSF is the so-called benchmark experiment, which is performed to study deep neutron penetration in single materials, such as iron, stainless steel, sodium, and boron carbide. Figure 11 (9) shows

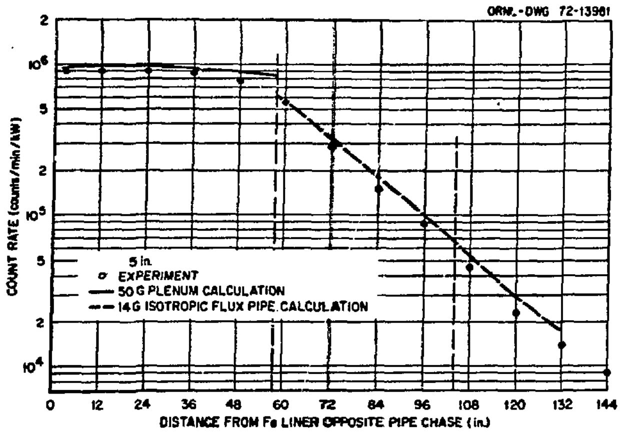



Figure 7. Comparison of 5-in. Bonner Ball Neutron Measurements with Calculations for the Pipe Chase Experiment

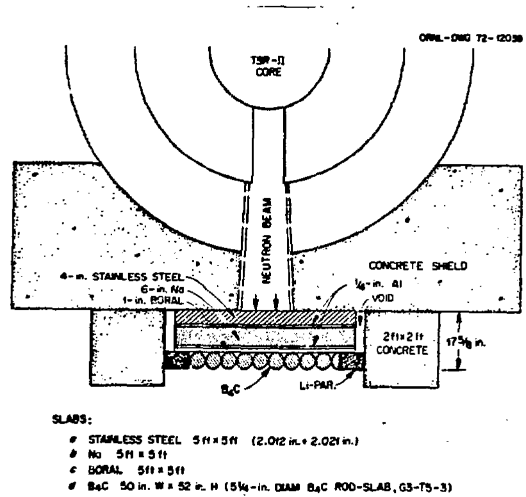



Figure 8. Spectral Modifier and GE $B_4C$ Rod Configuration

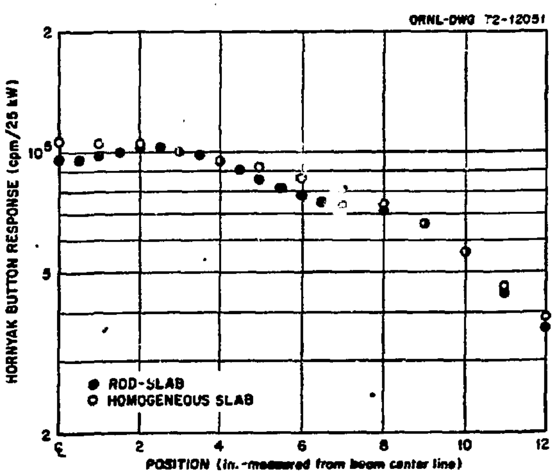



Figure 9. Comparison of Hornyak Button Measurements Beyond GE $B_4C$ Rod Configuration and Homogeneous Configuration

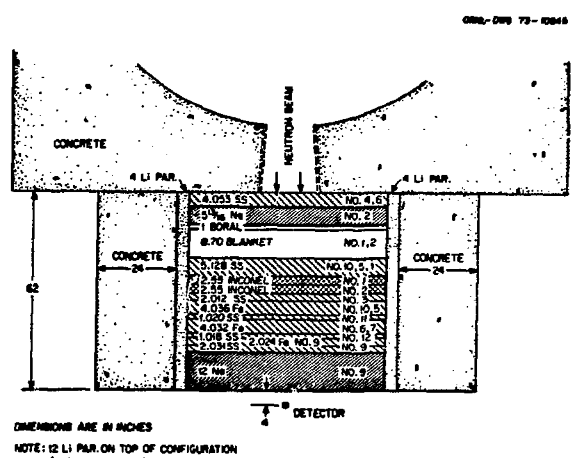



Figure 10. Westinghouse Radial Shield Configuration

Table III. Comparison of Calculated and Measured Counter Responses in the DEMO Stored Fuel First Fission Experiment

| | Detector Responses cpm/mW | | | |
|---|---|---|---|---|
| | $^{235}U_{cd}$ | $^{235}U_{bare}$ | $^{239}Pu_{cd}$ | $BF_{3\,cd}$ |
| **Calculated** | 3,863 | 4,287 | 694 | 98,077 |
| **Measured** | 5,894 | 9,973 | 1,457 | 145,510 |
| **Ratio** | 0.655 | 0.430 | 0.476 | 0.674 |

the configuration which was used to measure the neutron transport through 15 ft of sodium. This experiment was done with a very large sample so that the radiation transport effects would not be dominated by side leakage. The sodium tanks are shown in Figure 12 (9). These tanks are 11 ft in diameter and are surrounded by approximately 2 ft of concrete on all sides. The sodium is contained in aluminum cans having walls 1/2 in. thick.

