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Review of ORNL-TSF Shielding Experiments for the Gas-Cooled Fast Breeder Reactor Program

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REVIEW OF ORNL-TSF SHIELDING EXPERIMENTS FOR THE GAS-COOLED
FAST BREEDER REACTOR PROGRAM

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ABSTRACT

During the period between 1975 and 1980 a series of experiments was performed at the ORNL Tower Shielding Facility in support of the shield design for a 300-MW(e) Gas Cooled Fast Breeder Demonstration Plant. Although the GCFR program was cancelled in 1980, much of the information gained from the experiments and their analyses is generic and therefore applicable to other systems. In each case the approach was to mock up a region of the GCFR reactor core and/or shield and to measure the intensities (and energies when possible) of reactor radiation penetrating through the region. In addition, calculations of the penetrating radiations were performed, using the same methods that were to be applied or had been applied to the GCFR system itself. When required, new calculational techniques were developed. Agreement between the calculated and measured results validated the method for application in the analysis of the GCFR shielding design. This report reviews the experiments and calculations, which included studies of: (1) neutron streaming in the helium coolant passageways in the GCFR core; (2) the effectiveness of the shield designed to protect the reactor grid plate from radiation damage; (3) the adequacy of the radial shield in protecting the PCRV (prestressed concrete reactor vessel) from radiation damage; (4) neutron streaming between abutting sections of the radial shield; and (5) the effectiveness of the exit shield in reducing the neutron fluxes in the upper plenum region of the reactor. In general, good agreement between the calculated and measured fluxes was obtained for all experiments.

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1. INTRODUCTION

- 1.1. Background
- 1.2. ORNL Shielding Analysis Philosophy
- 1.3. The Tower Shielding Facility
- 1.4. Organization of Report
- 1.5. References

1.1. BACKGROUND

Studies of the concept of a Gas-Cooled Fast Breeder Reactor (GCFR) using helium as the coolant were initiated in the United States in 1962 by the General Atomic Company (GAC). In 1963 the Atomic Energy Commission (AEC) began funding the company's fuel element development work, and within a few years several U.S. utilities were supporting GAC in a broad GCFR program. By 1975 the total number of participating utilities had increased to 57, and 70 rural electric cooperatives and three European utility companies had joined the effort.

The first conceptual design of a 300-MW(e) GCFR demonstration plant was completed in 1970 (ref. 1) and was followed by intensive studies of the various systems and components essential to the final design. The utility-supported program at GAC emphasized work on the overall plant design, licensing, and siting, while the AEC-supported program included, in addition to the fuel element work, studies on reactor physics and component development. AEC also supported a number of other GCFR programs -- at Argonne National Laboratory (ANL) and Oak Ridge National Laboratory (ORNL), as well as at other laboratories -- in the areas of core flow loop testing, critical experiments, irradiation tests, safety experiments, shielding analysis, etc. This support was continued by the Energy Research and Development Administration (ERDA) and later by the Department of Energy (DOE).

Insofar as was practical, the design of the GCFR demonstration plant was based on a minimum extrapolation of the component technology for the High-Temperature Gas-Cooled Reactor (HTGR) and on the physics and fuel element technology of the Liquid Metal Fast Breeder Reactor (LMFBR). By 1974, advances in the HTGR component technology, together with detailed design and safety reviews and architect-engineer studies for the GCFR plant itself, had resulted in major changes to the 1970 design and a new design for the demonstration plant was released (ref. 2).

As part of its GCFR work, ORNL was asked by GAC to review the shielding system of the new GCFR design and to identify areas of concern. The concerns were subsequently discussed with GAC, whereupon GAC requested that ORNL develop a shielding program that would either confirm the design or point out regions in which problems existed.

Basically, ORNL proposed that the shielding for the GCFR be analyzed in a program similar to one that had been conducted for the Fast Flux Test Facility (FFTF) (ref. 3). By early in 1975, GA and ORNL had outlined an ORNL-GCFR shielding program that was to include a series of radiation transport calculations for specific regions of the GCFR system. In addition, the program was to include a series of supporting experiments and their analyses.

When the U.S. GCFR program was cancelled in 1980, six of eight planned experiments had been completed. Because much of the information gained from these experiments and their analyses is generic and therefore applicable to other reactor systems, it was felt that a review of the experiments in a single document would be of considerable value. This report presents that review.

1.2. ORNL SHIELDING ANALYSIS PHILOSOPHY

Inherent in the shielding analysis approach used by ORNL is the experimental verification of the calculational methods that are to be applied in the analysis of a particular reactor shielding system. As a result of an earlier lengthy methods development program, largely supported by the LMFBR program and the Defense Nuclear Agency, when the GCFR shielding program was initiated ORNL already had on hand two powerful computer codes for calculating radiation transport, and both had undergone experimental validation for application to other reactor-shield systems. One code was the discrete ordinates transport code DOT (ref. 4) and the other was the Monte Carlo code MORSE (ref. 5).

Whenever DOT or MORSE is applied to a new system, however, a certain amount of revalidation is necessary, especially if new calculational techniques are to be developed and introduced in the codes for calculations of reactor and/or shield regions that differ significantly from those in previously analyzed systems. In the validation experiments the region of concern is mocked up and measurements are made of the penetration of radiations through it. The experiments are then analyzed with the same calculational methods that have been or are to be applied to the actual reactor system. Agreement between the measured and calculated results ensure that the methods can be applied with confidence to similar regions in the reactor. When, on occasion, agreement is not

reached, the difference between the measured and calculated results is used as a bias factor to introduce conservatism in the reactor design.

In the case of the GCFR, several experiments were deemed necessary to test the application of the transport methods to a helium-cooled system, primarily because the methods had never been applied to a geometry containing such a complicated array of low-density regions through which neutrons born deep within the core could stream to regions outside the core. Predicting the streaming was expected to severely test the methods and to require the development of special techniques. Thus experiments were required both to aid in the development of the techniques and to test them.

1.3. THE TOWER SHIELDING FACILITY

For over 20 years the principal experimental shielding facility at ORNL has been the Tower Shielding Facility (TSF). The facility was originally built for studies of radiation interactions within the atmosphere (for the Aircraft Nuclear Propulsion Program) and was designed so that a reactor could be suspended between four 315-ft-high towers — hence its name. For current shielding experiments, however, the facility's reactor, called the Tower Shielding Reactor II (TSR-II), remains on the ground and is surrounded by a shield penetrated by a single large collimator. Experimental mockups are placed against the exit edge of the collimator, and reactor radiations emerging from the collimator are used as the source for the experiment. Measurements are made with detectors placed both behind and within the experimental configurations. A view of the shielded TSR-II at the TSF site is shown in Fig. 1.1.

1.4. ORGANIZATION OF REPORT

The six GCFR experiments performed as part of this program are described here in Chapters 5 through 10. By way of introduction to the experiments, the particular GCFR shield regions of concern are identified in Chapter 2, and the experimental and calculational techniques are described briefly in Chapters 3 and 4 respectively.

Fig. 1.1. The Tower Shielding Facility. Exit edge of collimator through TSR-11 beam shield shown at left. (Note: Geometry of reactor and shield is shown in Fig. 5.3, page 5-12.)

1.5. REFERENCES

1. "Gas-Cooled Fast Breeder Reactor Demonstration Plant Nuclear Steam Supply System." Gulf General Atomic, Report GA-10064 (May 15, 1970).
2. "300-MW(e) Gas-Cooled Fast Breeder Reactor Demonstration Plant," General Atomic Company, Report GA-A13045 (July 15, 1974)
3. Lorraine S. Abbott, F. R. Mynatt, W. W. Engle, Jr., and F. J. Muckenthaler, "Review of ORNL Radiation Shielding Analyses of the Fast Flux Test Facility Reactor," ORNL-5027 (July 1975); Lorraine S. Abbott, F. R. Mynatt, W. W. Engle, Jr., and Margaret B. Emmett, "Review of ORNL Radiation Shielding Analyses of the Fast Flux Test Facility Reactor (1975-1976)," ORNL-5166 (June 1976).
4. F. R. Mynatt, F. J. Muckenthaler, and P. N. Stevens, "Development of Two-Dimensional Discrete Ordinates Transport Theory for Radiation Shielding," Computing Technology Center, UCC-ND, CTC-INF-952 (1969); W. A. Rhodes and F. R. Mynatt, "The DOT-III Two-Dimensional Discrete Ordinates Transport Code," ORNL-TM-4280 (Sept., 1973); W. A. Rhodes, D. B. Simpson, R. L. Childs, and W. W. Engle, Jr., "The DOT-IV Two-Dimensional Discrete Ordinates Transport Code with Space-Dependent Mesh and Quadrature," ORNL/TM-6529 (January 1979); W. A. Rhodes and R. L. Childs, "An Updated Version of the DOT-4 One- and Two-Dimensional Neutron/Photon Transport Code," ORNL-5851 (in review).
5. E. A. Straker, P. N. Stevens, D. C. Irving, and V. R. Cain, "The MORSE Code -- A Multigroup Neutron and Gamma-Ray Monte Carlo Transport Code," ORNL-4585 (1970); M. B. Emmett, "The MORSE Monte Carlo Radiation Transport Code System," ORNL-4972 (February, 1975).

2. THE GAS COOLED FAST BREEDER REACTOR

2.1. Principal Features of the Nuclear Steam Supply System

2.2. Reactor Shielding

2.1. PRINCIPAL FEATURES OF THE NUCLEAR STEAM SUPPLY SYSTEM

The 1974 concept of the nuclear steam supply system for the 300-MW(e) GCFR demonstration plant is depicted in Figs. 2.1 and 2.2. In this design the reactor core consisted of a 39.2-in.-high fueled region contained within 265 fuel and blanket elements. The elements were hexagonal in cross section (6.6 in. across the flats of the hexagon) and had an overall length of 11.5 ft. A unique feature of this design was the suspension of the fuel and blanket elements from a 24-in.-thick stainless steel grid plate above the core.

A standard element contained 270 fuel rods in which the fueled section was comprised of mixed plutonium-uranium oxide pellets. Pellets of depleted uranium oxide inserted in each end of the rods formed the axial blanket above and below the core. Radial blanket elements surrounding the fuel elements contained pellets of uranium or thorium oxide.

The cooling system in the demonstration plant design consisted of a three-loop main helium cooling system backed up by a three-loop core auxiliary cooling system. The helium flow was downward through the core and then upward around the outside of the core thermal shielding and out to the steam generator cavity. The flow continued downward through the steam generator, back up around the outside of the steam generator to the main helium circulators, and then back to the top of the core cavity to begin the cycle again. The coolant inlet temperature was about 300°C and the outlet temperature was 550°C.

The reactor, its cooling system, and its auxiliary systems were all contained within a prestressed concrete reactor vessel (PCRV) that had an outside diameter of 84 ft and a height of 80.5 ft. The reactor cavity within the PCRV was 20.5 ft in diameter and 41.8 ft in height. The PCRV technology was based on that developed for the PCRV of the Fort St. Vrain Reactor, which is the only HTGR in operation in the U.S. (Outside the U.S., experience with PCRVs has been extensive.)

2.2. REACTOR SHIELDING

The primary function of the reactor shielding in the GCFR design was to protect the PCRV and the core support system from radiation damage. The PCRV itself provided ample

Fig. 2.1. Nuclear Steam Supply System for the 300-MW(e) GCFR Demonstration Plant (Downflow Design).

Fig. 2.2. Vertical Section Through Reactor Cavity (Downflow Design).

biological shielding. Also, the helium coolant would not become radioactive or dissolve and carry corrosion products, which would reduce the shielding requirements for maintenance on the steam generators, circulators, etc.

As has been pointed out in Chapter 1, the passageways required for the helium coolant to flow through the core provided paths along which neutrons could stream relatively uninhibited to the outside edges of the core. And with the GCFR being a fast reactor, the fluxes of high-energy neutrons produced in the core could be expected to be considerably higher than the fluxes produced in a thermal reactor. Thus the GCFR design included a number of shield regions immediately around the core.

The first shield studied with a supporting experiment was the grid-plate shield. As shown in Fig. 2.2, in the 1974 GCFR design the grid plate supported the core from above. The grid-plate shield consisted of material added in the upper ends of the fuel elements. When the TSF "grid-plate shield design confirmation experiment" was performed in January, 1977, the GCFR design in which the grid plate was above the core still prevailed; however, after the experiment was performed, the GCFR was redesigned with the helium coolant flowing upward rather than downward and the reactor grid plate positioned below the core rather than above it. The design of the grid plate and the fuel elements remained essentially unchanged, however, and thus the TSF experiment was still pertinent to the study of the grid-plate shield, which then resided in the lower ends of the elements. A sketch of the final (upflow) fuel element is shown in Fig. 2.3, and the location of the grid-plate shield with respect to other sections of the GCFR shielding is shown in Fig. 2.4.

The grid-plate shield confirmation experiment is described in Chapter 7. While it comprised the first "shielding" experiment in the GCFR shield analysis program, it was preceded by two other experiments supporting the development of techniques for calculating the streaming of neutrons from the core. Since transport calculations for an extreme geometry such as that represented by the GCFR core had not been performed, devising techniques for determining the fluxes emerging from the core was prerequisite to analyzing the shields. These preliminary experiments are described in Chapters 5 and 6.

Fig. 2.3. GCFR Upflow Fuel Assembly.

Fig. 2.4. GCFR Shielding System Components (Upflow Design).

The second section of the reactor shield considered in the program was the radial shield protecting the PCRV. As noted in Fig. 2.4, the radial shield consists of an outer shield, which was permanent, and an inner shield, which was removable. The experiment and calculations for this shield are discussed in Chapter 8.

The radial shield study reported in Chapter 9 focused on the gross attenuation of radiation through solid shields using the same materials as are called for in the GCFR radial shield design. In the actual design the outer radial shield would consist of abutting blocks of material, and the question arose as to the extent of streaming that might occur between the blocks. The experiment described in Chapter 9 was performed to investigate this effect.

The final shield examined in the ORNL program was the "exit" shield, so named because it was installed in the helium exit end of the fuel elements to reduce the fluxes within the upper plenum. The location of this shielding in the fuel element can be seen in the upper left of Fig. 2.3 (see also Fig. 2.4). The exit shield experiment is described in Chapter 10.

Two additional experiments had been planned to investigate the adequacy of the shields surrounding the plenum regions, but the GCFR program was cancelled before they could be performed.

3. TSF EXPERIMENTAL TECHNIQUES

3.1. Detector Systems

3.1.1. Harayak Betton

3.1.2. NE-213 Spectrometer

3.1.3. Bonner Balls

3.1.4. Hydrogen Counter

3.1.5. Thermoluminescent Dosimeter (TLD)

3.2. Analysis of Experimental Errors

3.3. References

3.1. DETECTOR SYSTEMS

During the years of experimentation at the Tower Shielding Facility (TSF), several different detector systems have been successfully adapted to measurements of the intensities and energies of the neutrons and gamma-ray fields encountered in experiments such as those performed in the GCFR shielding program. These systems include a Hornyak button dosimeter, an NE-213 spectrometer, Bonner balls, a hydrogen counter, an NaI spectrometer, and thermoluminescent dosimeters (TLD). Brief descriptions of the detectors used during the GCFR experiments are given below. In general, this information has been lifted from earlier reports (refs. 1-3).

3.1.1. Hornyak Button

The Hornyak button has a response to neutrons that closely approaches that of a fast-neutron dosimeter. Described in detail elsewhere (ref. 4), the detector consists of a homogeneous mixture of fine lucite and zinc sulfide crystals formed into a button. When exposed to fast neutrons, the button acts as a scintillator whose pulses are magnified by a photomultiplier tube. For these experiments, the button was 0.635 cm in diameter and 0.159 cm thick, and it was mounted on an RCA photomultiplier tube.

The calibration procedure for the Hornyak button as used in the GCFR experiments consisted of first exposing the scintillator to a 2-R/h gamma-ray dose rate and adjusting the electronic gain to obtain a prescribed count rate for a pulse-height setting of 0.6 V (PHS = 060). The button was then exposed to a known intensity ^{252}Cf neutron source and a count rate was obtained for a PHS of 350. Calibrating at the higher PHS was necessary in order to bias against the maximum pulse height that would be reached from gamma-ray-induced pulses during the experiments. Calibrations and measurements were obtained at this PHS even though for some of the measurements the magnitude of the gamma-ray flux was considerably lower.

3.1.2. NE-213 Spectrometer

The NE-213 spectrometer (ref. 5) is used primarily to obtain the energy spectra of neutrons in the range from 800 keV to 15 MeV. In principle, it can also be applied as a

gamma-ray spectrometer; however, to date this has not been done very successfully at the TSF.

The sensitive component of the NE-213 spectrometer is a small cylinder of NE-213 liquid organic scintillator 4.72 cm in diameter and 4.71 cm long; it is oriented so that the neutrons are incident primarily on its curved surface. NE-213 is used as a compromise between sensitivity and resolution and because of its relatively isotropic response. A 14-cyode photomultiplier tube is optically coupled to the scintillator.

The pulses from the NE-213 scintillator due to neutron and gamma-ray interactions have a short decay time plus a longer component whose magnitude is greater for the proton recoil than for the electron, assuming equal initiating energies. Through integration and double differentiation, bipolar pulses are produced from the neutron-induced proton events and gamma-ray-induced electron events that cross the base line at different times, independent of amplitude. This separation in time provides the capability for electronically routing linear pulses from each of the events into separate memory locations in the computer. Thus both neutron and gamma-ray spectra are obtained. The resulting pulse-height distributions are then fed into the computer code (FERD) to obtain the energy spectra.

3.1.3. Bonner Balls

A Bonner ball detector measures an integral of the neutron energy flux weighted by the energy-dependent response function for that ball (ref. 6). The detection device in a Bonner ball consists of a 5.1-cm-diam spherical proportional counter filled with approximately 1/2 atm of BF_3 enriched to 96% in ^{10}B . The counter is enclosed in various thicknesses of polyethylene shells, the thickness determining the cutoff energies of the neutrons detected. In all cases the polyethylene shell is covered with a 30-mil thickness of cadmium to suppress the response of the detector to incident thermal neutrons. When this system is used, measurements made with the BF_3 counter bare and cadmium-covered (no polyethylene) are usually included in the series.

3.1.4. Hydrogen Counter

The hydrogen counter is used for neutron spectral measurements, usually in the energy region from about 50 keV to 1 MeV (ref. 7), although with good gamma-ray discrimination neutrons with lower energies can be detected. The counter consists of a hydrogen-filled stainless steel sphere with a tungsten wire located along one of its diameters, the sphere and wire acting as a cathode and anode respectively. A small alpha source on the midpoint of the wires at the center of the sphere is a calibration source. The counter is filled to different pressures (1, 3, and 10 atm) with hydrogen gas. Neutrons entering the counter suffer scattering collisions with hydrogen nuclei and produce recoil protons in the gas. Each proton then causes ionization, and the resulting electrons are collected on the positively charged anode. The energy of the recoil proton is determined from the amount of charge collected. Since the scattering of monoenergetic neutrons by hydrogen nuclei produces a range of recoil-proton energies from zero to the full neutron energy, a detailed analysis is required to obtain an energy spectrum from a polyenergetic neutron source. Pulse-height data from the hydrogen counter are unfolded with a computer code (SPEC-4).

3.1.5. Thermoluminescent Dosimeter (TLD)

Thermoluminescent dosimeters are used to measure the deposition of gamma-ray energy within a shield (ref. 8). The system is based on commercially available chips of ^7Li -enriched LiF and manganese-activated CaF_2 . Incident gamma rays impart energy to the orbital electrons associated with the ions making up the crystal structure, resulting in the electrons being ejected from and not immediately recaptured by the ions. Thermoluminescent materials have lattice imperfections called traps that can capture and temporarily hold the energetic electrons. If additional energy is imparted to the trapped electrons by heating, the trap will release the electrons and under certain conditions the excess kinetic energy of the electron will be emitted as light. Under carefully controlled heating conditions, a characteristic light output versus temperature "glow curve" is produced. Through a calibration procedure, the amount of intensity of the light associated with the glow curve can be related to the intensity of the radiation

field and/or the associated energy deposition by that radiation at the exposure position of the thermoluminescent material.

When the thermoluminescent material differs from the host medium, interpretation of the TLD responses requires spectral correction factors that are determined analytically. The response of the TLDs to neutrons is also an important consideration when TLDs are applied in reactor shields, which are usually subjected to characteristically high neutron-to-gamma ratios. Correction factors for the TSP TLDs are calculated from cross-section sets and other basic data.

3.2. ANALYSIS OF EXPERIMENTAL ERRORS

The errors in the measurements mainly result from a combination of the uncertainties in the reactor power determination, detector calibrations, count-rate statistics, detector positioning, configuration geometry and fabrication tolerances, and exposure of the experimental configurations to the weather. Of these, the effects due to changes in the weather are the least understood and are probably beyond simple estimation. The configurations are normally exposed to the weather and the uncertainty in the measurements is largely due to the variable amounts of moisture penetrating the space between the slabs forming the configurations. Some uncertainty is also introduced by water absorbed within the material, placed around the slabs to inhibit background contributions.

A significant source of error in the experiment usually lies in the determination of the reactor power. The power level for each measurement is determined from the outputs of two fission chambers located in the reactor shield along the reactor centerline. These detectors are calibrated daily and are periodically compared with gold-foil activation measurements at the same locations. During any one detector traverse in a given day, the variation in the reactor power as indicated by the monitors might be only a few percent. However, over a span of several months of reactor operation, the monitors may indicate that the power level has varied as much as 5% from the level indicated by the reactor control chambers. These variations may reflect the degree of reliability of the monitors or they may represent a real change in the actual power output.

Positioning the detectors with respect to the configuration is normally critical only when measurements are made very close to the experimental configuration where the radiation field varies significantly with the location. It is estimated that during a radial traverse behind a configuration the detector position might oscillate several millimeters on each side of a given line, which could affect the data by a few percent. At a distance of 30 cm behind a configuration, however, where many measurements are made, the positioning error would be negligible.

Configuration fabrication tolerances vary to some extent with the experiment, and count-rate statistics and calibration factors are unique to each detector system. For the Hornyak button, it is not uncommon to have a $\pm 5\%$ variation in the calibration factor. When combined with the errors from the reactor power determination, positioning, and counting statistics, the estimated standard error for the Hornyak button is $\pm 10\%$.

For the NE-213 spectrometer, counting statistics and unfolding errors are included in the unscrambling of the pulse-height spectra using the FERD code, with the resultant flux expressed in terms of lower and upper limits that represent a 68% confidence interval. Similar errors are expressed in the tabular data for the hydrogen counter measurements unfolded with SPEC-4. In neither case, however, is the error in the reactor power determination included in these values.

For the Bonner balls, the statistical error in the detector count rates is usually kept below 1 to 2% by maintaining an adequate count rate. Calibrations over an experimental period indicate about a 3 to 4% variation. When the error in the reactor power determination is included, the error in the Bonner ball data averages $\pm 5\%$.

In the case of TLD measurements, an attempt is made to calculate the error for each chip location in the configuration. The reliability of the results depends on many things, the most common being the determination of the weights of the chips and of their relative sensitivities to radiation, precalibration of the TLD reader and the subsequent calibration of the chips, the changes in sensitivity of the chips with use, the abuse suffered by the chips in handling, and exposure conditions. With established procedures used throughout the experiment to minimize these factors, the estimated error, excluding reactor power variations, usually falls within $\pm 5\%$.

3.3. REFERENCES

1. F. J. Muckenthaler, J. L. Hull, and J. J. Manning, "GCFR Plenum Shield Design — Shield Heterogeneity Experiment," ORNL/TM-7714 (May, 1981).
2. F. J. Muckenthaler, J. L. Hull, and J. J. Manning, "GCFR Plenum Shield Design — Exit Shield Experiment," ORNL/TM-7715 (May, 1981).
3. C. E. Clifford, F. J. Muckenthaler, F. R. Mynatt, and L. S. Abbott, "Study of Neutron Transmission Through Planar and Cylindrical Annular Slits in Iron Shields," ORNL/TM-3513 (September, 1980).
4. F. J. Muckenthaler, in "Applied Nuclear Physics Division Annual Report for Period Ending September 10, 1956," ORNL-2081, p. 199.
5. V. V. Verbinski et al., "Calibration of an Organic Scintillator for Neutron Spectrometry," ORNL-TM-2183 (1968).
6. R. L. Bramblett, R. I. Exling, and T. W. Bonner, Nucl. Instr. Meth. 9, 1 (1960); W. R. Burrus, in Neutron Phys. Div. Ann. Prog. Rep. Sept. 1, 1962, " ORNL-3360, p. 296; S. K. Mehta, in "Neutron Phys. Div. Ann. Prog. Rep. Aug. 1, 1964," ORNL-3714, p. 80.
7. An ORNL version of a Benjamin-type proportional counter; see P. W. Benjamin, C. P. Kamshall, and J. Redfearn, Nucl. Instr. Meth. 59, 77 (1968); see also, R. G. Graf, "The Implementation and Calibration of a Benjamin-Type Neutron Spectrometer for Energies Below 1.5 MeV," University of Tennessee Doctoral Dissertation, 1969.
8. C. E. Clifford, F. J. Muckenthaler, Wu Yoon, P. N. Stevens et al., "Thermoluminescent Dosimeter Measurements of Radiation Heating in Sodium and Stainless Steel Shields," ORNL-5329 (to be published).

4.1. THE MAJOR TRANSPORT CODES

As noted in Chapter 1, the ORNL shielding analysis methods are based on two major computer codes: the discrete ordinates transport code DOT (ref. 1) and the Monte Carlo transport code MORSE (ref. 2). These codes use two different approaches for solving the Boltzmann transport equation, the selection of one over the other usually being determined by the geometry of the shield region to be calculated. For reasons that will be apparent from the following discussion, the DOT code was used for most of the analyses of the TSF GCFR experiments.

In addition to DOT and MORSE, the ANISN discrete ordinates code (ref. 3) was used frequently in preliminary analyses because of the cost savings relative to DOT or MORSE. However, geometry restrictions with ANISN limited its usefulness in most of the final analyses.

4.1.1. The Discrete Ordinates Codes DOT and ANISN

The discrete ordinates method is a numerical technique that simultaneously yields highly differential results for the entire system being calculated. In the calculation, the system is divided into finite space cells and the calculation maintains a balance between particle gains and losses for each space cell. With the origin (source) of the particles specified and the boundary conditions set, the calculation begins with the highest energy group and sweeps through all lower energy groups and all space cells in the system for all particle directions. Several sweeps (iterations) are required before the results converge to give the number of particles from the source's high-energy group that have entered each space cell, together with their resulting energies and directions. The procedure is repeated for each energy group in decreasing order. The results are presented for space points located at the centers of the space cells and are taken to be the means of the number of particles within the given energy group at opposing faces of the cells.

The major shortcoming of the DOT discrete ordinates code (ref. 1) is that it is limited to two-dimensional geometries. Thus, if a TSF experimental configuration (or a sector-shield system) is to be analyzed by the DOT code, it must be describable in two

dimensions. In the TSF experiments, symmetry is attempted by making the configuration as symmetrical as is possible around the axis of the TSR-II beam collimator. The configuration usually resembles a right circular cylinder with one end of the cylinder adjacent to the outer edge of the beam collimator (see Fig. 1.1). In practice, of course, the configuration is seldom an exact cylinder, and at times would even sacrifice its mission if were symmetrical about the beam axis. But for calculational purposes, the TSR configurations are mocked up in symmetrically cylindrical geometry, and the resulting calculated fluxes are symmetrical around the axis of the cylinder.

Another problem with the DOT code is that it is difficult for it to handle multiregioned geometries in which abutting regions vary greatly in density. As will be discussed in Chapters 5 and 6, the first two experiments performed in the GCFR shielding program were designed to aid in the development of calculational techniques that could be applied in DOT to overcome this problem.

Against these limitations, DOT has numerous advantages, not the least of which is its ability to produce large volumes of detailed flux data for essentially all points within a system in a relatively short time and at a relatively economical cost.

The ANISN code (ref. 3) has the same limitations as DOT plus it is restricted to one-dimensional geometries. Even so, the fast setup and running time for ANISN makes the code a powerful tool for preliminary analyses, parametric studies, and diagnostic assessments.

4.1.2. The Monte Carlo Code MORSE

The MORSE code (ref. 2) utilizes the Monte Carlo calculational method, which is a numerical procedure based on statistical theory and is the only method by which the Boltzmann transport equation can be solved rigorously in three dimensions. The Monte Carlo method can be thought of as following individual particles as they execute "random walks" through a system, the steps of the walk consisting of a series of flights between interactions. The positions at which the interactions occur and the results of the interactions are determined from a range of possibilities by sets of random numbers -- hence the name Monte Carlo.

The major difficulty in applying the Monte Carlo method to shielding calculations is that (1) an unduly long time is required to generate enough particle histories to arrive at reliable flux values, and (2) calculations can be practically performed for only a few locations within a system. As a result, MORSE calculations are not routinely carried out. Rather they are performed primarily for difficult geometries that cannot be reduced to two dimensions or to obtain check points for comparison with DOT calculations. On the other hand, the development of the MORSE code over the years has resulted in techniques that have greatly reduced the time requirements and costs associated with Monte Carlo shielding calculations.

4.2. AUXILIARY CODES

In addition to DOT, ANISN, and MORSE, several auxiliary codes were used in the discrete ordinates analyses of the GCFR experiments. No attempt will be made here to describe these codes; however, to complete the documentation of the analysis methods, they are listed and referenced as follows:

SPACETRAN III (ref. 4) - transports DOT fluxes emerging from the side and end of a cylindrical configuration to detector points beyond the configuration using a DOT-III flux file as input.

DISKTRAN (ref. 5) - performs a function similar to SPACETRAN for disk boundary (cylinder ends) sources using the DOT top/bottom boundary flux arrays as input.

FALSTF (ref. 6) - performs a last-event estimation from discrete source points within a system to points outside the system to determine responses based on the collided flux. Provision is made for estimating the contribution to the response from a boundary source at the end of a cylinder.

GRTUNCL (ref. 7) - generates first-collision source.

SWANLAKE (ref. 8) - calculates sensitivity of responses to specific interactions, shield composition, cross-section data, etc.

VIP (ref. 9) - uses two-dimensional forward and adjoint flux moments to produce one-dimensional $\phi^*\phi$ matrices required by SWANLAKE.

FORSS (ref. 10) - a system of codes to facilitate sensitivity calculations.

FACT (ref. 11) - converts the angular quadrature and spatial meshes used in one shield region to the quadrature and mesh required for the next region. Also computes a detector response by folding adjoint boundary fluxes with the forward boundary source incident on one of the cylindrical surfaces.

ANPX (ref. 12) - a system of codes to facilitate cross-section processing.

DOGS (ref. 13) - a system of codes to provide graphical output of calculated data.

4.3. COMPARING CALCULATED AND MEASURED RESULTS

As has been stated before, the calculational methods and techniques developed for the analysis of the shields of large reactor systems cannot be applied with confidence unless they have been confirmed experimentally, which means, of course, that the measured results and the results calculated in the analysis of the experiment must be presented in the same terms.

DOT and MORSE calculations yield "point" fluxes (that is, fluxes at specific points in a medium). The measurements are obtained as count rates, pulse heights, etc. due to particle interactions with the sensitive volume of the detector. In some cases calibration factors, which might include the efficiency of the detection system, are applied to the measured data in order to obtain results in the same units as quantities normally calculated. In other cases, the calculated data are converted to the same units as the measured responses.

In the case of the Bonner balls (see Section 3.1.3), the calculated fluxes are converted to Bonner ball count rates using Bonner ball response functions (ref. 14) that are folded with the calculated energy fluxes. Similarly, response functions are available for the thermoluminescent detectors (TLDs) (ref. 15) and the Hornyak button.

In the case of the spectrometers (the NE-213 and hydrogen counter), the pulse-height data obtained with the spectrometers are "unfolded" via the use of computer codes to obtain spectra that can be compared directly with the calculated spectra. The calculated

spectra are usually smoothed with a Gaussian resolution function which corresponds to the detector resolution in the spectrometer system.

As will be noted in the following chapters, the bulk of the comparisons are for Bonner ball count rates. This is because as the configurations increase in thickness, only the Bonner ball detectors are sensitive enough for measurements to be made within a reasonable length of time.

4.4. REFERENCES

1. F. R. Mynatt, F. J. Mackenthaler, and P. N. Stevens, "Development of Two-Dimensional Discrete Ordinates Transport Theory for Radiation Shielding," Computing Technology Center, UCC-ND, CRC-INF-952 (1969); W. A. Rhodes and F. R. Mynatt, "The DOT-III Two-Dimensional Discrete Ordinates Transport Code," ORNL-TM-4280 (Sept., 1973); W. A. Rhodes, D. B. Simpson, R. L. Childs, and W. W. Engle, Jr., "The DOT-IV Two-Dimensional Discrete Ordinates Transport Code with Space-Dependent Mesh and Quadrature," ORNL/TM-6529 (January 1979); W. A. Rhodes and R. L. Childs, "An Updated Version of the DOT-4 One- and Two-Dimensional Neutron/Photon Transport Code," ORNL-5851 (In review).
2. E. A. Straker, P. N. Stevens, D. C. Irving, and V. R. Cain, "The MORSE Code -- A Multigroup Neutron and Gamma-Ray Monte Carlo Transport Code," ORNL-4585 (1970); M. B. Emmett, "The MORSE Monte Carlo Radiation Transport Code System," ORNL-4972 (February, 1975).
3. W. W. Engle, Jr., "User's Manual for ANISN, A One-Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering," Computing Technology Center, Union Carbide Nuclear Division, K-1693 (1967); available from ORNL Radiation Shielding Information Center's computer code collection as CCC/254 (IBM-360 version) or CCC/255 (CDC version).
4. S. W. Cramer and M. Solomito, "SPACETRAN: A Code to Calculate Dose at Detectors at Various Distances from the Surface of a Cylinder," ORNL-TM-2592 (June 17, 1969); theory of SPACETRAN discussed by Mynatt et al. in CTC-INF-952 (see ref. 1 above); SPACETRAN III is an updated version of the original code.

5. DISKTRAN was written by C. O. Slater; see C. O. Slater, S. N. Cramer, and D. T. Ingersoll, "Analysis of the ORNL/TSF GCFR Grid-Plate Shield Design Confirmation Experiment," ORNL-5551 (August, 1979).
6. The FALSTF code, developed by R. L. Childs of ORNL, does a last-event estimation from discrete source points in the system to external detectors to obtain responses based on the collided flux. The source is the total scattering source which can be generated from the DOT flux moments and the cross-section expansion.
7. GRTUNCL, developed by R. L. Childs, has not been documented separately; however, it is available in the DOT code package obtainable from the ORNL Radiation Shielding Information Center.
8. D. E. Bartine, F. R. Mynatt, and E. M. Obloy, "SWANLAKE, A Computer Code Utilizing ANISN Radiation Transport Calculations for Cross Section Sensitivity Analysis," ORNL-TM-3809 (May, 1973).
9. R. L. Childs, D. E. Bartine, and W. W. Engle, Jr., "Perturbation Theory and Sensitivity Analysis for Two-Dimensional Shielding Calculations," Trans. Am. Nucl. Soc. 21, p. 543 (1975).
10. C. R. Weisbin, E. M. Obloy, F. R. Mynatt, and G. F. Flanagan, "The FORSS Sensitivity Analysis Code System," Trans. Am. Nucl. Soc. 22, 792 (1975); C. R. Weisbin et al., "Application of FORSS Sensitivity and Uncertainty Methodology to Fast Reactor Benchmark Analysis," ORNL/TM-5563 (ENDF-236) (December, 1976).
11. The FACT code, developed by M. L. Gritzner, W. W. Engle, Jr., W. A. Rhodes, D. R. Vondy, and L. R. Williams, is undocumented; the current version is maintained by W. W. Engle, Jr., ORNL.
12. N. M. Greene, W. E. Ford, III, J. L. Lucius, J. E. White, L. M. Petrie, and R. Q. Wright, "AMPX: A Modular Code System for Generating Coupled Multigroup Neutron-Gamma Libraries from ENDF/B," ORNL/TM-3706 (March, 1976).
13. D. T. Ingersoll and C. O. Slater, "DOGS - A Collection of Graphics for Support of Discrete Ordinates Codes," ORNL/TM-7188 (March, 1980).

14. R. E. Maerker et al., "Calibration of the Bonner Ball Neutron Detectors Used at the Tower Shielding Facility," ORNL/TM-3465 (June 30, 1971); C. E. Burgart and M. B. Emmett, "Monte Carlo Calculations of the Response Functions of Bonner Ball Neutron Detectors," ORNL/TM-3739 (April, 1972).
15. R. E. Maerker, "Analysis of TSF Experiments on Radiation Heating in a Stainless Steel-Sodium ORBR Radial Shield Mockup Using a 32-Inch Diameter Collimated Beam Source," ORNL/TM-5992 (October, 1977).

5. STUDIES OF NEUTRON STREAMING IN GCFR CORE MOCKUPS: SINGLE-CELL EXPERIMENT

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5.1. Introduction

5.2. The Experiment

5.3. Preanalysis of the Experiment

5.3.1. Selection of Angular Quadrature Set

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5.4. Postanalysis of the Experiment; Comparison of Calculated and Measured Results

5.4.1. Definition of Source

5.4.2. Transport Calculations for Spectrum Modifier

5.4.3. Transport Calculations for Cell Assemblies

5.5. References

5.1. INTRODUCTION

Since the ORNL Tower Shielding Facility experiments for the GCFR program were to test the effectiveness of specific shield designs for the GCFR, it was first necessary to ensure that pertinent experiments could be performed and, moreover, that they could be fully understood — that is, to ensure that the experiments would actually represent the design points in question and also that they could be analyzed.

It was immediately recognized that designing and analyzing shield experiments for the GCFR presented unique problems because of the design of the GCFR core itself, the primary problem being that the passageways between fuel rods that allowed the helium coolant to flow down through the core (or up through the core) would also allow neutrons to stream directly from the innermost regions of the core. Thus any structure above or below the core and its axial blanket would be bombarded by intense pockets of high-energy neutrons corresponding to the coolant passageways, and if any mockup experiments of shields to protect such structure were to be meaningful, some technique for including the streaming sources would have to be devised.

Including the streaming sources in experiments would not alone suffice, however. It was also necessary to determine the points of origin and energies of the emerging neutrons and to identify their streaming paths, which meant that the experimental streaming source would have to be calculable. But at the time this program was begun, no technique was available for calculating the extreme and detailed experimental geometry that would be required, nor, of course, for calculating the geometry of the GCFR proper. Describing such a complicated system in its actual three-dimensional geometry for calculations with the Monte Carlo method would have overwhelmed the computer and thus was not feasible. The problem could have been reduced to manageable proportions by homogenizing the materials of the core in a two-dimensional cylindrical geometry for calculations with the discrete ordinates method but that would have eliminated the streaming paths as identifiable regions. The question arose, then, as to whether the streaming paths in the core could be adequately represented with a heterogeneous two-dimensional model consisting of alternating cylindrical annuli of high-density and low-density materials and whether the neutrons could be followed through the low-density regions.

As a first step in investigating whether streaming could be calculated with heterogeneous cylindrical models, several simple streaming calculations were performed with the DOT discrete ordinates code for comparable homogeneous and heterogeneous cylinders. The first heterogeneous model consisted of an iron rod (7.239 mm in diameter by 762 mm long) surrounded by an air-streaming gap, the thickness of the gap (2.5654 mm) simulating the triangular pitch between fuel rods within a GCFR fuel element. Thus the configuration roughly represented a single cell in a fuel element. The corresponding homogeneous model consisted of the same materials homogenized over the entire region of the cylinder. The calculations for these two cylinders, plus calculations for other pairs of Fe-air and UO_2 -air cylinders, some in which the angular distribution of the source was varied, indicated that streaming did indeed impact the results for the heterogeneous model. The calculations did not prove, however, that the streaming was accurately calculated with the model. Such proof could be obtained only by agreement of calculated streaming results and measured streaming results for the same cylindrical configuration. As a result, the GCFR "shielding" program at ORNL began not with shielding mockup experiments but rather with a single-cell core-streaming experiment and a corresponding analysis effort. The experiment, performed at the Tower Shielding Facility in mid-1975, is described in detail in ref. 1 and summarized below.