In the sodium experiment, measurements were taken using Bonner ball detectors, which are designed to measure neutron fluence with different energy-dependent weighting functions. The Bonner ball is essentially a spherical boron trifluoride proportional counter surrounded by various thicknesses of polyethylene layers and an outside cadmium cover. Bonner ball intensity measurements were made as a function of thickness at 2-1/2-ft intervals for the sodium samples. The results are presented in Figure 13 (9). In this figure the counting rates are normalized in the bare beam readings giving a counting rate of 1 with no shielding present. The bare Bonner ball is the $BF_3$ counter with nothing around it and responds to thermal neutrons. The cadmium-covered ball has a 1/v cross section above the cadmium cutoff. The modified Bonner ball was shielded by a thick layer of $^{10}B$ and was designed to measure the neutrons which would penetrate the steel shielding above the sodium pool of the FFTF.

A comparison of the measured and calculated results for the Bonner ball detectors behind 15 ft of sodium (not included here) shows an agreement within 30 percent for most of the Bonner balls. The calculated results were obtained using the multigroup Monte Carlo code MORSE and the two-dimensional discrete ordinates code DOT III. One hundred group data sets were developed from both preliminary and final versions of the ENDF/III set (MAT=1156) for sodium for use in the calculations. The modified Bonner ball responded only to neutron fluxes above 50 kV and for the spectrum penetrating the 15 ft of sodium, the counting rate peaked in the energy region between 100 and 500 kV. Even though the counting rates for this detector beyond 15 ft of sodium were quite low, the agreement between the measurement and calculation was still near 30 percent.

Figure 14 (9) shows the experimental results for the Benjamin neutron spectrometer for the measurement behind 12-1/2 ft of sodium. The energy range covered in these measurements was from 70 kV to 1-1/2 MeV. The two calculated spectra shown are the result of DOT-III calculations using the preliminary and final cross-section sets. Both integral and shape comparisons favor the preliminary set. The calculated spectra using the final ENDF/B-III cross section sets are 50 to 60 percent higher than the measured results. This discrepancy is reflected in the ratio of the integrals between the calculated and measured spectra, which is 1.61.

Figure 15 (9) shows the measurement of the fast neutron spectra which penetrates 10 ft of sodium. These measurements were taken with the NE-213 neutron spectrometer and cover the energy range from 1 to 10 MeV. Here again the calculations are higher (approximately 30 percent) than the measurements when the ENDF/B-III cross section data sets are

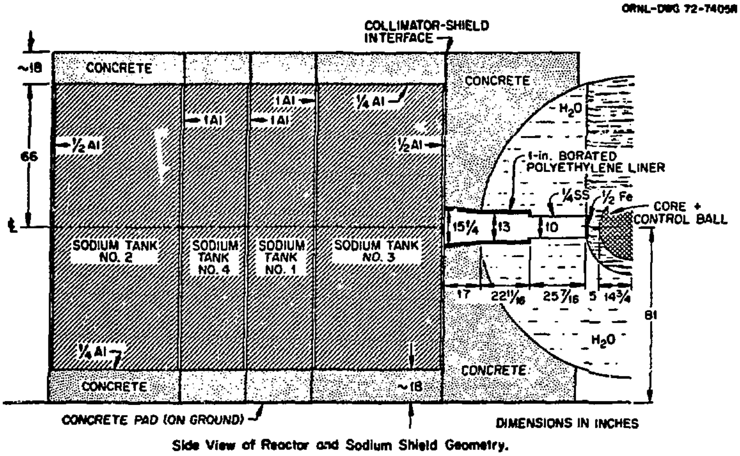



Figure 11. Configuration for Measurements of the Deep Penetration of Neutrons in Sodium

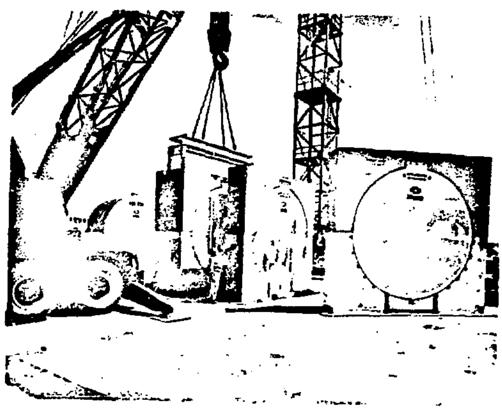


Figure 12. 15-ft-diam Sodium Tanks Surrounded by Concrete

used. This may indicate that the fast neutron cross sections for the sodium used in ENDF/B-III might be 5 percent too low.