5.2. THE EXPERIMENT

The design of the single-cell experiment began with a modification of the Tower Shielding Reactor II (TSR-II) radiation source to more nearly represent the neutron energy spectrum of the GCFR. Earlier calculations had indicated that filtering the TSR-II beam through 102.64 mm of iron and 25.4 mm of boral would give the desired spectrum. Therefore, a spectrum modifier of this design placed in the beam adjacent to the reactor comprised the first component of the experimental configuration.

Each mockup behind the spectrum modifier consisted of a 11.48-mm-diam "fuel pin" centered in a 15.748-mm-diam by 1.374-m-long aluminum tube, the fuel pin surrounded by a 2.134-mm-thick air annulus within the tube and the entire cell (tube) surrounded by water (see Fig. 5.1). In some cases the "fuel pin" consisted of a UO_2 rod and in others it consisted of an Fe rod. The UO_2 pin, shown in Fig. 5.2, had an overall length of 1.3716 m

-COLLIMATOR (TWO SIZES)

ORNL - DWG 76-8213R

DIMENSIONS IN CENTIMETERS

Fig. 5.2. Schematic of UO_2 Pin.

and contained a 1.219-m-long "active" section comprised of 1.3% ^{235}U -enriched fuel pellets. The iron pin also had an overall length of 1.3716 m, but it had a 76.2-mm long by 8.204-mm-diam hole drilled in the source end.

Two different air-filled source collimators were used in the beam preceding the fuel pins: a 14.92-cm-diam collimator, which was larger than the diameter of the cell; and a 4.928-cm-diam collimator, which was smaller than the diameter of the fuel pin. Both collimators were 26.194 cm long. (Diameters specified in Fig. 5.1 include wall thicknesses - 0.16 cm for outer wall and 0.07 cm for inner wall.)

Measurements of neutrons transmitted through the experimental configurations were made with Bonner ball detectors both behind the spectrum modifier and behind the individual cell configurations. In addition to the cell assemblies described above, measurements were made for a configuration in which the cell was filled with air only and for another in which the entire assembly was flooded with water.

With the various cell configurations it was anticipated that several differences would be observed. Measurements for the voided-cell configuration preceded by the large collimator would represent the upper bound of the experiment, while measurements for the water-flooded configuration would represent the lower bound (i.e., the background). Also, a comparison of these measurements would isolate the neutrons transported through the cell from those transported through the water tank.

A comparison of the measurements for the Fe pin configuration with the background case would indicate the extent to which the air annulus enhanced neutron streaming through the cell, while a comparison of the measurements for the Fe pin configuration with those for the UO_2 pin configuration would indicate how the production of neutrons within the pin itself (from ^{235}U fissions) would increase the streaming. A comparison of the large- and small-diameter collimator measurements would indicate the relative importance of neutrons entering the cell through the pin and those entering the cell through the annulus. And finally, a comparison of the measurements on the cell axis with those made off the axis would indicate the relative magnitude of the "uncollided flux," that is, the flux of neutrons travelling through the air annulus directly to the detector without undergoing a change in direction through scattering.

As it turned out, the measurements for configurations in which the small (4.928-mm diam) collimator preceded the cells containing fuel pins were essentially the same as those for the water-filled cell (background measurement) and thus were not meaningful. Also, no significant differences were observed between the measurements for the Iron pin and those for the UO_2 pin and therefore contributions from fissions in the pin could not be determined. But before these results were known, a preanalysis of all the experimental configurations had been undertaken both to verify that the experiment had been properly designed and to further develop the analysis technique.

5.3. PREANALYSIS OF THE EXPERIMENT

5.3.1. Selection of Angular Quadrature Set

Because the long and thin geometry of the experimental cells would severely test the angular quadratures used in the DOT calculations, the first step in the preanalysis was to select an angular quadrature set. Two detailed quadrature sets were available: a 147-direction negative-eta-biased set and a 150-direction set which included all the directions of the previous set plus an additional eta level containing three more directions. In order to check whether either of these sets would be appropriate for the experimental analyses, they were both used for calculations of an additional "single-cell" configuration. The case calculated was that of a totally absorbing rod surrounded by a void, the rod having a volumetric shell source on one end. Point detectors were located along the length of the rod 0.2 cm away from the surface, with all locations having a "line-of-sight" to portions of the source. DOT results were obtained with the two quadrature sets and were compared with results obtained with an analytic calculation and a Monte Carlo calculation that were nearly identical and believed to be correct. Both DOT calculations tended to overpredict for locations close to the source, giving approximately the same results. At distant locations both DOT calculations tended to underpredict, with the 150-direction quadrature set yielding somewhat better results. It was concluded that precise calculations could not be obtained with DOT for the extreme case represented by the hypothetical cell. The experimental cell would not be nearly so extreme, however, and it was felt that the 150-direction set, which had 115 angles biased in the negative eta direction would be adequate for the experimental analysis.

5.3.2. Definition of Source

When the preanalysis was begun it was known that the experiment would utilize a new TSR-II beam collimator whose cross-sectional area was about five times the area of a smaller collimator beam that had been in use for a number of years; however, an analysis of the energy and angular distribution of the new beam was not available. For the preanalysis it was assumed that the energy spectra of the large and small beams were identical (refs. 2 and 3) and that the intensity of the large beam was 200 times that of the small beam.

5.3.3. Transport Calculations for Spectrum Modifier

A 50-group unit source having the TSR-II spectrum was collapsed to a 10-group structure and placed on the reactor side of the spectrum modifier. A one-dimensional ANISN transport calculation was then performed to obtain the emergent angular fluxes, neglecting scattering back into the spectrum modifier. The resulting angular fluxes were converted to two-dimensional angular fluxes in the 150-direction quadrature set and were then used as the boundary source for DOT calculations of neutron transport through the experimental configurations.

5.3.4. Transport Calculations for Cell Assemblies

The DOT calculations were performed for all the experimental configurations described above (assuming a void for air) and were checked with MORSE Monte Carlo calculations in some cases. The DOT calculations provided neutron leakages from the experimental assemblies, and additional calculations with the FALSTF code transported the fluxes to points beyond the assembly.

The preanalysis results were given both as neutron fluxes in three energy groups (>122.8 keV; 2.382 eV to 122.8 keV; and <2.382 eV) and as Bonner ball count rates. Neutron (flux) leakages from all the configurations were given as total leakages from the system, as leakages from the cell (subdivided into the annulus and pin), and as leakages from the surrounding tank. Neutron fluxes were also calculated for various locations beyond the configuration specified by the distance r and angle theta (θ) shown in Fig. 5.1. The fluxes were converted to Bonner ball count rates for the locations beyond the configurations only.

The total neutron leakage from the configuration with the air-filled (voided) cell following the large collimator was calculated to be 9.184×10^{-5} (neutrons s^{-1})/(incident neutron $cm^{-2} s^{-1}$), with the leakage through the cell comprising about one-half of the total leakage (see Table 5.1). The total leakage from the voided-cell configuration was higher by a factor of 13 than the leakage from the water-flooded configuration, the increase being largely attributable to the streaming of the higher energy neutrons through the cell: the leakage of >122.8 -keV neutrons was a factor of 153 higher through the cell than through the surrounding tank and the leakage of intermediate-energy neutrons was a factor of about 17 higher.

Table 5.1. DOT-Calculated Relative Neutron Leakages from Single-Cell Assemblies (Preliminary)

System	Ratio:		
	System Leakage	Ratio: Cell Leakage/Tank Leakage	
	Background	Total	>122.8 keV
Water-filled (background)	1		
Void, large collimator	12.9	0.526	153
Void, small collimator	3.1	0.481	8.99
Iron pin, large collimator	1.6	0.075	9.63
Iron pin, small collimator	1.2	0.021	4.74
UO ₂ pin, large collimator	3.4	0.036	8.14
UO ₂ pin, small collimator	*	0.147	1.83

*Calculated leakage below background due to lack of convergence of thermal group.

When the small collimator preceded the voided cell, the total leakage from the system was only a factor of 3 higher than that from the water-flooded system, and the cell-to-tank leakage ratio was about 9 for the >122.8 -keV neutrons and 2.4 for the intermediate-energy neutrons.

When the Iron pin was introduced into the configuration with the large collimator, the total leakage from the system was higher by a factor of 1.6 than that from the water-flooded configuration, the leakage through the cell comprising less than 8% of the

total leakage. In this case the leakage of >122.8 -keV neutrons was a factor of 9.6 higher from the cell than from the surrounding tank and the leakage of intermediate-energy neutrons was a factor of 1.8 higher from the cell.

When the small collimator preceded the cell with the iron pin, the total leakages from the system and from the water tank were not greatly reduced from those for the iron-pin large-collimator system, but the leakage from the cell was reduced by a factor of about 3.6, largely because of a reduction in the leakage of >122.8 -keV neutrons from the cell (cell-to-tank ratio = 4.7) and in spite of an increase in the leakage of intermediate-energy neutrons (cell-to-tank ratio = 3).

When the UO_2 pin was substituted for the iron pin in the large-collimator configuration, the total leakage from the system doubled, nearly all of the increase being reflected in an increase in low-energy neutrons (<2.382 -eV neutrons) emerging from the tank. The leakage from the pin also more than doubled, with increases in all energy groups. However, the leakage from the annulus remained dominant in the overall cell leakage and was essentially the same as that for the iron-pin case. The overall result was that the cell-to-tank leakage ratio for the UO_2 case was less than one-half that for the iron case.

Since the UO_2 total cross section is larger than the iron total cross section, the increase in leakage with the UO_2 pin would appear to be attributable to fissions in the UO_2 pin with the subsequent thermalization of the fission neutrons in the tank. The observation that fissions are contributing significantly is supported by the preanalysis results for the UO_2 pin in the small-collimator configuration. Although not detailed in Table I, the leakages from the tank and annulus in the small-collimator UO_2 -pin configuration are greatly reduced from the leakages for the large-collimator UO_2 -pin configuration. But the leakage from the pin itself is significantly increased, all of the increase being in the <2.382 -eV energy group. On the other hand, the total leakage calculated for the system is below background owing to a lack of convergence of the low-energy group.

Monte Carlo (MORSE) calculations of the leakage from the annulus were performed for the large-collimator voided-cell case and the small-collimator iron-pin case. The

>2.38-eV neutron leakage calculated with MORSE for the voided-cell case was in fairly good agreement with the corresponding DOT results; however, all other comparisons showed the MORSE fluxes to be significantly lower than the DOT fluxes. Some of the discrepancy could possibly be due to an insufficient number of events occurring deep within the tank and in the vicinity of the cell in the MORSE calculations.

DOT calculations of the fluxes beyond the cell indicated that the total fluxes for the small-collimator pin configurations were about equal to those for the water-flooded system (background measurements), as, of course, was discovered to be the case when the measurements were actually made (see above).

Finally, the DOT fluxes for positions beyond the configurations were converted to Bonner ball count rates for subsequent comparisons with the measurements and with the postanalysis.

5.4. POSTANALYSIS OF THE EXPERIMENT; COMPARISON OF CALCULATED AND MEASURED RESULTS

Since the cells with the small collimator could not be properly measured and since the Iron pin and UO_2 pin cases did not differ much in the preanalysis, the postanalysis was limited to the large-collimator configuration including the voided cell and the large-collimator configuration including the Iron pin. The results from these calculations are compared with the corresponding experimental results in the following discussion.

5.4.1. Definition of Source

By the time the postanalysis was performed, the energy spectrum and angular distribution of the neutron fluxes in the TSR-II large beam had been calculated and reported in a 51-energy-group, one-dimensional spherical S_{16} quadrature (non-zero source in eight positive directions) structure (ref. 3). Because the angular fluxes were defined with respect to the hemispherical surface of the TSR-II containment tank, they had to be transformed to a plane surface for use as a boundary disk source in the DOT calculations. This was done by a technique devised by Maerker, and the resulting plane source was assumed to be located in the plane coincident with the inside surface of the concrete collimator, i.e., on the reactor side of the concrete shown in Fig. 5.3.

CONTINUED.

Abstract – This paper presents a new approach to the design of a control system for a two-link manipulator. The proposed method is based on the use of a neural network to approximate the inverse dynamics of the manipulator. The results show that the proposed method can achieve high tracking accuracy and robustness to parameter variations.

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validity of the calculational method but also the validity of the energy spectrum calculated in the analyses for the neutrons emerging from the spectrum modifier. As shown in Fig. 5.4, the spectrum modifier depletes the low-energy end of the TSR-II spectrum and shifts the peak flux from about 2 MeV to about 0.7 MeV. This spectrum is harder than spectra calculated for three positions within the GCFR demonstration plant (ref. 4); therefore on a unit source basis, calculations using this spectrum should result in greater neutron penetration than calculations using the other three spectra.

5.4.3. Transport Calculations for Cell Assemblies

The calculation of the transport of neutrons from the spectrum modifier through the experimental configurations was carried out as in the preanalysis except that the 51-group neutron energy structure was used rather than the 10-group structure. The results from this part of the calculation were, as before, given as the neutron fluxes emerging from the experimental assembly.

Fig. 5.4. Comparison of the Calculated Spectra of Neutrons Leaking from Surface of the TSR-II Reactor Vessel and from the Spectrum Modifier.

Attempts to use a modified SPACETRAN code for calculating the neutron fluxes at the detector positions beyond the experimental assemblies were unsuccessful; therefore, a code was written to calculate the uncollided fluxes by doing a point source $1/r^2$ kernel integration from the source at the spectrum modifier, multiplying by an attenuation factor, and integrating the product of the computed fluxes and the adjoint angular currents over the Bonner ball hemispheres, using a high-order quadrature. The resulting uncollided fluxes were then added to the collided fluxes.

The collided fluxes were calculated with FALSTF for several distances behind the tank and at several radial positions at each of these distances. For Bonner balls on the cell axis, the fluxes were folded with the adjoint angular distribution and integrated by the trapezoidal rule. (The assumed adjoint flux had a cosine shape; the angles were determined by the triangle formed by a point on the spectrum modifier centered on the cell, a point at the center of the Bonner ball, and a point on the surface of the ball.) Interpolation was used so that the number of positions at which the responses were computed could be limited.

Calculations of this type are very time consuming when the detector is off the cell axis since the source azimuthal integration must be performed in addition to the radial integration. Consequently the analysis was confined to on-axis detectors, except for a few r , which no point on the ball fell within the radius of the hole (that is, the center of ball was at $r > 4.763$ cm for the 3-in. detector, at $r > 8.573$ cm for the 6-in. detector, and at $r > 11.551$ cm for the 10-in. detector).

Comparisons of on-axis measured and calculated Bonner ball count rates are shown in Table 5.2. For the voided-cell configuration, the calculated responses were 4 to 40% lower than the measured responses. In the case of the iron-pin configuration the calculated responses were in some cases too low (by as much as 13%) and in other cases too high (by as much as 61%).

The calculated responses for the voided-cell case approached the measured responses only for on-axis locations at large distances (> 300 cm) beyond the experimental assembly and then only for the 3-in. and 6-in. balls. Since at these locations the responses are largely due to uncollided fluxes, the near-agreement would seem to indicate (1) that the source on the face of the spectrum modifier was fairly accurately defined and (2) that

Table 5.2. Comparison of Calculated and Measured On-Axis Bonner Ball Count Rates Beyond Single-Cell Assemblies (Postanalysis)

Distance Behind Assembly (cm)	Ratio: Calculated/Measured					
	Voided Cell			Cell with Fe Pin		
	3-in.	6-in.	10-in.	3-in.	6-in.	10-in.
12.7	0.706	0.792	0.609	1.041	1.368	1.182
30.48	0.775	0.863	0.739	0.982	1.147	0.977
152.4	0.836	0.843	0.703	0.875	1.013	0.875
243.84	0.891	0.836	0.719*	1.260	1.064	0.929
304.8	0.963	0.897	0.700	1.429	1.112	1.005
438.68	0.974	0.961	0.721	1.608	1.259	1.025

*Measurement at 242.89 cm.

the discrepancies at closer distances were due to difficulties in calculating the collided fluxes. At all locations behind the voided-cell configuration the calculations underpredicted the responses for the 10-in. ball, which is most sensitive to high-energy neutrons. This may indicate a failure of a P_3 moments expansion of the flux to give accurate forwardly peaked scattering sources for the high energy groups. On the other hand, the larger diameter ball may have needed a finer quadrature than the other balls to obtain a precise integral of the forward flux and adjoint current over the ball surface.

In the case of the iron-pin configuration, the closest agreement between the calculated and measured responses occurred on the axis at distances between 150 and 250 cm for the 3-in. and 6-in. balls and at distances between 300 and 400 cm for the 10-in. ball. Fluctuations between overpredictions and underpredictions probably were due to overpredictions of the collided flux and underpredictions of the uncollided flux. With the cell not open, the discrete surface integration can lead either to overprediction or underprediction when the contribution from a particular source point to a point on the surface of the detector is assigned too great a weight or too small a weight. One prediction for an off-axis location of the 10-in. detector (at a distance of 30.48 cm) was actually 58% higher than the measured count rate.

Some possible difficulties in the measurements should also be noted. Radial traverses with the 3-in. detector 30.48 cm behind both configurations and with the 6-in. detector 243.84 cm behind the iron-pin configuration yielded measurements that were highly asymmetric for points equidistant from the axis. And in some cases the 6-in. ball gave measurements that differed by large factors at the same locations. In the experimental count rates reported, all these measured responses were given equal weight and averaged. Such a strong azimuthal variation in the six-in measurements may indicate that the experimental system was not really two-dimensional, or it may indicate imprecise locations of the Bonner balls in a sharply varying radiation field. Certainly a detector which more nearly resembles a point detector would have been preferable.

In conclusion, it would have to be admitted that the goal of designing a single-cell experiment that would represent a typical GCFR fuel-pin cell was not achieved. This was mainly because the small diameter of the pin size required a source and collimator so small that the system appeared to be homogeneous. Also, the lack of an experimental detector that could measure point fluxes was a severe disadvantage for this experiment, since it was inappropriate to assume that the fluxes emerging from the small-diameter cell would be uniform over the relatively large-diameter Bonner balls. On the other hand, the calculations and measurements were not in such radical disagreement as to discredit the 150-direction quadrature set which was to be used for the analysis of the later GCFR experiments.

5.5. REFERENCES

1. C. O. Slater and J. R. Knight, "Analysis of the TSF GCFR Single-Cell Neutron Streaming Experiment," ORNL/CCX-80-16 (July 1980).
2. R. E. Meeker and F. J. Muckenthaler, "The Absolute Neutron Spectrum Emerging Through a 15-1/4-In. Collimator from the TSR-II Reactor at the Tower Shielding Facility," ORNL-TM-4010 (December 1972).
3. R. E. Meeker and F. J. Muckenthaler, "The Absolute Neutron Spectrum Emerging Through the Large Beam Collimator from the TSR-II Reactor at the Tower Shielding Facility," ORNL-TM-5183 (January 1976).
4. General Atomic Co. Project Staff, "300-MW(e) Gas-Cooled Fast Breeder Reactor Demonstration Plant," General Atomic Company Report GA-A13045 (July 1974).

6.1. INTRODUCTION

In order to further investigate whether a capability could be developed for calculating neutron streaming in the helium coolant passageways between the GCFR fuel and blanket elements, a second experiment was designed in which an array of UO_2 fuel rods was used. While the experimental fuel pin lattice was not typical of the proposed 300-MW(e) GCFR core and blanket, fission neutrons would be produced throughout the array, making it sufficiently similar to the GCFR core that measurements of neutron streaming would give some indication of streaming effects in the reactor.

The fuel-lattice experiment was performed in 1975 as a follow-on of the single-cell experiment. The experiment and its analysis are described in refs. 1 - 3 and summarized below.

6.2. THE EXPERIMENT

The experimental arrangement for the fuel-lattice experiment was similar to that for the single-cell experiment except that the small-diameter tube containing the single-cell configuration was replaced with a 0.459-m-ID by 1.42-m-long steel pipe containing the fuel-lattice configuration. The fuel lattice consisted of 894 of the enriched UO_2 pins (see Fig. 5.2) arranged on a triangular pitch of 1.425 cm. The pins were supported within the steel pipe by three 2.77-mm-thick steel grid plates placed approximately at the center and about 7.6 cm from each end of the pipe. The full configuration is shown in Fig. 6.1.

Fig. 6.1. Configuration for the GCFR Fuel-Lattice Neutron-Streaming Experiment.

Since the configuration through the spectrum modifier was the same as in the single-cell experiment, the measurements behind the spectrum modifier were not repeated

(see Section 5.1). The measurements beyond the fuel lattice were made with the 3-, 6-, and 10-in.-diam Bonner balls and also with two spectrometers: the TSF NE-213 spectrometer, covering energies from approximately 1 to 15 MeV; and the TSF hydrogen counter, covering energies from approximately 0.02 to 1.5 MeV. All the detectors were located on an arc whose radius was 9.144 m from the center of the exit face of the lattice.

The unfolded spectra for $E > 0.015$ MeV based on the two spectrometer measurements are shown in Fig. 6.2. It should be noted that the data show a sharp drop over small angular traverses from the center line. The total flux at the 0-deg position is about 84 times that at the 7.5-deg position, while the total flux at the 2.5-deg location is about a factor of 17 larger than that at 7.5 deg.

The Bonner ball measurements are discussed in Section 6.2 below.

Fig. 6.2. Energy Spectra of Neutrons ($E > 0.015$ MeV) Measured Along an Arc 9.144 m Beyond Fuel Lattice.

6.2. ANALYSIS OF THE EXPERIMENT; COMPARISON OF CALCULATED AND MEASURED RESULTS

6.3.1. Definition of Source

The "source" for the analysis of the fuel-lattice experiment consisted of the angular fluxes calculated to be emerging from the spectrum modifier in the earlier postanalysis of the single-cell experiment (see Section 5.3.2). The scalar flux tape on which the fluxes were saved provided an internal boundary source (nonzero for negative eta only).

6.3.2. DOT Transport Calculations for Fuel Lattice

The fuel-lattice configuration was calculated first with the two-dimensional discrete ordinates code DOT using two different models. In the first model, the materials of the core mockup [uranium, aluminum, air (void), etc.] were assumed to be blended together and distributed evenly within the steel pipe; that is, the model did not include neutron streaming paths per se. In the second model, annular rings of void were interspersed between rings of homogenized fuel pin material: representing the fuel pin array, the rings being chosen to maintain the non-pin volume (44%) in the experimental configuration. The first heterogeneous model had nine void annuli and eight fuel annuli. In each case, only four outer iterations were allowed to converge the induced fission source. As an indication of the convergence after four outer iterations, the DOT iteration monitor showed a final source ratio of 0.998, or convergence to within 0.2%.

For both the homogeneous model and the heterogeneous model, the calculated emergent fluxes were transported to the detector positions via a modified version of the SPACETRAN III code. The fluxes were then folded with the detector response functions to obtain neutron count rates (for comparison with the Bonner ball measurements) and neutron energy spectra (for comparison with the NE-213 spectrometer and hydrogen counter measurements).

Comparisons of the calculated and measured Bonner ball count rates are shown in Table 6.1. It is apparent that with the homogeneous model the calculation badly underestimates the count rates at 0 and 2.5 deg, but improves at 5 deg, and is reasonable at 7.5 deg. The calculation with the heterogeneous model overestimates the count rates at all angles.

Table 6.1. Comparison of Calculated and Measured Bonner Ball Count Rates on Arc 9.144 m Beyond Fuel Lattice

Detector	Detector Angle (deg)	Ratio: Calculated/Measured	
		Homogeneous Model	Heterogeneous Model ^a
3-in.	0	0.019	4.01
	2.5 ^b	0.079	1.39
	5.0 ^b	0.731	2.42
	7.5 ^b	0.896	1.77
6-in.	0	0.012	4.37
	2.5	0.061	1.61
	5.0	0.687	2.63
	7.5	0.861	1.71
10-in.	0	0.010	4.23
	2.5	0.052	1.52
	5.0	0.656	2.51
	7.5	0.802	1.50

^aModel with nine void annuli and eight fuel annuli.

^bBackground greater than 40% of total measurement.

Comparisons of the calculated and measured spectra reveal these same trends. Figure 6.3 shows, for example, that for the 0-deg position the homogeneous model underestimates the neutron flux in all energy groups and, moreover, the calculated and measured spectra differ in shape. Conversely, the heterogeneous model overestimates the flux; however, the calculated and measured spectra are similar in shape, especially for energies above 0.1 MeV. At the 7.5-deg position, both models give better agreement with the measurements (see Fig. 6.4).

It was expected that the results obtained with a heterogeneous model would move toward those obtained with the homogeneous model as the number of annular rings of fuel and void were increased. This theory was tested with a second heterogeneous model consisting of 24 void ring, and 23 fuel rings, the rings being chosen to more nearly represent the void and fuel fractions of the experimental assembly as a function of radius. The result was that the total calculated fluxes decreased slightly at the 0- and 7.5-deg positions and increased slightly at the 2.5-deg position. That is, the calculated fluxes did move toward the homogeneous fluxes (and toward the measurements) at the 0- and 7.5-deg locations, but they moved away from them at the 2.5-deg location, probably because the thinner rings near the center of the model were allowing more

Fig. 6.3. Comparison of Measured (0-deg) Spectrum of Neutrons Beyond Fuel Lattice with Spectra Calculated Using Heterogeneous and Homogeneous Models. Detector 9.144 m beyond configuration.

Fig. 6.4. Comparison of Measured (7.5-deg) Spectrum of Neutrons Beyond Fuel Lattice with Spectra Calculated Using Homogeneous and Heterogeneous Models. Detector on arc 9.144 m beyond configuration.

uncollided neutrons to reach the 2.5-deg detector position. Interestingly, the neutron leakage from the fuel-lattice assembly calculated with either heterogeneous model was less than a factor of 2 higher than the leakage calculated with the homogeneous model, but at a distance of 9.144 m the heterogeneous fluxes were factors of approximately 400, 30, and 2 higher at the 0-, 2.5-, and 7.5-deg positions, respectively. Thus a comparison of leakages alone would not be the proper procedure for correcting homogeneous results for streaming effects at the two small angles, but it might be appropriate for angles greater than 7.5 deg.

A comparison of the uncollided fraction of the total flux calculated with the two heterogeneous models showed that at the 0- and 2.5-deg positions the uncollided fluxes dominated; therefore, it would seem that the measured fluxes at these positions also would have been largely uncollided fluxes (uncollided after leaving the spectrum modifier). However, at the 0-deg position the uncollided fluxes obtained with both models were higher than the measured fluxes by factors greater than 3. To investigate this further, uncollided fluxes were calculated with a third heterogeneous model which had 46 fuel rings and 47 void rings. In this case, the uncollided fluxes at all three locations were lower than the measured fluxes and the flux at the 0-deg location was almost identical with the measured flux. Additional calculations showed that the shapes of the 0- and 2.5-deg spectra calculated with the first heterogeneous model agree with the shape of the spectrum calculated for the neutrons leaking from the spectrum modifier, all of which tends to confirm that the fluxes at 0- and 2.5-deg are largely uncollided fluxes. By contrast, the spectra calculated with both the homogeneous and the heterogeneous models at the 7.5-deg location are similar. The 7.5-deg spectra are also comprised of lower energy neutrons, which indicates that the fluxes at 7.5 deg are largely collided fluxes.

With SPACETRAN apparently giving reliable results for positions beyond the configuration, one calculation was performed using the FALSTF code to calculate the collided fluxes, still using SPACETRAN for the uncollided fluxes since FALSTF did not seem to work properly for the uncollided fraction. The FALSTF collided fluxes were factors of 11.7, 2.2, and 1.4 times smaller than the SPACETRAN collided fluxes at 0, 2.5, and 7.5 deg, respectively. The result was somewhat better agreement with the measured fluxes, the calculated to measured ratios being .17 at 2.5 and 7.5 deg and 3.39 at 0 deg.

6.3.3. Monte Carlo Transport Calculations for Fuel Lattice

With such large discrepancies between the DOT-calculated and measured results, a Monte Carlo calculation was needed as a calculational benchmark to separate modelling errors from other errors. Although in principle the Monte Carlo technique will allow the experimental configuration to be modelled exactly, the large number of fuel pins used in the experimental assembly precluded including in the model the one-half of the assembly (180-deg segment) needed for an exact calculation of the responses at all detector positions. Therefore, the calculations were begun with a 15-deg segment, and to reduce the size of the calculation, the fuel pins were homogenized (cladding and UO_2 were combined) and the water layer beyond the fuel pin lattice was limited to a 167.4-mm thickness which sensitivity calculations (see below) had indicated would be adequate. An albedo boundary was placed at the 167.4-mm depth, and albedo data generated with the DOT code were used in the calculation.

The resulting total $E > 0.015$ MeV flux calculated for the 0-deg detector position was higher than the measured flux by a factor of 1.68, with 89.6% of the calculated flux being uncollided. The corresponding DOT-calculated fluxes had been higher than the measured fluxes by factors of 4.35 and 4.11 (first and second heterogeneous models). Subsequently a Monte Carlo calculation was successfully performed for a 90-deg segment of the fuel lattice and it yielded similar results: total fluxes that were 1.83 times higher than the measured fluxes, with 91.2% of the calculated fluxes being uncollided. In this case a heterogeneous model was used, but the fuel-pin cladding and UO_2 were not made distinct zones; that is, they were homogenized to form a single pin zone.

On the basis of these comparisons, it can be stated that the DOT calculations with heterogeneous models are very conservative, overestimating the detector responses by several factors. Of course, the Monte Carlo calculations also overestimate the flux, but much less so than the discrete ordinates calculations. The Monte Carlo discrepancy is largely attributed to the basic data used to describe the experiment since the Monte Carlo statistics for the total flux are reasonably good (10% standard deviation).

6.3.4. Sensitivity Analysis

As pointed out above, the Monte Carlo calculations were guided somewhat by a discrete ordinates sensitivity analysis, which was performed with the two-dimensional homogeneous model. In fact, the main purpose for performing a sensitivity analysis was to identify the "importance regions" (both geometric regions and energy regions) for the Monte Carlo calculation, and, in particular, to determine the minimum thickness of water that had to be included in the calculational model.

The general procedure in a sensitivity analysis is to link the usual "forward" discrete ordinates calculation (that is, the homogeneous calculation discussed earlier) with an "adjoint" (backward) calculation in which the source is the detector response function. The calculated results then show to which geometric regions or to which particle interactions the detector response is most sensitive.

In the sensitivity analysis performed for the fuel lattice configuration two adjoint calculations were performed, one using as its source the response function of the 6-in. Bonner ball at a location 1.524 m behind the configuration and the other using the response function at a location 3.048 m behind the configuration. The two detector positions were required in order to determine the variation of the sensitivity with detector position. The code GRTUNCL was used to generate a first-collision source for the adjoint calculations so that a point response rather than a volume-averaged response could be obtained, and the VIP code was used to prepare from the two-dimensional forward and adjoint flux moments the one-dimensional form required by the SWANLAKE sensitivity code.

In addition to guiding the Monte Carlo calculations, the sensitivity analysis revealed the cross sections and material compositions to which the 6-in. Bonner ball responses were most sensitive. This, in turn, provided a priority list of areas where uncertainties should be reduced for subsequent GCFR calculations.

6.4. CONCLUSIONS

The fuel-lattice experiment demonstrated conclusively that neutron streaming could be effected in an experiment. Analyses of the experiment with the discrete ordinates code DOT showed that a homogeneous cylindrical model underestimated the streaming effect

and a heterogeneous model consisting of concentric annuli of fuel and void overestimated the effect. Thus DOT calculations can be used to bracket the streaming that could be expected in a GCFR-type core.

The analysis also showed that the discrete ordinates calculations gave energy spectra that generally agreed in shape with the experimental spectra. For angles less than 5 deg from the axis of the configuration, the spectra were very similar to the energy spectrum of neutrons immediately behind the spectrum modifier, which indicated that the detector response was largely due to neutrons transmitted through the fuel lattice without undergoing scatterings (that is, to the uncollided fraction).

The overall conclusion was that neutron streaming in a GCFR core mockup or in the GCFR proper could be approximately calculated with a combination of discrete ordinates and Monte Carlo calculations. Discrete ordinates calculations with a heterogeneous model would establish the upper limits of the streaming and determine the energy spectra. The results could then be normalized to a Monte Carlo calculation to more nearly predict the magnitude of the streaming with assurances that the predictions would be conservative.

6.5. REFERENCES

1. C. O. Slater and D. E. Bartine, "Preliminary Analysis of a TSF Experiment on Neutron Streaming Through a Lattice of GCFR-Type Fuel Pins," GCR-76/37 (November 1976).
2. C. O. Slater and M. B. Emmett, "Final Analysis of a TSF Experiment on Neutron Streaming Through a Lattice of GCFR-Type Fuel Pins," ORNL-GCR-78/5 (February 1978).
3. C. O. Slater and M. B. Emmett, "Analysis of a Fuel-Pin Neutron-Streaming Experiment to Test methods for Calculating Neutron Damage to the GCFR Grid Plate," in Proc. Fifth Int. Conf. Reactor Shielding, pp. 873-880, Science Press, Princeton, N. J., 1977.

7. GCFR GRID-PLATE SHIELD DESIGN CONFIRMATION EXPERIMENT

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7.1. INTRODUCTION

As has been emphasized in Chapters 5 and 6, from the earliest GCFR designs it was apparent to shield designers that axial neutron streaming through the helium passageways within the reactor core could result in intense fluxes of high-energy neutrons impinging on structure above and below the core. Of particular concern was possible neutron damage to the grid plate, a 61-cm-thick stainless steel plate which was to hold the fuel and blanket elements firmly in position. At the time these studies were being performed, the GCFR design specified that the grid plate would be top-mounted and would contain 16.2-cm-diam holes that would mate with the cylindrical upper sections of the fuel elements. Protective shielding was to be provided in the upper sections of the fuel elements (see Fig. 2.3 in Chapter 2) but just how effective the shielding would be in preventing damage to grid plate could not be accurately predicted. On the basis of the studies described in the two preceding chapters it was known that the streaming effect could be bracketed with available analytical techniques, but more precise information was needed since the grid plate would be required to last the lifetime of the reactor and overdesign of its shielding would entail a high economic penalty. Thus experimental quantification of the neutron streaming in a prototypic geometry seemed essential, and the decision was made to perform what was to be labeled the "GCFR grid-plate shield design confirmation experiment."

To ensure that an experiment could be performed that would provide the needed answers, exhaustive calculations were carried out both to guide the experimental design and to predict the measurements that would be obtained with the as-designed configuration. These calculations were described in an ORNL GCFR progress report (ref. 1) and are summarized below in Section 7.3, which is preceded by a brief description of the experiment as it was finally performed at the ORNL Tower Shielding Facility in 1978 (see Section 7.2). The experiment is presented in more detail in refs. 1, 2 and 3, and its final analysis, summarized here in Section 7.4, is described in refs. 3 - 5.

7.2. THE EXPERIMENT

Design calculations for the grid-plate shield confirmation experiment (see Section 7.3) were begun in January, 1977, following a meeting with General Atomic at which ORNL presented a preliminary conceptual design for the experiment and preliminary estimates of

the adequacy of the design. Primary concern was that the source come mainly from the fuel pins and that the seven subassemblies would adequately represent large array effects. Detailed design features were developed during preanalysis and during several meetings with General Atomic Company.

The full configuration as finally designed is shown in Fig. 7.1. The first section, the simulation of the GCFR core and axial blanket, consisted of slightly enriched UO_2 fuel pins (1.99% ^{235}U) arranged in seven subassemblies that resembled GCFR subassemblies. The fuel pins were formed from 10.95-mm-diam fuel pellets placed end-to-end in aluminum tubing having 0.815-mm-thick walls. The fueled section of each pin was 121.92 cm long and contained 1035.49 g of uranium. Aluminum studs on each end of the fuel pins extended their overall length to 127 cm.

Fig. 7.1. Experimental Configuration for TSF-GCFR Grid-Plate Shield Confirmation Experiment. (Note: "Bottom" of core is adjacent to shadow shield.)

The casing for each of the seven fuel subassemblies was hexagonal in cross section and was constructed from 0.30-cm-thick carbon steel. Two types of subassemblies were fabricated: one set in which each subassembly contained 127 pins on a pitch of 1.58 cm; and another in which each subassembly contained 91 pins on a pitch of 1.80 cm. Support grids that were 0.318 cm thick were placed at each end of the assembly and at the midsection to maintain the proper pitch of the fuel pins. The distance between the facing surfaces (across flats) of the subassemblies was 18.3 cm, and, like the fuel pins, the subassemblies were 127 cm long.

Details of the fuel pins and subassemblies are shown in Figs. 7.2 - 7.5. The small steel pieces visible in Fig. 7.4 near the ends and middle of the outer flat surfaces of the subassemblies were used to vary the spacing between subassemblies for some of the measurements. In the experiment the cluster of seven subassemblies (a central subassembly surrounded by six subassemblies) was surrounded by a 0.95-cm-thick 18-sided iron cylinder (see Fig. 7.5). The cylinder size was fixed; thus, as the spacing between fuel subassemblies decreased, the void thickness between the outer surface of the fuel cluster and the surrounding cylinder increased.

Preceding the core-blanket section in the experimental configuration was a 76.2-cm-diam by 30.5-cm-thick concrete "shadow shield." As noted in Fig. 7.1, this shield surrounded the "bottom" section of the fuel and was itself surrounded by a void that extended "upward" beyond the shield to surround a large fraction of the fuel section. With this arrangement the section simulating the GCFR core and blanket was protected from direct radiation from the TSR-II and the primary radiation source for the experiment became the neutrons produced by fissions in the fuel pins, the fissions being induced by TSR-II neutrons that scattered around the shadow shield into the fuel section. Both the shadow shield and the concrete surrounding the void were lined with iron. The outer surface of the shadow shield and the face of the shadow shield adjacent to the fuel pins were covered with 0.318 cm of iron, while the surrounding concrete was lined with 0.95 cm of iron.

The second section of the experimental configuration, the grid-plate shield simulation, consisted of seven 45-cm-long subassemblies that were also hexagonal in cross section to match the "upper" ends of the seven experimental fuel subassemblies. (In the GCFR the shield is integral with the fuel assembly and is situated between the end of the

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Fig. 7.3. Fuel-Pin Support Grids at Center and at Each End of Experimental Fuel Subassemblies.

ORNL PHOTO 4517-77

Fig. 7.4. Experimental Fuel Pin Subassemblies: Empty Subassembly (Right); Full Subassembly (Left); and Complete Subassembly with End Plates (Center).

ORNL PHOTO 5154-77

Fig. 7.5. Experimental Mockup of Seven Fuel Pin Subassemblies (Surrounded by Iron-Lined Concrete).

fuel pins and the grid plate.) A concrete envelope surrounding the shield subassemblies had an 18-sided inner iron liner that matched the iron cylinder surrounding the fuel subassemblies, and the spacing between the shield subassemblies was the same as that between the fuel subassemblies.

Details of the grid-plate shield subassemblies are shown in Figs. 7.6 - 7.8. The shielding consisted of B_4C that filled a space between inner and outer steel liners of the subassemblies. The B_4C was vibra-packed to densities varying from 76.2 to 77.2% of the theoretical density in the GCFR, and its thickness varied from 1.1 cm at the fuel end of the subassembly to 3 cm at the grid-plate end. These thicknesses were defined by the thicknesses of iron pieces at each end of the subassembly, which reduced the overall length of the B_4C section to 32.3 cm.