In addition to the sodium benchmark experiments, a very detailed study has been completed on neutron transport through mild steel, which consists of 99 percent iron. The initial detailed analysis of these experiments indicated that problems existed in the iron total cross sections for the energy region from 24 kV to 3 MeV. This led to new cross-section measurements and revisions in the valleys of the cross-section data used in the calculations. A comparison of the measurements with subsequent calculations of the neutron spectra behind 22 in. of iron is shown in Fig. 16 (4). As a result of this study, the ENDF/B-IV cross-section set for iron is considerably revised from the ENDF/B-III set.

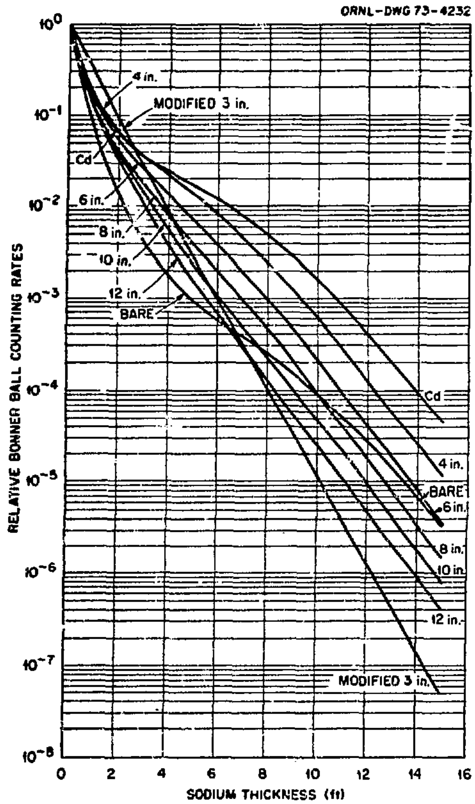



Figure 13. Relative Bonner Ball Counting Rates Beyond Various Thicknesses of Sodium

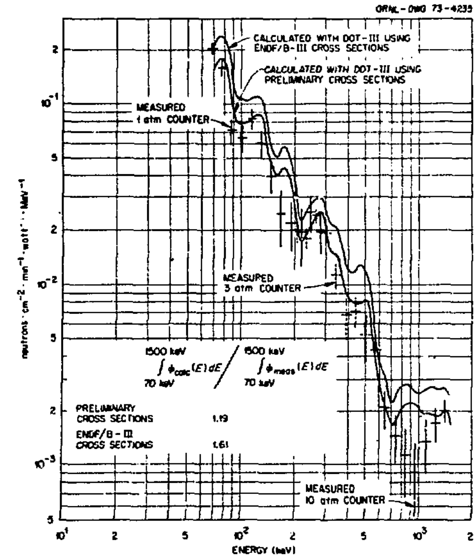



Figure 14. Comparison of Measured and Calculated Low-Energy Neutron Spectra Beyond 12.5 ft of Sodium

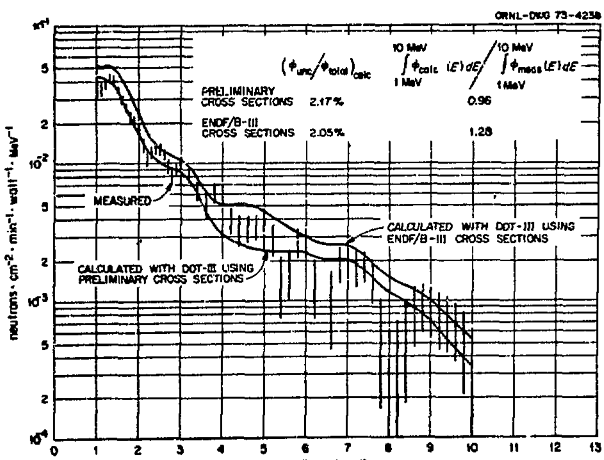
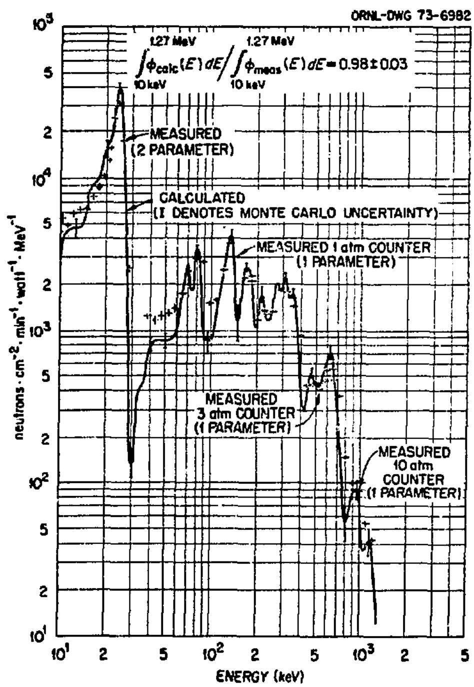



Figure 15. Comparison of Measured and Calculated Fast Neutron Spectra Beyond 10 ft of Sodium




Comparison of Calculated and Measured Spectra Behind 22 in. of Iron

Figure 16. Comparison of Calculated and Measured Spectra Behind 22 in. of Iron

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