Fig. 7.6. Experimental Grid-Plate Shield Subassemblies: Fuel-Pin Interface (Left); and Grid-Plate Interface (Right).

Fig. 7.7. Schematic of Grid-Plate Shield Subassembly.

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Fig. 7.8. Experimental Mockup of Seven Grid-Plate Shield Subassemblies (Surrounded by Iron-Lined Concrete).

The final section of the experimental configuration, the grid-plate simulation, consisted of a 63.5-cm thickness of iron containing seven 16.5-cm-diam holes whose axes coincided with the axes of the fuel and shield subassemblies (to simulate continuous passageways for helium coolant flow and control rod drives). Removable sleeves filled with B_4C followed by 5.9 cm of iron mocked up the portion of the GCFR grid-plate shield designed to project into the grid plate (see Fig. 7.1). The 63.5-cm thickness for the grid-plate mockup was chosen to represent the 61-cm-thick GCFR grid plate plus a 2.5-cm thickness of iron as (1) a spacer to properly position the cylindrical sections with respect to the grid plate and (2) filler to partially simulate the thick steel section of the subassembly duct walls where a transition is made from a hexagonal shape to cylindrical shape.

A summary of the various configurations measured in the experiment is given in Table 7.1. Spectral measurements were made beyond the configurations with the TSF NE-213 spectrometer and with the TSF hydrogen counters. Other detectors used were the Hornyak button and the various diameter Bonner balls. The results of the measurements are compared with the calculated results in Section 7.4 below.

7.3. DESIGN AND PREANALYSIS OF THE EXPERIMENT

As pointed out above, the configuration described in Section 7.2 was designed on the basis of a series of neutron transport calculations in which the various components were modeled. Since the configuration was to be as prototypic as was practical, it was necessary that calculations also be performed for the GCFR proper in order to judge how well the results predicted for the experimental configuration agreed with those predicted for the GCFR. Throughout the design an attempt was made to meet all the requirements set for the experiment by General Atomic.

One of the primary requirements specified by GA was that the core section adequately resemble a semi-infinite fuel rod lattice corresponding to the actual reactor and that it have sufficient multiplication that the radiation source for the experiment would emanate primarily from within the rods. Another was that the spacing between the fuel rods and between subassemblies of the fuel rods be varied to essentially bracket the core design options under consideration.

Table 7.1. Descriptions of the GCFR Grid-Plate Shield Experiment Configurations*

Configuration	Description
II.A	Empty cavity hidden by concrete shadow shield and surrounded by steel-lined concrete
II.B	II.A plus seven subassemblies spaced 1.27 cm apart and containing 127 fuel pins per subassembly
II.C	II.B plus steel-lined concrete section for grid-plate shields
II.D	II.C plus seven grid-plate shield subassemblies spaced 1.27 cm apart
II.E	II.D plus a 61-cm-thick stack of iron plates located 15.24 cm from the grid-plate shields
II.F	II.D followed by the prototypic grid-plate mockup
III.A	II.A plus seven subassemblies spaced 0.635 cm apart and containing 127 fuel pins per subassembly
III.B	III.A plus seven grid-plate shield subassemblies spaced 0.635 cm apart
III.C	III.B plus a 61-cm-thick stack of iron plates located 15.24 cm from the grid-plate shields
IV.A	II.A plus seven subassemblies spaced 1.27 cm apart and containing 91 fuel pins per subassembly
IV.B	IV.A plus seven grid-plate shield subassemblies spaced 1.27 cm apart
IV.C	IV.B plus a 61-cm-thick stack of iron plates located 15.24 cm from the grid-plate shields
V.A	IV.A minus seven fuel pins from the center of the central fuel subassembly
V.B	IV.B minus seven fuel pins from the center of the central fuel subassembly
VI.A	II.A plus seven subassemblies spaced 0.318 cm apart and containing 127 fuel pins per subassembly
VI.B	VI.A plus seven grid-plate shield subassemblies spaced 0.318 cm apart

*Note that all components--fuel pin subassemblies, the grid-plate shield subassemblies, and the grid plate--are surrounded by concrete in the radial direction.

In order to simulate the axial streaming of the GCFR, it was necessary that the flux of neutrons coming directly from the TSR-II core be minimized, which dictated that a shadow shield like that shown in Fig. 7.1 be used between the reactor and the simulated GCFR fuel-blanket section. With a sufficiently thick shadow shield, the uncollided neutron source that had dominated the measurements described in Chapters 5 and 6 could be essentially eliminated, and the radiation source in the experiment would largely consist of neutrons that were born within the simulated GCFR core and blanket and then streamed from the core along the helium coolant passageways.

The first design-support calculations therefore were performed to establish the thickness of the shadow shield for the experiment and to ascertain whether adequate flux levels could be attained by modeling only seven GCFR-type fuel subassemblies. The calculations were two-dimensional DOT-IV calculations using the 51-group GCFR cross sections and a symmetric 70-angle (S_{10}) quadrature (ref. 1). As in Section 6, the source for the calculations was obtained by projecting the angular fluxes emerging from the hemispherical TSR-II onto a disk via the Marker technique (ref. 6). These calculations indicated that a 30.5-cm-thick shadow shield would be adequate and that fissions in seven subassemblies of GCFR design would provide high enough flux levels.

Subsequently the calculations were repeated for the slightly enriched UO_2 rods available for the experiment. These calculations, for which the 51-group cross sections were collapsed to seven groups* and a symmetric 30-angle (S_6) quadrature was employed, showed that the fission streaming source from the rods would still be adequate, and, moreover, that nearly 90% of the neutron leakage from the pin array would be due to fissions within the array. The calculations also indicated that while most fissions in the fuel pin array would occur near the shadow shield, the fissions that occurred near the grid-plate shield (i.e., within the axial blanket) would be more important in providing a grid-plate shield source.

Concern about the effect of a variable water content in the concrete on the neutrons scattering into the pin array, and therefore on the fissions produced in the array, prompted another seven-group S_6 calculation, which showed the effect would be minimal.

*A comparison of 51-group and 7-group results showed that the coarse-group calculations were adequate for determining integral quantities, such as total fission source and total neutron leakage, but not for predicting energy distributions.

A third calculation investigated the radial distribution of fissions within the cluster of fuel pin subassemblies. The distribution was found to peak significantly at the outside edge of the cluster, which would not fulfill the semi-infinite lattice requirement, but the distribution became flatter as the thickness of a steel liner surrounding the subassemblies and lining the concrete was increased (owing to absorption of thermal neutrons in the steel). With the liner thicknesses finally selected, the flux still peaked but because of an apparent redistribution of the neutron population through scattering collisions, the flux impinging on the grid plate shield was predicted to be relatively flat.

Another set of calculations concentrated on the details of the design of the grid-plate shield subassemblies. Because a thinner steel casing was being used for the experimental shield subassemblies than would be used in the GCFR subassemblies, the total B_4C volume in the experimental subassemblies exceeded that in the GCFR proper, but the less dense packing of the B_4C in the experimental configuration was found to be a compensating factor. The calculations indicated, however, that the B_4C sleeves shown in Fig. 7.1 should be included in the configuration; the sleeves would be present in the cylindrical end section of the GCFR fuel subassembly and their absence in the experiment would have drastically altered the measured results and their relevance to the GCFR. The use of steel sleeves in the experiment was considered, but the calculations indicated that with the steel sleeves the neutron flux emerging from the grid plate would be 40% higher than it would be with the B_4C sleeves.

Calculations were also performed to predict how variations in the widths of the gaps between the fuel and shield subassemblies would affect the measurements. It had been specified that gap widths of 3.2, 6.4, and 12.7 mm should be used, but it was feared that as the gap width was decreased, neutron streaming between the outer surface of the pin array and its surrounding steel liner might dominate the measurements. The calculations indicated that when the grid plate was not present, the peripheral gap would have no noticeable effect on the neutron flux emerging from the shield subassemblies. When it was assumed that the grid plate was in position, however, the flux leaving the shield subassemblies increased a maximum of 7% a few centimeters inside the peripheral gap for the maximum gap width. This difference did not appear to warrant plugging of the gap in the experiment.

Finally, calculations using the 51 energy groups and a symmetric 96-angle (S_{12}) quadrature were performed to determine whether the neutron energy spectrum within the

experimental assembly, would be prototypic of a GCFR-type spectrum, especially in the grid-plate shield and grid-plate regions. Comparisons of calculated spectra incident on and exiting from the grid plate for the experimental configuration (91 pins per subassembly) and the GCFR design (low-pressure-drop, beginning-of-life case) are shown in Fig. 7.9. While the spectra of the experimental assembly are slightly harder, it was concluded that the experimental spectra resembled the GCFR spectra sufficiently to provide relevant and interpretable data.

7.4. ANALYSIS OF THE EXPERIMENT: COMPARISON OF CALCULATED AND MEASURED RESULTS

Table 7.1 lists a total of 16 experimental configurations. Of these, eight were examined in detail during the final analysis of the experiment, the analysis being carried out concurrently with the measurements and extending into early 1979. The specific configurations for which calculations were performed were the empty cavity configuration (II.A), four configurations extending through the fuel-pin section (II.B, III.A, IV.A and VI.A), two configurations extending through the grid-plate shield section (II.D and IV.B), and the final configuration that included the grid-plate mockup itself (II.F). In addition, some inferences were made from the measured results for other configurations.

Fig. 7.9. Comparison of Neutron Spectra Calculated for the TSF Experimental Assembly and the GCFR Design. Spectra were calculated for neutrons emerging from the grid-plate shield (left) and for neutrons emerging from the grid plate (right). The TSF calculation included 91 pins per assembly and the prototypic grid plate; the GCFR calculation was for the low-pressure-drop beginning-of-life design.

7.4.1. Definition of Source

As has been pointed out in the discussion above, the penetrating radiation source for the grid-plate shield design confirmation experiment consisted almost entirely of neutrons born from fissions in the fuel pins simulating the GCFR core, the contribution from direct TSR-II neutrons being reduced by the insertion of a thick concrete shadow shield in the TSR-II beam ahead of the fuel pins. The beam was thus directed around the shadow shield and forced into a near perpendicular incidence onto the fuel pins.

In the analysis the initial TSR-II "driver" source was obtained in the same manner as has been described in Chapters 5 and 6. That is, the angular fluxes emerging from the hemispherical surface of the TSR-II tank were projected onto a disk-shaped source by a technique similar to Moerker's (ref. 6), the location of the disk being at the inner surface of the large-beam collimator as shown in Fig. 5.3. This source was then used as the input source for a two-dimensional discrete ordinates DOT-3.5 calculation to determine the first of a series of internal boundary sources that would be used as input sources throughout the analysis. The two-dimensional geometry extended through the beam collimator and on through the shadow shield, so that the internal boundary source, which was output in the plane including the inner surface (reactor side) of the shadow shield, would include the effects of backscattering from the shadow shield.

The resulting output boundary source was then used as the input source for a transport calculation extending through the fuel-pin section of the experimental configuration. Similarly, an internal boundary source output from the calculation for the fuel-pin section was used as the input source for a transport calculation through the grid-plate shield section, and so forth.

The initial source calculation used the 51-group GCFR cross sections with P_3 expansion and the 150-angle quadrature set. "As the analysis progressed through the configuration, any changes in the angular quadrature and spatial meshes required between succeeding calculations were effected with the FACT code."

*The FACT code was written for use with DOT-III; therefore, boundary sources output from DOT-IV had to be preprocessed before being input into FACT. Conversely, output sources from FACT had to be postprocessed before being input into DOT-IV.

7.4.2. Transport Calculations for Empty Cavity Configuration

The empty cavity configuration, shown in Fig. 7.10, is described in Table 7.1 as Configuration II.A. In the experiment measurements were made both along the axis of the cavity and along a radial traverse 30.5 cm behind the concrete shield containing the cavity. In addition, measurements were made at a point approximately 244 cm behind the configuration as a check on the calculations of the emerging neutron spectrum.

Calculations for this configuration were performed with the discrete ordinates code DOT-IV (two dimensions, 51 neutron groups, P_3 cross sections, and the 150-direction biased quadrature set) and with the Monte Carlo code MORSE. Comparisons of the calculated and measured Bonner ball and Hornyak button responses along the cavity axis are shown in Fig. 7.11, where the sketch of the surrounding void and concrete at the bottom of the figure aids in identifying the relative locations of the measurements.

Figure 7.11 shows that except for the point nearest the shadow shield, the DOT and MORSE results agree with each other and that they agree in shape with the measured results. However, the calculated responses are higher in magnitude than the measured responses: by 45% for the Hornyak button, by 30% for the 2-in. Bonner ball (cadmium-covered), and by 55% for the 5-in. Bonner ball, all of which exceed the desired maximum discrepancy of 20%. The same trends were observed for the radial traverses at 30.5 cm; and for the measurements at 244 cm, the calculations overpredicted 3-, 6-, and 10-in. Bonner ball count rates by factors of 1.52, 1.46, and 1.40, respectively.

Since the MORSE and DOT results were in general agreement, the discrepancy between the calculated and measured results would appear to lie within a common input, such as the description of the concrete composition, the DOT-generated source used in both calculations, etc. In an attempt to resolve the discrepancy, several additional calculations were performed, the first being to repeat the DOT calculations with a symmetric 96-angle (S_{12}) quadrature substituted for the 150-angle biased quadrature. The calculated results remained essentially unchanged, however.

Next, a sensitivity analysis was performed for the 5-in. Bonner ball response at 30.5 cm from the end of the configuration, using the VIP code and the FORSS system. This analysis showed that the count rate was mainly sensitive to the water content in the concrete, particularly the concrete of the shadow shield.

Fig. 7.10. Empty Cavity Configuration (Showing 30.5-cm Thick Concrete Shadow Shield and Surrounding Concrete).

Fig. 7.11. Relative Measured and Calculated Detector Responses Along Axial Traverses in Empty Cavity Configuration (II.A). BB = Bonner ball response (counts/min·W); HB = Hornyak button response ($\text{ergs} \cdot \text{g}^{-1} \cdot \text{h}^{-1} \cdot \text{W}^{-1}$).

In order to investigate the effect of the concrete composition further,^a the percent changes in the atom densities of the elemental components of several concrete mixes from the atom densities assumed in the analysis were multiplied by the computed sensitivity coefficients. The results, presented in detail in ref. 5, indicated significant changes in the response of the 5-in. Bonner ball with changes in the concrete composition. This observation was supported by seven-group calculations run for the various concretes; however, the only conclusion that could be deduced from these calculations is that a knowledge of the concrete composition is very important for accurate calculations of this configuration.

The forward and adjoint calculations from the sensitivity analysis were also used to produce a channel theory plot that showed the flow of important neutrons (contributions) through the configuration. This plot revealed that the major flow was around the shadow shield, as was intended. To test whether the mesh used for the shadow shield was adequate, additional forward calculations were performed with seven energy groups using both the original mesh and a mesh in which the intervals were doubled in both the radial direction and the axial direction. The results from the two calculations differed little from each other or from the earlier 51-group results. Thus, the cause for the discrepancies between the calculated and measured results was not definitely established.

7.4.3. Transport Calculations for Fuel-Pin Configurations

The fuel-pin configurations included in the experiment, that is, the configurations containing only the simulation of the GCFR fuel and axial blanket (see Fig. 7.1), are those listed in Table 7.1 as II.B, III.A, IV.A, V.A and VI.A. The differences among these configurations are in the number of fuel pins used per subassembly and in the spacing between subassemblies (see in Table 7.2). As noted earlier, the range of spacings was selected to cover the design options under consideration for the GCFR. Configuration V.A was a special case in that seven fuel pins were removed from the center of the central subassembly to simulate a control-rod channel.

^aNote: The composition for concrete used in this analysis was the "standard TSF concrete" (see Table 10.2 in Chapter 10).

Table 7.2. Fuel and Subassembly Arrangements in Fuel-Pin Configurations

Configuration	Subassembly Spacing	Fuel Pins per Subassembly	Total Number of Fuel Pins in Configuration
II.B	1.27 cm	127	889
III.A	0.635 cm	127	889
IV.A	1.27 cm	91	637
V.A	1.27 cm	91*	630*
VI.A	0.318 cm	127	889

*Only 84 fuel pins in central subassembly, seven fuel pins having been removed from the center of the subassembly to simulate a control-rod channel.

This first section of the experimental configuration was studied in considerable detail for several reasons. First, it was necessary to define the source leaving the axial blanket region and entering the grid-plate shield region and, moreover, to determine whether that source could be accurately calculated. Successfully calculating the source not only was essential to the analysis of the full experimental configuration but also would indicate that the source leaving the axial blanket region of the GCFR proper could be calculated. Studying these configurations also provided an additional opportunity to examine neutron streaming in GCFR-type fuel-pin lattices, this time in the absence of an overpowering "uncollided" source that originated outside the fuel pins (as occurred in the experiments described in Chapters 5 and 6). Finally, being able to compare the results for the different fuel spacings would especially provide insight into the streaming mechanism.

In the experiment, Bonner ball and Hornyak button measurements were made along an axial traverse behind each configuration and also along radial traverses beyond Configuration II.B. In addition, NE-213 and hydrogen counter spectral measurements were made behind Configuration II.B.

Two-dimensional discrete ordinates calculations (51 groups, P_3 cross sections, 150-angle biased quadrature) were performed to obtain calculated detector responses for all the fuel pin configurations except Configuration V.A, which was not analyzed. Three-dimensional Monte Carlo calculations were performed for Configuration II.B only.

For points out to 30.5 cm beyond the configurations, the two-dimensional calculations used the DOT-IV code alone, but for distant points (specifically, at 138.1 cm and 305 cm beyond the configurations) the DOT results through the configuration were transported on to the detector point via the DISKTRAN code. In all these calculations the fuel-pin region was modeled as concentric annuli of void, fuel, steel, etc. The major boundaries were determined by the eighteen outer steel walls surrounding the outer subassemblies and the six (double) interior walls around the central subassembly. The remaining twelve walls were assumed to be intimately mixed with the fuel pins of the outer six subassemblies, the fuel regions being divided, according to their respective volume fractions, into alternating fuel and void rings.

For the MORSE calculation of Configuration II.B, a 1/12th segment of the geometry was mocked up almost exactly, with cladding and fuel regions included for each pin. The MORSE calculation was actually a combination of two calculations, one in which the boundary source generated by the DOT calculation was used as a fixed source and another in which the fission density distribution in the fuel pins calculated by DOT was used as a fixed source. The results were combined by multiplying the separate results for a unit source by the total source for each and adding the two results. The DOT sources were used because it would have been extremely difficult for MORSE to converge the fission source in the presence of the fixed boundary source.

Comparisons of the DOT-calculated and measured Bonner ball and Hornyak button responses on the axis 30.5 cm beyond the four analyzed fuel-pin configurations are shown in the upper half of Table 7.3, along with the measured values for Configuration V.A. Similar comparisons of the DOT/DISKTRAN-calculated and measured responses on the axis 305 cm beyond the configurations are shown in the lower half of Table 7.3. It is apparent that for on-axis detector locations close behind the configurations, the DOT-calculated results agree well with the measured responses of both the 2-in. cadmium-covered Bonner ball and the 5-in. Bonner ball. The calculated results for the Hornyak button, however, are 35% to 70% higher than the measured responses. Comparisons of the radial traverses at 30.5 cm beyond Configuration II.B show the same trends.

Comparisons of the DOT/DISKTRAN-calculated and measured Bonner ball responses on the axis 305 cm beyond the fuel-pin configurations (see lower half of Table 7.3) do not show

Table 7.3. Comparison of DOT-Calculated and Measured Detector Responses on Axis 30.5 cm and 305 cm Beyond Various Fuel-Pin Configurations

Detector	Configuration				
	II.B (1.27,127)*	III.A (0.635,127)	IV.A (1.27,91)	V.A (1.27,91) [†]	VI.A (0.318,127)
Response at 30.5 cm Beyond Configuration					
Bonner ball (counts/min-W)					
2-in. Cd-covered ball					
Calculated	8.30+1 [‡]	8.48+1	1.17+2		8.43+1
Measured	8.80+1	8.63+1	1.43+2	1.46+2	8.69+1
Calc./Meas.	0.94	0.98	0.82		0.97
5-in. ball					
Calculated	8.53+3	8.38+3	1.18+4		8.15+3
Measured	8.53+3	7.95+3	1.25+4	1.29+4	7.88+3
Calc./Meas.	1.0	1.05	0.94		1.03
Hornyak button (ergs/g-h-W)					
Calculated	4.18-2	4.10-2	6.39-2		3.97-2
Measured	2.99-2	2.55-2	4.75-2	4.24-2	2.34-2
Calc./Meas.	1.40	1.61	1.35		1.70
Response at 305 cm Beyond Configuration					
Bonner ball (counts/min-W)					
2-in. Cd-covered ball					
Calculated	6.32+0	6.39+0	8.87+0		6.45+0
Measured	4.32+0	3.28+0	7.28+0	7.21-0	3.61+0
Calc./Meas.	1.46	1.95	1.22		1.79
5-in. ball					
Calculated	4.65+2	4.96+2	6.68+2		4.91+2
Measured	3.48+2	3.16+2	5.30+2	5.50+2	3.03+2
Calc./Meas.	1.34	1.57	1.26		1.62
Hornyak button (ergs/g-h-W)					
Calculated	2.87-3	3.13-3	4.71-3		3.09-3
Measured	1.68-3	1.43-3	2.75-3	2.79-3	1.35-3
Calc./Meas.	1.71	2.19	1.71		2.29

* (1.27,127) indicates 1.27 cm between fuel subassemblies, 127 fuel pins per subassembly.

[†] Only 84 fuel pins in central subassembly.

[‡] Read: 8.30×10^1 .

a similar agreement. All the calculations substantially overpredict the measurements, the results for the Hornyak button again reflecting the greatest disagreement. The radial traverses beyond Configuration II.B at 305 cm showed that the overprediction decreased as the detectors moved away from the axis (that is, from the line-of-sight of the fuel lattice), but only in the case of the 5-in. Bonner ball did the calculations approach the measurements and then only for off-axis locations greater than 40 cm. In general, the overpredictions appeared to be greatest for those fuel lattices having the highest fuel density.

Since the 5-in. Bonner ball has a response nearly proportional to the total neutron flux, intercomparing the 5-in. Bonner ball responses for the fuel-pin configurations indicates the effect of the fuel spacing on the transmitted flux. Table 7.3 shows that for the 30.5-cm location, increasing the subassembly spacing from 0.318 cm to 1.27 cm (that is, going from Configuration VI.A to Configuration II.B) increases the calculated 5-in. Bonner ball response by 5% and the measured response by 8%.

Comparing the Hornyak button measurements (fast neutron fluxes > 0.1 MeV) for Configurations IV.A and II.B indicate the effect of the void fraction within the fuel assemblies on the high-energy transport. At 30.5 cm, the measured value for Configuration IV.A, which had a void fraction of 60.2%, is higher by a factor of 1.6 than the measured value for Configuration II.B, which had a void fraction of 44.5%. The difference is due both to a lower effective material density and to increased streaming in Configuration IV.A. The calculations indicated that the scattering and fission sources per unit volume of material within the fuel subassemblies were nearly the same in the two assemblies.

While calculations were not performed for Configuration V.A, comparing the measured values for the configuration with those for Configuration IV.A shows that removing the seven central fuel pins from the central fuel subassembly (to simulate a control channel) increased the total fluxes (as deduced from the 5-in. Bonner ball responses) by 3.5% at both the 30.5-cm location and the 305-cm location. On the other hand, removing the seven fuel pins decreased the fast-neutron fluxes (as deduced from the Hornyak button responses) by 11% at the 30.5-cm location and increased them by 1.5% at the 305-cm location. This implies that the grid-plate shield (and grid plate) would have a lower

fast-neutron exposure with a control subassembly in place, and thus using the fast-neutron exposure obtained with a full fuel subassembly for design purposes would be the most conservative approach. On the other hand, the percentage increase in the total flux with the channel open is probably greater than it would be in the GCFR since the experiment had an inward flux gradient that drove the neutrons into the streaming channel (that is, TSR-II neutrons were scattered into the channel and also fission neutrons produced in the pins were backscattered from the shadow shield into the channel).

Another observation apparent from the tables is that, in general, the calculated-to-measured ratios for all the detectors appear to increase with a decreasing subassembly spacing and to decrease with a decreasing number of fuel pins.

The Monte Carlo calculations for Configuration II.B were performed for detector locations 0, 20, and 40 cm off the axis for both of the radial traverses (at 30.5 and 305 cm). For the 5-in. Bonner ball the Monte Carlo responses at 30.5 cm were in good agreement with both the DOT responses and the measured responses, the statistical mean of the Monte Carlo responses being 3 to 5% higher than the measured responses. For the 2-in. cadmium-covered Bonner ball, the Monte Carlo calculations were 4 to 7% higher and for the Hornyak button they were 18 to 21% higher. The differences were larger at 305 cm, the calculated response being higher than the measured response by 6 to 22% for the 2-in. Bonner ball, by 4 to 21% for the 5-in. Bonner ball, and by 27 to 64% for the Hornyak button (27% on the axis). The persistent disagreement of all the calculations with the Hornyak button measurements was viewed as a normalization problem with the Hornyak button response function.

Finally, the DOT-calculated and measured neutron spectra at an on-axis location 138.1 cm beyond Configuration II.B are compared in Fig. 7.12. The measured spectrum is a combination of NE-213 spectrometer measurements (0.8 to 15 MeV) and hydrogen counter measurements (0.08 to 1.5 MeV). As is apparent, the measured and calculated spectra were found to be in good agreement. Not unexpectedly, it was also found that when the measured fluxes were integrated over energy above 0.9 MeV and above 0.12 MeV, the measured fluxes and the corresponding calculated fluxes were again in agreement. In addition, calculated-to-measured ratios for 3-, 6-, and 10-in. Bonner balls at the 138.1-cm location were acceptable (1.27, 1.16, and 1.08, respectively). This

Fig. 7.12. Comparison of Calculated and Measured Neutron Spectra on the Axis 138.1 cm Beyond Fuel-Pin Configuration II.B.

demonstrated ability of the DOT code to calculate the high-energy neutron fluxes for all these detectors again indicates a normalization problem for the Hornyak button (see above).

From the preceding it is apparent that, except for the Hornyak button responses, the calculated detector responses for distances out to approximately 140 cm were in good agreement with the measured responses. However, the disagreement at the detector position of 305 cm remained. In an attempt to pinpoint the discrepancy, a second DOT/DISKTRAN calculation was performed for the 305-cm detector location beyond Configuration II.B using only seven energy groups but employing a finer mesh in the fuel-pin region. The result was a calculated 5-in. Bonner ball count rate that was only 11% higher than the measured count rate on the axis and 13% higher off the axis. It was concluded therefore that inadequate mesh spacing in the earlier calculations was the major source for the discrepancies. On the other hand, the mesh spacing appeared to be adequate for calculations through the configuration, which meant that the boundary source that was to be obtained from this calculation for the succeeding calculation for the grid-plate shield mockup should be adequate.

7.4.4. Transport Calculations for Grid-Plate Shield Configurations

The grid-plate shield configurations included in the experiment are those listed in Table 7.1 as II.D, III.B, IV.B, V.B, and VI.B. Like the fuel-pin configurations discussed in the preceding section, they differed in the spacings between the fuel subassemblies (which, of course, determined the spacings between the shield subassemblies) and in the number of fuel pins used per subassembly. These differences are noted in Table 7.4.

Table 7.4. Fuel and Subassembly Arrangements in Grid-Plate Shield Configurations

Configuration	Fuel and Shield Subassembly Spacing	Fuel Pins per Subassembly	Total Number of Fuel Pins in Configuration
II.D	1.27 cm	127	889
III.B	0.635 cm	127	889
IV.B	1.27 cm	91	637
V.B	1.27 cm	91*	630*
VI.B	0.318 cm	127	889

*Only 84 fuel pins in central assembly, seven fuel pins having been removed from the center of the subassembly to simulate a control-rod channel.

The first configuration measured in the grid-plate shield series was actually Configuration II.C, which included the iron-lined concrete of the grid-plate shield section of the mockup but did not include the seven shield subassemblies (see Fig. 7.1). The subassemblies in the fuel-pin section were spaced at 1.27 cm and contained 127 pins each. On-axis measurements were made 30.5 cm beyond the configuration to provide data check points for the analysis. The results were 5.78×10^1 and 4.11×10^3 counts/min-W for the 2-in. cadmium-covered Bonner ball and the 5-in. Bonner ball, respectively, and 1.09×10^{-2} ergs/g-h-W for the Hornyak button.

Configuration II.D was formed by inserting the shield subassemblies, which, like the fuel subassemblies, were spaced 1.27 cm apart. Bonner ball and Hornyak button measurements were made along radial traverses immediately behind the configuration and

at locations of 30.5 and 305 cm beyond the configuration. In addition, measurements were made along an axial traverse that extended from the outer edge of the fuel section through the shield section and out to 30.5 cm beyond the configuration. Similar measurements were made for the other shield mockup configurations, except that the only other configuration for which an axial traverse was performed was Configuration V.B.

The analysis of the grid-plate shield configurations was limited to Configurations II.D and IV.B, which used the same subassembly spacing but differed in fuel pin density. Calculations were performed with the two-dimensional DOT code (and DOT/DISKTRAN) for both configurations (51 groups, 150-direction quadrature) and with the three-dimensional MORSE code for Configuration II.D.

For the two-dimensional calculations, the tapered hex-shaped shields shown in Figs. 7.6 - 7.8 were converted to cylindrical stepped-shaped shields as shown in Fig. 7.13. In the model the outer liners were maintained as distinct regions, but the inner liners were homogenized with the B_4C . For the MORSE calculation a 1/12th segment of the shield region was mocked up exactly.

The source for the DOT calculations was an internal boundary source output from the appropriate fuel-pin calculations (see Section 7.3.3) at a location 64.5 cm from the end of the fuel pins. The source for the MORSE calculation was the DOT-calculated angular flux 2.5 cm from the end of the fuel pins for Configuration II.B.

A comparison of the calculated and measured Hornyak button responses along the radial traverse close behind Configuration II.D is shown in Fig. 7.14 and the corresponding comparison for the 2-in. cadmium-covered Bonner ball responses is shown in Fig. 7.15. The purpose of these traverses was to determine the extent of local peaking of the fluxes that would impinge on the grid plate, the Hornyak button largely detecting high-energy neutrons and the 2-in. cadmium-covered Bonner ball largely detecting low-energy neutrons. The lower sketch in the figures shows a portion of the end of the actual experimental configuration, and the upper sketch shows a portion of the corresponding geometry assumed in the calculation. Because the voided cylindrical annulus used in the calculational model for the outer shield subassemblies extended

Fig. 7.13. Comparison of Actual Grid-Plate Shield Geometry and a Two-Dimensional Approximation to That Geometry. (The tapered portion of the shield is first approximated by stepped hexagons and then converted to stepped cylinders.)

Fig. 7.14. Comparison of Calculated and Measured Hornyak Button Responses for a Source 1.1 cm Behind Grid-Plate Shield Configuration 11.D.

Fig. 7.15. Comparison of Calculated and Measured 2-in. Cadmium-Covered Bonner Ball Count Rate for a Radial Traverse 3.18 cm Behind Grid-Plate Shield Configuration 11.D.

uninterrupted the full 360 deg. It was narrower than the actual void in the subassemblies, and this is reflected in the narrower widths of the calculated peak fluxes about the axis of the subassemblies.

As has been characteristic of the Hornyak button responses, the calculated responses shown in Fig. 7.14 for Configuration 11.D are higher than the measured responses because of the difficulty in normalizing the Hornyak button response. Also the peak depicting the flux at the location between subassemblies is much more prominent in the measurements than in the DOT calculations. The measurements show a fast-neutron peak-to-minimum exposure ratio of 1.5 across the interassembly gap, while the DOT-calculated ratio is 1.3. This is primarily due to the DOT code overestimating the local minimum rather than underestimating the local maximum.

The MORSE calculation yielded point values for Configuration 11.D that were in nearly complete agreement with the DOT-calculated Hornyak button responses (Fig. 7.14) but were considerably lower than the DOT-calculated 2-in. cadmium-covered Bonner ball responses (Fig. 7.15). This is due to MORSE having trouble generating low-energy fluxes.

Plots of measured and DOT-calculated results for 2-in. cadmium-covered Bonner ball and Hornyak button radial traverses directly behind Configuration IV.B were essentially identical in shape with those for Configuration II.D, but were higher in magnitude. The peak 2-in. Bonner ball measurement between subassemblies, for example, was 6.2 counts/min-W for Configuration IV.B, whereas, as shown in Fig. 7.15, it was 4.5 counts/min-W for Configuration II.D. Similarly, the peak Hornyak button measurement between subassemblies was 8.01×10^{-3} ergs/g-h-W for Configuration IV.B, whereas, as shown in Fig. 7.14, it was 5.01×10^{-3} ergs/g-h-W for Configuration II.D. The difference between the two configurations is, of course, in the number of fuel pins per subassembly, Configuration IV.B having only 91 and Configuration II.D having 127. Corresponding peak Hornyak button measurements for Configurations III.B and VI.B, each of which had 127 fuel pins per subassembly but had gap widths of 0.635 cm and 0.318 cm respectively, were 3.74×10^{-3} and 3.24×10^{-3} ergs/g-h-W, indicating that the peak fast-neutron flux emerging from the gap decreases with decreasing gap width.

The peak-to-minimum fast-neutron exposure ratios across the interassembly gap are somewhat smaller for Configuration IV.B than those for Configuration II.D. The measurements give a ratio of 1.4 (compared to 1.5 quoted above for II.D) and the calculations give a ratio of 1.2 (compared to 1.3 quoted above for II.D). Configurations II.D and IV.B both have interassembly gaps of 1.27 cm, the maximum gap tested. Hornyak button radial traverses directly behind the configurations with smaller interassembly gaps showed that the fast-neutron peak-to-minimum exposure ratio decreased with decreasing gap width, the ratio being 1.3 for the 0.635-cm gap (Configuration III.B) and 1.15 for the 0.318-cm gap (Configuration VI.B).

For the Hornyak button axial traverse in Configuration II.D from the fuel section to beyond the configuration, the DOT-calculated responses were again typically higher by a factor of 2 than the measured responses. The Monte Carlo results agreed with the DOT results deep within the shield but approached the measurements near the edge of the shield and beyond the configuration. For the 2-in. cadmium-covered Bonner ball axial traverse, the DOT-calculated and measured responses were essentially in agreement over the full traverse. Again, the Monte Carlo calculations agreed with the DOT calculations (and with the measurements) out to the edge of the configuration, but beyond the edge the

Monte Carlo values fell below the measured values. While the statistics of the Monte Carlo calculation were good, it is believed that in both cases the MORSE results dropped off because neutrons failed to penetrate a sufficient distance into the shield to provide adequate sampling of the scattering distribution near the end of the shield. Thus the responses were consistently underpredicted.

Comparisons of the DOT (and DOT/DISKTRAN) calculations and the measurements at on-axis detector locations 30.5 cm and 305 cm beyond the configurations are shown in Table 7.5. As has come to be expected, the calculations overpredicted the Hornyak button responses by about a factor of 2 in all cases and they overpredicted the Bonner ball responses for the 30.5-cm location. At the 30.5-cm location, the calculated and measured Bonner ball responses are in good agreement. It is to be noted that the lower material density and increase in transmission in Configuration IV.B increases the Hornyak measurements (relative to Configuration II.D) by factors of 1.9 and 1.6 at the 30.5-cm and 305-cm locations respectively. Also, the closer fuel subassembly spacing of Configuration VI.B apparently had little effect on the measured responses (compare with Configuration III.B).

Comparing the measured values for Configurations V.B and IV.B shows that removing the seven central fuel pins from the central subassembly to simulate the control channel has an even greater effect on the fluxes emerging from the grid-plate shield section than it did on the fluxes coming from the fuel-pin section. In this case, the fast-neutron fluxes (Hornyak button responses) for the configuration with the control channel (Configuration V.B) were significantly higher at both the 30.5-cm location and the 305-cm location, by 29% and 18% respectively. On the other hand, the peak fast-neutron exposure to the grid plate in Configuration V.B (7.59×10^{-3} ergs/g-h-W), as deduced from Hornyak button radial traverses immediately behind the configuration, was actually 6% lower than the peak exposure to the grid plate in Configuration IV.B. The total fluxes (5-in. Bonner ball count rates) were higher by 9% and 5% respectively.

Overall, the study of the grid-plate shield configurations indicate that the neutron flux impinging on the grid plate can be calculated reasonably well with two-dimensional geometry, and while the extent of localized peaking was less pronounced in the calculations than in the measurements, the peak damage to the grid plate was calculated

Table 7.5. Comparison of DOT-Calculated and Measured Detector Responses on Axis 30.5 cm and 305 cm Beyond Various Grid-Plate Shield Configurations

Detector	Configuration				
	II.D (1.27,127)*	III.B (0.635,127)	IV.B (1.27,91)	V.B (1.27,91) ⁺	VI.B (0.318,127)
Response at 30.5 cm Beyond Configuration					
Bonner ball (counts/min·W)					
2-in. Cd-covered ball					
Calculated	5.20+0 [†]		9.37+0		
Measured	4.92+0	4.75+0	9.91+0	1.23+1	4.66+0
Calc./Meas.	1.06		0.95		
5-in. ball					
Calculated	1.12+3		1.73+3		
Measured	9.76+2	8.88+2	1.52+3	1.65+3	8.58+2
Calc./Meas.	1.15		1.14		
Hornyak button (ergs/g·h·W)					
Calculated	5.94-3		1.08-2		
Measured	2.91-3	2.44-3	5.40-3	6.95-3	2.26-3
Calc./Meas.	2.04		2.00		
Response at 305 cm Beyond Configuration					
Bonner ball (counts/min·W)					
2-in. Cd-covered ball					
Calculated	1.66+0		2.70+0		
Measured	1.22+0	9.36-1	2.14+0	2.30-0	8.20-1
Calc./Meas.	1.36		1.26		
5-in. ball					
Calculated	1.87-2		2.65+2		
Measured	1.18+2	9.51+1	1.77+2	1.86+2	8.87+1
Calc./Meas.	1.58		1.50		
Hornyak button (ergs/g·h·W)					
Calculated	1.25-3		1.99-3		
Measured	5.94-4	4.32-4	9.58-4	1.21-3	4.17-4
Calc./Meas.	2.10		2.08		

* (1.27,127) indicates 1.27 cm between fuel subassemblies, 127 fuel pins per subassembly.

⁺ Only 84 fuel pins in central subassembly.

[†] Read: 5.20×10^0 .

with good precision. At large distances, however, the DOT-calculated results are persistently higher than the measured results. As was the case for the fuel-pin configurations, the overprediction is believed to be attributable to the fact that the spatial mesh used in the calculations was too coarse.

7.4.5. Transport Calculations for Prototypic Grid-Plate Configuration

The full grid-plate configuration, which included the fuel-blanket section, the grid-plate shield section, and the grid-plate section, was identified as Configuration 11.F and is the configuration depicted in Fig. 7.1. The fuel section of this configuration consisted of the seven subassemblies that contained 127 fuel pins each and were spaced 1.27 cm apart (see Section 7.4.3), and the grid-plate shield section consisted of the seven B_4C -filled subassemblies at the same spacing. Details of the grid-plate section are shown in Fig. 7.16. As noted earlier, the 63.5-cm thickness selected for the grid plate included an extra 2.5 cm of iron for reasons noted in Section 7.2. The B_4C inserts shown in Figs. 7.1 and 7.16 represent extensions of the shield subassemblies into the grid plate.

The measurements for the full grid-plate configuration depicted in Fig. 7.1 included a Hornyak button radial traverse directly behind the grid-plate configuration (for a distance of 2.62 cm) and NE-213 and hydrogen counter spectral measurements as close behind the configuration as was feasible (at 34.3 cm). In addition, radial traverses were made with the Hornyak button and the 2-in. cadmium-covered and 5-in. Bonner balls at distances of 305 cm and 30.5 cm. Finally, on-axis measurements were made at the 305-cm location with the 3-, 6-, 8-, and 10-in. Bonner balls and at the 34.3-cm location with the 3-in. Bonner ball. At some of these locations, additional measurements were made with the B_4C inserts removed.

The DOT calculations of the full grid-plate configuration were performed with the 51-group P_3 cross sections and the 150-direction biased quadrature set used in the earlier calculations; however, in this case the calculational model included the entire preceding section of the configuration (that is, the grid-plate shield section) and the internal boundary source (output from the calculation for Configuration 11.D) was located

Fig. 7.16. Details of the Prototypic Grid Plate and Cylindrical Grid-Plate Shield Inserts.

2.54 cm from the end of the fuel section. Like the shield region, the grid-plate region was converted to equivalent cylindrical annuli, with the void regions matching in the two sections.

The MORSE calculations again used a 1/12th segment of the geometry and the same boundary source as the DOT calculations.

The measured and calculated Hornyak button responses along a radial traverse directly behind Configuration II.F are shown in Fig. 7.17, where again the lower sketch depicts the actual geometry of the grid plate (the central hole and one of the six surrounding outer holes) and the upper sketch shows the geometry used in the calculational model.

From Fig. 7.17 it is seen that the DOT calculations overpredicted the Hornyak button measurements by a factor of about 3 and the MORSE calculations overpredicted the measurements by factors of 1.7 to 2; however, the shapes of both calculated curves are in

Fig. 7.17. Comparison of Calculated and Measured Hornyak Button Responses Along a Radial Traverse 2.62 cm Behind Grid-Plate Shield Configuration II.F (with B_4C Inserts).

good agreement with the shape of the measured curve. The measurements indicate that the peak fast-neutron exposures to components in line with the inner and outer grid-plate holes would be 9.34×10^{-4} and 1.01×10^{-3} ergs/g.h.W respectively, the measured value dipping to 2.19×10^{-4} ergs/g.h.W behind the steel between the voids and plunging rapidly as the radial distance from the outer void increases. Without the B_4C inserts, the measurements are somewhat higher (see Table 7.6).

At distances of 30.5 and 305 cm beyond the configuration, DOT and MORSE also overpredicted the Hornyak button responses along radial traverses, DOT by factors of 3 and 2.7 respectively and MORSE by factors of 1.5 to 2 and 1.7 respectively. The measured peak exposures at these distances are included in Table 7.6.

As has been noted in earlier sections, the general pattern of the calculations overpredicting the Hornyak button responses throughout the grid-plate shield study appear to be due to the difficulty encountered in normalizing the Hornyak button response

Table 7.6. Peak Fast-Neutron Exposures (Hornyak Button Responses)
Beyond Grid-Plate Configuration II.F

Distance Beyond Configuration (cm)	Peak Measured Response (ergs/g-h-W)		Calc./Meas.	
	Opposite Inner Void	Opposite Outer Void	DOT	MORSE
2.62 (with B ₄ C Inserts)	9.34-4	1.01-3	~3	1.7 - 2
2.62 (w/o B ₄ C Inserts)	1.07-3	1.14-3		
30.5 (with B ₄ C Inserts)	5.96-4	6.67-4	~3	1.5 - 2
30.5 (w/o B ₄ C Inserts)	6.51-4	7.31-4		
305 (with B ₄ C Inserts)	2.78-4	2.50-4	~2.7	~1.7

function. But even with the normalization factor taken into account, the DOT results still appear to be high. For example, if it could be assumed that the MORSE curve shown in Fig. 7.17 were correct except for the normalization factor, and thus that the MORSE curve could be normalized to the measurements and the DOT curve lowered accordingly, the DOT curve would still be high by a factor of 1.5 to 1.75.

In the case of the 2-in. cadmium-covered Bonner ball, the DOT results are in good agreement with the measured results for the radial traverse at 30.5 cm, while the MORSE results are low except for the point directly in line with the outer grid-plate hole, where both calculated values and the measured value are all close. At 305 cm the MORSE-calculated points lie on the measured curve, while the DOT-calculated curve is higher than the measured curve by a factor of 2 behind the grid-plate holes. At larger radial distances, the DOT results are in somewhat better agreement with the measured values.

For the 5-in. Bonner ball, the DOT results for the 30.5-cm radial traverse are higher than the measured results by a factor of 1.25 behind the central hole and by a factor of 1.55 behind the outer hole. Again the MORSE results for the 30.5-cm traverse are too low except for the one point in line with the outer grid-plate hole, where the MORSE and measured points are in agreement.

At 305 cm the DOT results for the 5-in. Bonner ball remain a factor of 2 higher than the measured results behind the grid-plate holes with somewhat better agreement at radial distances beyond the holes (see Fig. 7.18). As was the case for the 2-in. Bonner ball at this distance, the MORSE-calculated points for the 5-in. ball essentially lie on the measured curve, this time with such good statistics that the error bars were smaller than the plotting symbols and thus could not be shown.

The curves for the radial traverse measurements at 305 cm are similar in shape for all three detectors in that there is a relatively flat region behind the grid-plate holes, a rapid dropoff from 30 to 80 cm, and a flattening beyond 80 cm. The curves for the DOT calculations have a similar shape except that the flattening begins beyond 100 cm.

The neutron spectral measurements made on the axis at a distance of 34.29 cm beyond the grid plate are compared with DOT-calculated and MORSE-calculated spectra in Figs. 7.19 and 7.20 respectively. While the the DOT-calculated spectrum and the measured spectrum are very similar in shape, the calculated integral flux is about 50% higher than the measured integral flux. Conversely, the MORSE-calculated spectrum and the measured spectrum do not agree too well in shape; however, above 0.9 MeV the integrated fluxes are relatively close, the MORSE-calculated flux being 0.89 ± 0.13 of the measured flux.

On-axis measurements at the 34.29-cm location were also made with the 3-in. Bonner ball, and at the 305-cm location with several Bonner balls. These results are compared with the corresponding DOT- and MORSE-calculated values in Table 7.7. It is noted that the DOT-calculated count rate for the 3-in. Bonner ball at 34.29 cm is higher than the measured count rate by a factor of 1.4, and, as pointed out above, the DOT-calculated fast-neutron flux at this location is high by about a factor of 1.5, while the MORSE-calculated fast-neutron flux is low.

At the 305-cm location, all the DOT-calculated Bonner ball count rates (covering most of the neutron spectrum) are about a factor of 2 higher than the measured count rates. These results indicate that the angular flux collimated through the grid plate may be overpredicted by a factor of 2 since it would be the collimated component that would make the largest contribution at large distances behind the grid plate. As has

Fig. 7.18. Comparison of Calculated and Measured 5-in. Bonner Ball Count Rates Along a Radial Traverse 305 cm Behind Grid-Plate Configuration II.F.

Fig. 7.19. Comparison of DOT-Calculated and Measured Neutron Spectra 34.3 cm Beyond Grid-Plate Configuration II.F.

Fig. 7.20. Comparison of MORSE-Calculated and Measured Neutron Spectra 34.3 cm Beyond Grid-Plate Configuration 11.F.

Table 7.7. Comparison of Calculated and Measured Detector Responses on Axis 34.29 and 305 cm Beyond Grid-Plate Configuration 11.F

Type of Response	Measured Response	Calculated Response		Calc./Meas.	
		DOT	MORSE	DOT	MORSE
Response at 34.3 cm Beyond Configuration					
Integrated spectrometer fluxes (neutrons/cm ² ·min·W)					
E > 0.9 MeV	26.61	40.99	23.62±3.42	1.54	0.89±0.13
E > 0.12 MeV	99.22	146.4	72.0±4.38	1.48	0.73±0.04
Bonner ball (counts/min·W)					
3-in. ball	42.6	59.0	—	1.4	—
Response at 305 cm Beyond Configuration					
Bonner ball (counts/min·W)					
2-in. ball (Cd-covered)	0.493	1.08	0.462±0.074	2.1	0.94±0.15
3-in. ball	12.3	25.6	—	2.1	—
5-in. ball	55.1	116	62.6±6.89	2.1	1.14±0.12
6-in. ball	62.1	127	—	2.0	—
8-in. ball	50.8	101	—	2.0	—
10-in. ball	29.6	62.6	—	2.1	—
12-in. ball	16.8	36.4	—	2.2	—

been pointed out in earlier sections, these overpredictions at large distances are thought, at least in part, to be due to the use of an inadequate spatial mesh in the calculations.

7.5. SUMMARY

When the TSF-GCFR grid-plate shield confirmation experiment was begun, one of the major concerns was whether the streaming of neutrons through a fuel lattice simulating a section of the GCFR core could be accurately predicted and, further, whether changes in the spacing between fuel subassemblies and/or fuel pins would markedly affect the neutron transmission from the core through the grid plate. Another major concern, of course, was whether the GCFR grid plate, and also components "above" the grid plate, could be adequately shielded from radiation damage without overly restricting the reactor coolant flow. In an attempt to answer these questions, the experiment was designed so that neutrons from the Tower Shielding Reactor II would induce fissions in a fuel lattice mocking up seven GCFR-type fuel-pin assemblies and the lattice would be the dominant source in the experiment. The experimental procedure then called for detailed measurements (and calculations) to be made for various sections of the configuration as they were added behind the fuel lattice. Not only did this allow the adequacy of the available calculational methods to be tested for successive regions, it also made it possible to measure (and calculate) the effect of variations within the individual regions.

As has been described in earlier sections, the measurements were performed with a combination of detectors. A Hornyak button responded primarily to fast neutrons, a 2-in. cadmium-covered Bonner ball to low-energy neutrons, and a 5-in. Bonner ball to the total flux. Neutron spectra were obtained with an NE-213 spectrometer that covered neutron energies from 0.8 to 15 MeV and a hydrogen counter that covered energies from 0.08 to 1.5 MeV. In addition, measurements with a series of Bonner balls of different diameters essentially covered the full spectrum.

*When this study was performed the GCFR design specified a top-mounted grid plate, above which control-rod mechanisms were to operate in an upper plenum. Subsequently, the design was revised to have a bottom-mounted grid plate.

Most of the calculations were performed with the DOT discrete ordinates code using two-dimensional geometric models simulating the experimental configuration. In order to include streaming paths, a heterogeneous model was employed which consisted of concentric annuli representing low-density and high-density regions as required to simulate the region being calculated. Check points were calculated with the MORSE Monte Carlo code using "exact" three-dimensional models.

The experimental and calculational results are summarized below.

7.5.1. Fuel-Pin Configurations

In investigating the neutron streaming in GCFR-type fuel lattices, it was important to include variations in the spacing between fuel subassemblies that would bracket those being considered for the GCFR and also to include a range of fuel densities within the subassemblies. Thus, the experiment covered fuel subassembly spacings from 0.318 cm to 1.27 cm and included two types of subassemblies, one containing 127 fuel pins and another containing 91 fuel pins. The calculated-to-measured ratios for locations beyond fuel-pin configurations with various combinations of these variables are summarized in Tables 7.8 and 7.9.

The results presented in Table 7.8 show that for the Bonner balls and the neutron spectrometers the calculations and measurements are in good agreement over a distance extending to 140 cm beyond the configurations. At a distance of 305 cm, however, the calculations consistently overpredict the Bonner ball responses, and at all locations the calculations overpredict the Hornyak button responses. On the basis of these and similar Hornyak button results for subsequent configurations, it was concluded that the difficulty with calculating the Hornyak button response was a normalization problem with the detector's response function. And from a series of additional calculations for Configuration 11.B it was concluded that the other DOT-calculated responses for the 305-cm location would approach the measured responses if a more detailed spatial mesh were used in the calculation. Thus, it appears that the transmission through the GCFR core can be calculated.

Table 7.8. Summary of Calculated-to-Measured Detector Responses
Beyond Various Fuel-Pin Configurations

Type of Response	Calc./Meas.			
	II.B (1.27,127)*	III.A (0.635,127)	IV.A (1.27,9†)	VI.A (0.318,127)
30.5 cm Beyond Configuration				
2-in. Cd-covered Bonner ball				
DOT/Meas.	0.94	0.98	0.82	0.97
MORSE/Meas.	~1.05			
5-in. Bonner ball				
DOT/Meas.	1.0	1.05	0.94	1.03
MORSE/Meas.	~1.04			
Hornyak button				
DOT/Meas.	1.4	1.61	1.35	1.70
MORSE/Meas.	~1.2			
138.1 cm Beyond Configuration				
3-in. Bonner ball, DOT/Meas.	1.27			
6-in. Bonner ball, DOT/Meas.	1.16			
10-in. Bonner ball, DOT/Meas.	1.08			
Neutron spectrum, DOT/Meas.	Agreement in shape and magnitude for II.B			
305 cm Beyond Configuration				
2-in. Cd-covered Bonner ball				
DOT/Meas.	1.46	1.95	1.22	1.79
MORSE/Meas.	1.22			
5-in. Bonner ball				
DOT/Meas.†	1.34	1.57	1.26	1.62
DOT/Meas.	1.11			
MORSE/Meas.	1.21			
Hornyak button				
DOT/Meas.	1.71	2.19	1.71	2.29
MORSE/Meas.	1.27			

*(1.27,127) indicates 1.27 cm between fuel subassemblies, 127 fuel pins per subassembly.

†DOT calculation repeated with finer spatial mesh (see Section 7.4.3).

Table 7.9. Summary of Calculated and Measured Effects of Changes in Fuel Subassembly Spacing and in Fuel Density (Fuel-Pin Configurations)

Type of Response	Distance (on Axis) Beyond Configuration (cm)	Subassembly Spacing Effect, II.B/VI.A (1.27/0.318) ^a	Fuel Density Effect, IV.A/II.B (91/127) ^b	Control Channel Effect, V.A/IV.A (W/WO) ^c
5-in. Bonner ball Calc.	30.5	1.05	1.38	
	305	0.95	1.44	
Meas.	30.5	1.08	1.47	1.03
	305	1.16	1.5	1.04
Hornyak button Calc.	30.5	1.05	1.53	
	305	0.95	1.6	
Meas.	30.5	1.28	1.59	0.89
	305	1.24	1.64	1.01

^aRatio of response for configuration II.B with a 1.27-cm spacing between fuel subassemblies to response for configuration VI.A with a 0.318-cm spacing between subassemblies.

^b91 fuel pins in subassembly versus 127 fuel pins in subassembly.

^cConfiguration with control channel in central subassembly versus configuration without control channel.

With respect to variations within the fuel lattice, both the calculated and the measured 5-in. Bonner ball responses in Table 7.9 indicate only a small effect on the neutron streaming from the fuel lattice when the spacing between fuel subassemblies is increased from 0.318 cm to 1.27 cm. The measured responses reflect the larger impact, but only an 8% increase at 30.5 cm and a 16% increase at 305 cm. The Hornyak button responses, however, show an increase in the fast-neutron flux of 28% at a distance of 30.5 cm and 24% at a distance of 305 cm. Thus, varying the gap size probably would not have a large overall effect on the neutron leakage from the GCFR axial blanket but it could affect the fraction of the leakage comprised of fast neutrons.

The fuel-pin spacing (i.e., fuel density) appears to have a greater neutron transmission effect than the subassembly spacing, as is indicated by the IV.A/II.B ratios. At both axial locations the Bonner ball responses (both calculated and measured) show a 40% to 50% increase in the total flux when the fuel-pin spacing is increased by reducing the number of pins within the seven assemblies from 127 to 91. The Hornyak button responses indicate slightly larger increases in the fast-neutron fluxes — about 60% at both locations.

Comparing Configurations IV.A and II.B with each other, however, does not really show how much the streaming in the individual lattices is enhanced by the presence of the low-density (helium) passageways. This was investigated by performing additional DOT calculations with a homogeneous cylindrical model for these two configurations. When the results were compared with the corresponding values obtained with the heterogeneous model (i.e., with the cylindrical-annuli model used for this study), the streaming effect for a location immediately behind the configurations was about 9% for Configuration II.B (with 127 pins per subassembly) and about 11% for Configuration IV.A (with 91 pins per subassembly). Thus the streaming in both configurations was enhanced approximately 10% by the helium passageways.

Underpredictions of only 10% in the neutron streaming effect would not be serious in GCFR-type calculations and use of a homogeneous model to simplify design calculations would be attractive. However, for distances beyond the experimental configurations the homogeneous and heterogeneous calculations gave widely divergent results. The homogeneous calculations underpredicted the Bonner ball count rates at a distance of 305 cm by 20% to 30% for Configuration II.B and by about 40% for Configuration IV.A, whereas the heterogeneous calculations overpredicted the same count rates by 35% to 50% for Configuration II.B and about 25% for Configuration IV.A. Thus for conservatism, and especially when the neutron penetration through subsequent regions is to be calculated, the heterogeneous model should be retained.

Finally, the measured results in Table 7.9 (no calculations performed for Configuration V.A) indicate that removing seven fuel pins from the central fuel subassembly had little effect on either the total flux or the fast-neutron flux at either axial location.

7.5.2. Grid-Plate Shield Configurations

Studies of the grid-plate shield configurations focussed on two questions: Could the region of the grid plate that would receive the peak exposure to fast neutrons be identified and could that exposure be calculated? And could the neutron fluxes impinging on the grid-plate region be accurately predicted?

It was shown in Section 7.4.4 by both the Hornyak button measurements directly behind the grid-plate shield and the corresponding calculations that the peak fast-neutron exposure to the grid plate would occur on the "bottom" of the grid plate at points midway between the fuel-shield subassemblies. The peak occurred at this location regardless of the spacing that was used between subassemblies; however, both the peak exposure and the ratio of the peak exposure to the minimum exposure within the spacing gap varied with the spacing and with the fuel density within the subassemblies. Table 7.10 shows, for example, that when the fuel density was maintained at 127 pins per subassembly, the peak exposure increased with an increasing gap width. With the results from Configuration VI.B used as a base, the ratio r of the peak exposure for a configuration with a gap width d (in centimeters) to that for Configuration VI.B was shown to fit the equation

$$r = 0.8833 + 0.3150d + 0.1653d^2$$

for gap widths between 0.3175 and 1.27 cm.

With the 91-pin subassemblies having only one spacing in the experiment (1.27 cm), data were not available to test whether the equation would hold for other fuel densities. It is apparent, however, that decreasing the number of pins within the subassemblies from 127 to 91 substantially increased the peak exposure, although removing an additional 7 pins from the central fuel element had little further effect. The increase in the peak exposure with the 91-pin subassemblies (by a factor of 1.6) compared to that for the 127-pin subassemblies was primarily due to the reduced fuel density in the core of Configuration IV.B, since, as noted in Section 7.5.1 above, the streaming effects for the two fuel-pin configurations were approximately equal.

The measured fast-neutron peak-to-minimum exposure ratios for the various grid-plate shield configurations vary from 1.15 (for Configuration VI.B') to 1.5 (for Configuration II.D) (see Table 7.10). For the two cases for which calculations were performed, the calculated peak exposures (Hornyak button responses) are typically higher than the measured peak exposures, but the calculated peak-to-minimum ratios are lower than the measured ratios, which indicates that the calculations give a flatter distribution of the flux across the gap. Since the Hornyak button response function available for the calculations had been judged to not be correct, it was not possible to estimate how well

Table 7.10. Fast-Neutron Peak-to-Minimum Exposure Ratios
Across Gap Between Shield Subassemblies^a

Configuration	Relative Peak Exposure	Peak-to-Minimum Exposure Ratio Across Gap
Configuration II.D (1.27,127) ^b	1.55	1.5 (meas.) 1.3 (calc.)
Configuration III.B (0.635,127)	1.15	1.3 (meas.)
Configuration IV.B (1.27,91)	2.47	1.4 (meas.) 1.2 (calc.)
Configuration V.B (1.27,91) ^c	2.34	—
Configuration VI.B (0.318,127)	1.0	1.15 (meas.)

^aBased on Hornyak button measurements and corresponding calculations directly behind grid-plate shield.

^b(1.27,127) Indicates 1.27 cm between fuel (and shield) subassemblies and 127 fuel pins per subassembly.

^cOnly 84 fuel pins in central assembly.

the calculations were predicting the peak exposure. However, if it could be assumed that the DOT calculations of the fast flux immediately behind the configuration were as accurate as the DOT calculations of the total flux 30.5 cm beyond the configuration, then the on-axis calculated-to-measured ratio of the 5-in. Bonner ball responses could be used to normalize the calculated Hornyak button curve. As shown in Table 7.11, this ratio is 1.15.

Normalizing the calculated Hornyak button response on the axis behind Configuration II.D to 1.15 times the measured response resulted in a calculated peak exposure in the inter-assembly gap that is 0.98 times the measured value. Similar adjustment of the calculated response for Configuration IV.B gave a calculated peak exposure that was 1.0 times the measured value. Thus, if the normalization is correct, the calculations are in excellent agreement with the measurements.

All the above results were, of course, obtained in the absence of the grid plate itself, and no suitable measurement scheme was devised to measure behind the grid-plate shield with the grid plate in position. However, calculations for the grid-plate

Table 7.11. Summary of DCT-Calculated-to-Measured Detector Responses Beyond Grid-Plate Shield Configurations II.D and IV.B

Type of Response	Calc./Meas.	
	II.D (1.27,127)*	IV.B (1.27,91)
30.5 cm Beyond Configuration		
2-in. Cd-covered Bonner ball	1.06	0.95
5-in. Bonner ball	1.15	1.14
Hornyak button	2.04	2.00
305 cm Beyond Configuration		
2-in. Cd-covered Bonner ball	1.36	1.26
5-in. Bonner ball	1.58	1.50
Hornyak button	2.10	2.08

* (1.27,127) indicates 1.27 cm between fuel (and shield) subassemblies and 127 fuel pins per subassembly.

configuration (see Section 7.5.3 below) showed that with the grid plate present the peak-to-minimum exposure ratio for Configuration II.D would be 1.12 rather than 1.3. If a similar reduction in the measured ratio could be assumed, then the measured peak-to-minimum ratio would be 1.3 rather than 1.5, which is certainly an acceptable differential.

It is to be remembered, however, that all these calculations were performed with a 150-angle biased quadrature set that may be impractical for GCFR design calculations. Whether such a detailed quadrature set would be necessary for calculating the grid-plate exposure was investigated by performing an additional calculation for Configuration II.F with a symmetric S_{10} (70-angle) quadrature. The result was that the location of the peak exposure moved toward the edges of the holes in the grid plate and also the peak exposure was somewhat lower, but not significantly lower. Thus it appears that design calculations using the lower-order quadrature would not seriously underpredict the peak exposure to the grid plate.

Finally, it can be deduced from the 2-in. and 5-in. Bonner ball results shown in Table 7.11 that the low-energy and total neutron fluxes calculated to be impinging on the grid plate are approximately correct. However, the high-energy fraction of the flux, as deduced from the Hornyak button results, is overestimated, as has consistently been the case throughout this study. As has also consistently been the case, all detector responses for a location 305 cm beyond the configuration are overpredicted, which, while not directly affecting the calculation of neutron transport through the grid plate, portends problems with predicting the fluxes beyond the grid plate.

7.5.3. Grid-Plate Configuration

In the TSF experiment, the region beyond the grid plate corresponded to the upper plenum in the original GCFR design. Since much of the upper plenum would consist of low-density helium, a demonstrated capability for calculating the neutron fluxes in the air beyond the grid-plate section of the experimental configuration would indicate that the plenum fluxes could also be calculated, which, in turn, would imply that the exposure of important components within the plenum could be determined.

The primary problem with calculating the neutron transmission through the grid plate was due to the large coolant holes through it. Not only was it important that these holes be adequately modeled for the DOT calculations, it was also important that the angular quadrature be sufficiently detailed to properly calculate the energy and angular distributions of the emerging neutrons, some of which would be passing uncollided through the grid-plate holes while others would be traversing through the full 30.5-in. thickness of steel or bouncing between the two mediums. As was the case for the earlier sections of the experimental configuration, the holes in the grid plate were modeled as low-density cylindrical annuli interspersed between high-density annuli, and the first calculations were performed with the 150-angle biased quadrature.

The resulting comparisons of the measurements and calculations are summarized in Table 7.12. As would be expected, for the locations close behind the configuration (< 34.3 cm), the measured and calculated responses reflected large peaks in the fluxes at detector positions directly in line with the grid-plate holes, while for the 305-cm

Table 7.12. Summary of DOT-Calculated-to-Measured Detector Responses Beyond Grid-Plate Configuration 11.^a

Response Type	Calc./Meas.
2.62 cm Beyond Configuration	
Hornyak button	$\sim 3 - \sim 3.2^a$
30.5 cm Beyond Configuration (on Axis)	
Hornyak button	$\sim 3 - \sim 3.3^a$
2-in. Cd-covered Bonner ball	$\sim 1.1 - \sim 1.6^b$
5-in. Bonner ball	$\sim 1.2 - \sim 1.6^b$
34.3 cm Beyond Configuration (on Axis)	
Integrated spectrometer fluxes ^b	
E > 0.9 MeV	1.54
E > 0.12 MeV	1.48
3-in. Bonner ball	1.4
305 cm Beyond Configuration	
Hornyak button	~ 2.7
All Bonner balls ^c	2.0 - 2.2

^aFirst number compares responses in line with central grid-plate hole; second number compares responses in line with outer grid-plate hole; in all cases the measured and calculated curves for these close-in positions agreed in shape.

^bMeasured and calculated spectra agreed in shape.

^cTo cover the energy spectrum, measurements were made with the following Bonner balls: 2-in. Cd-covered, 3-in., 5-in., 6-in., 8-in., 10-in., and 12-in.

location the detectors viewed the grid plate as a point source. Thus for the close-in positions, ratios are given for locations corresponding to both holes but for the 305-cm location only one ratio is given.

Table 7.12 shows that the calculations overpredict the close-in fast-neutron fluxes (i.e., Hornyak button responses) by factors of approximately 3 and the distant fluxes by a factor not much less than 3. With the exception of the on-axis close-in measurements for the 2-in. cadmium-covered Bonner ball and the 5-in. Bonner ball, all other detector responses were also overestimated - by factors varying from 1.4 to 2.2. On the other

hand, the shapes of all the corresponding measured and calculated curves were in good agreement, and the curves approached each other as the radial distance from the axis increased.

The above comparisons indicate that the calculated angular fluxes in the most forward directions were too high. They also indicate that GCFR design calculations using the same biased quadrature set would give extremely conservative results. But most design calculations had used a symmetric S_{10} quadrature set which had only 70 angles and some had even used an S_6 set (30 angles). As a result, the adequacy of the design calculations could not be inferred from the calculation for the experiment.

The above being the case, one of the design calculations was recalculated with the 150-angle biased quadrature set. For a location at the top center of a grid-plate hole, the resulting total flux was a factor of 1.4 higher than the total flux calculated with the S_{10} quadrature set and the $E > 1.0$ MeV fluxes and the $E > 0.1$ MeV fluxes were higher by factors of 2.2 and 1.5 respectively. Thus the results from the biased-quadrature GCFR design calculation were higher than the results from the S_{10} quadrature GCFR design calculation by approximately the same factors that the results from the biased-quadrature calculation for the experiment were higher than the measured results. This seemed to imply that an S_{10} quadrature might be more suitable for the calculation of the experiment.

A recalculation of the experiment showed that the S_{10} quadrature was totally inadequate, however, the S_{10} quadrature fluxes at the top of the grid plate being lower than the biased-quadrature fluxes by factors of 3 to 7. At 34.3 cm beyond the grid plate the detector responses were underpredicted by factors of 4 to 12, and at 305 cm they were underpredicted by factors of 35 to 851. But at 305 cm the 5-in. Bunker ball count rate 122 cm off the axis was only 1.2 times higher than the measured value.

Since the differences between the calculated results obtained with the two quadratures sets in the GCFR design calculations were not nearly so large as those between the calculations for the experiment, it could be expected that the design calculations would not underpredict the fluxes within the plenum by such large factors. For one thing, the neutron reflections from the upper shields and the PCRV surrounding

the plenum would create a more uniform flux field in the upper plenum than would exist in the open space behind the grid plate configuration.

In summary, it is conceivable that the design calculations underpredicted the inlet plenum fluxes. Based on the above and other considerations, it is estimated that the S_{10} quadrature design calculations may have underestimated the $E > 1.0$ MeV fluxes by a factor of approximately 4 and the $E > 0.1$ MeV fluxes and the total neutron fluxes by a factor of approximately 2.5. These factors are perceived as upper bounds to the underpredictions, and most design constraints for the upper axial region were sufficiently higher than previously calculated design neutron responses that errors of this magnitude could be allowed.

7.6. CONCLUSIONS

The results of the GCFR grid-plate shield confirmation experiment have shown that axial neutron streaming in GCFR-type cores can be calculated for a variety of fuel-pin and subassembly spacings and also that the position and magnitude of the peak fast-neutron exposure to the grid plate can be calculated. However, current methods are inadequate for calculating the neutron leakage from the top of the grid plate into the upper plenum. Since the results of the plenum calculation were sensitive to the quadrature set used, emphasis should be given to the development of an appropriate quadrature set for plenum-type calculations.

7.7. REFERENCES

1. D. T. Ingersoll, F. J. Muckenthaler, C. O. Slater, and M. L. Williams, "Grid Plate Shield Design Confirmation Experiment," p. 109 ff. In Gas-Cooled Reactor Program Annual Progress Report for Period Ending December 31, 1977, ORNL-5426 (August, 1978).
2. F. J. Muckenthaler, J. L. Hull, and J. J. Manning, "The GCFR Grid Plate Shield Design Confirmation Experiment," ORNL/TM-6580 (January, 1979).
3. C. O. Slater, S. M. Cramer, D. T. Ingersoll, M. L. Williams, F. J. Muckenthaler, J. J. Manning, and J. L. Hull, "Measurement and Calculation of the Effectiveness of the Gas-Cooled Fast Breeder Reactor Grid-Plate Shield," Nuclear Technology 52, 354 (1981).

4. C. O. Slater, S. N. Cramer, and D. T. Ingersoll, "Analysis of the ORNL/TSF GCFR Grid Plate Shield Design Confirmation Experiment," Trans. Am. Nucl. Soc. 32, 641, 1979.
5. C. O. Slater, S. N. Cramer, and D. T. Ingersoll, "Analysis of the ORNL/TSF GCFR Grid-Plate Shield Design Confirmation Experiment," ORNL-5551 (August, 1979).
6. R. E. Maerker and F. J. Muckenthaler, "The Absolute Neutron Spectrum Emerging Through the Large Beam Collimator from the TSR-II Reactor at the Tower Shielding Facility," ORNL-TM-5183 (January 1976).

8. GCFR RADIAL BLANKET AND SHIELD INTEGRAL EXPERIMENT

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- 8.1. Introduction**
- 8.2. The Experiment**
- 8.3. Preeanalysis and Experimental Design**
- 8.4. Postanalysis of the Experiment; Comparison of Calculated and Measured Results**
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8.1. INTRODUCTION

Whereas the experiments described in Chapters 5 through 7 were designed to study the effects of neutrons streaming upward through vertical helium-coolant passageways in the GCFR core, out the top of the core, and on through large holes in the grid-plate shield,* the experiment described in this chapter was designed to study the effects of neutrons leaving the core in the radial direction and penetrating through thick and dense laminated regions. More specifically, the experiment was designed to verify the effectiveness of the GCFR radial shield in protecting the prestressed concrete reactor vessel (PCRV) from radiation damage.

The design of the GCFR radial shield had been specified in considerable detail and designated as Conceptual Shielding Configuration 1 (CSC.1). As shown in Fig. 8.1 and Table 8.1, the shielding consisted of a removable inner shield (in two rows) and a permanent outer shield. The first row of the inner shield consisted of boronated graphite clad with stainless steel and the second row consisted entirely of stainless steel. The permanent outer shield consisted of graphite bounded on each side by boronated graphite and clad with stainless steel. These shields were to limit the total lifetime fluence at any point in the PCRV to less than 10^{19} neutrons per square centimeter, and GCFR design calculations at General Atomic and at ORNL had indicated that this criterion could be met if the first row of the inner shield were replaced every three years and the second row every six years. However, a number of uncertainties existed in the design and it was important that the range of the uncertainties be bracketed -- and narrowed if possible -- in order to reduce costs resulting from excessive design conservatism.

The uncertainties of most concern were those associated with the neutron cross sections used in the design calculations. While earlier TSF experiments had provided data to test the cross sections needed to calculate deep neutron penetration in iron and stainless steel (refs. 1 and 2), the neutron data required for the graphite and boronated graphite regions had not been similarly checked. Moreover, large uncertainties were associated with the neutron cross sections used in calculating the transport through the

*At the time this experiment was planned, the GCFR demonstration plant design still called for a down-flow coolant system and a top-mounted grid plate.

LEGEND

- 304 OR 316 SS
- 304 OR 316 SS
- H_2O
- $C-B_4C$
- U
- C
- SiO_2

Fig. 8.1. Schematic of Reference Downflow GCFR Shield Des.gn: Conceptual Shielding Configuration 1 (CSC.1).

Table 8.1. Geometry and Materials for Conceptual Shielding Configuration 1 (CSC.1) at the Core Midplane Level

Region and Material	Inner Radius (cm)	Outer Radius (cm)	Thickness (cm)
Core: Enrichment Zone 1	0.0	77.9	77.9
Enrichment Zone 2	77.9	96.9	19.0
Enrichment Zone 3	96.9	115.6	18.7
Radial Blanket: Row 1, UO_2	115.6	134.1	18.5
Row 2, UO_2	134.1	152.6	18.5
Row 3, UO_2	152.6	171.0	18.4
Helium Gap	171.0	194.0	23.0
Inner Shield 1: SS-316	194.0	195.0	1.0
C + B ₄ C*	195.0	207.7	12.7
SS-316	207.7	209.0	1.3
Helium Gap	209.0	214.1	5.1
Inner Shield 2: SS-316	214.1	229.1	15.0
Helium Gap	229.1	234.2	5.1
Outer Shield: SS-316	234.2	239.3	5.1
C + B ₄ C*	239.3	244.4	5.1
Graphite	244.4	267.0	22.6
C + B ₄ C*	267.0	272.1	5.1
SS-316	272.1	277.2	5.1
Helium Gap	277.2	312.9	35.7
PCRV Liner: Fe	312.9	314.7	1.8
PCRV	314.7		

*25 wt.% boron mixture of graphite and B₄C at 1.6 g/cc bulk density.

thorium blanket region, which, of course, introduced uncertainties in the source calculated to be impinging on the inner surface of the radial shield. Finally, the uncertainties in the neutron cross sections introduced uncertainties in calculations of the production of secondary gamma rays and the associated energy deposition throughout the configuration.

A major uncertainty also lay in the CSC.1 design itself, since even as the experiment was being planned, modifications to the design were under consideration. In particular, an "upflow GCFR" was being proposed which would necessitate changes in the shield and the surrounding region. Also, the use of a ThO₂ blanket rather than a UO₂ blanket was being suggested, which, of course, would affect the source impinging on the

shield. In the absence of an alternate design, however, the CSC.1 design was retained as the basis for the experimental design, with the objectives of the experiment stated as follows:

- (1) To verify the accuracy of the nuclear data used in the design calculations,
- (2) To define experimental and analytic uncertainties in terms of their implications to design calculations and design criteria, and
- (3) To provide experimental comparisons of the shielding effectiveness of alternate shield designs.

As was the case with the grid-plate shield confirmation experiment, the design of the experimental configuration was completed only after a detailed preanalysis had indicated that the configuration would meet its objectives. The preanalysis calculations are summarized here in Section 8.3, following a description of the experiment itself in Section 8.2. A more detailed description of the preanalysis is presented in two ORNL GCFR progress reports (refs. 3,4), in a paper presented to the American Nuclear Society (ref. 5), and in a topical report (ref. 6). The experiment is also described in the cited progress reports and in a topical report (ref. 7). The final analysis of the experiment is summarized in Section 8.4 and is presented in more detail in refs. 8 - 10.

8.2. THE EXPERIMENT

The GCFR radial blanket and shield experiment was performed at the ORNL Tower Shielding Facility in 1979 (March through September) with an experimental configuration that included mockups of the blanket region, the inner and outer radial shields, and a portion of the PCRV. As in the earlier experiments, it was necessary to modify the spectrum of the neutrons emerging from the TSR-II collimated beam so that it would resemble the spectrum of neutrons that would be emerging from the GCFR core, this time those neutrons leaving the core in the radial direction. Calculations performed during the preanalysis of the experiment (see Section 8.3) indicated that the desired spectrum could be obtained by filtering the beam through 10.2 cm of iron, 9 cm of aluminum, and 2.54 cm of Boral; therefore a spectrum modifier including these materials was inserted in the beam adjacent to the TSR-II shield-collimator. The individual slabs used in the

spectrum modifier were 1.5 m on a side, and the slabs were surrounded with lithiated paraffin and concrete blocks, both to minimize neutron scattering back into the slabs and to reduce the background at the detectors. The resulting configuration, identified as Configuration 1.A, is shown in Fig. 8.2.

As shown in Table 8.2, Configuration 1.A was the first of 24 configurations included in the experiment. The experiment began with detailed measurements on the beam centerline at various distances beyond this configuration, the measurements including neutron flux measurements with Bonner balls, neutron spectral measurements with the NE-213 spectrometer and the hydrogen counter, and gamma-ray spectral measurements with the NE-213 spectrometer.

Next, two types of blanket configurations were tested - one that used slabs of ThO_2 , which was the alternate material being considered for the GCFR blanket, and another that used slabs of UO_2 , which was the material specified in the CSC-1 design. Again the slabs were 1.5 m on a side and were surrounded by lithiated paraffin and concrete, as were all the other materials subsequently added to the configurations.

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Fig. 8.2. Mockup of Spectrum Modifier Surrounded by Lithiated Paraffin and Concrete (Configuration 1.A).

Table 8.2. Summary of Configurations and Measurements for GCFR Radial Blanket and Shield Experiment

Configuration	Description ^a	Measurements ^b
I.A	Spectrum modifier (SM) only	BB,S
I.B	SM + 1/2 ThO ₂ blanket row	BB
I.C	SM + 1 ThO ₂ blanket row	BB
I.D	SM + 2 ThO ₂ blanket rows	BB,S,TLD
I.E	SM + 3 ThO ₂ blanket rows	BB,S,TLD
I.F	Conf. I.E + Fe reflector	TLD
II.A	3 UO ₂ blanket rows	BB,S,TLD
II.B	Conf. II.A + reference inner shield	BB,S
III.A	Conf. I.E + boronated graphite (first row of inner shield)	BB,S
III.B	Conf. III.A + stainless steel (completing reference inner shield)	BB,S
IV.A	Conf. III.B + 1/3 reference outer shield	BB
IV.B	Conf. III.B + 2/3 reference outer shield	BB
IV.C	Conf. III.B + reference outer shield	BB,S
IV.D	Conf. IV.C + Fe + concrete	BB,TLD
IV.E	Reference inner and outer shields (blanket removed)	BB,S
V.A	Conf. II.A (repeat)	BB
V.B	Conf. V.A + graphite	BB,S
V.C	Conf. V.B + boronated graphite + steel	BB,S,TLD
V.D	Conf. V.C + steel	BB,TLD
V.E	Conf. V.A + boronated graphite + steel	BB,TLD
VI.A	Lead only	S
VI.B	Lead + spectrum modifier	S
VI.C	UO ₂ only	S
VI.D	UO ₂ + spectrum modifier	BB,S

^aReference shields refer to "Conceptual Shielding Configuration 1."

^bBB = Bonner ball; S = spectrometers; TLD = thermoluminescent dosimeter.

The ThO_2 slabs were fabricated by filling four type 304L stainless steel containers with ThO_2 powder, two of the containers having a nominal ThO_2 thickness of 6.2 cm and two having a nominal thickness of 13.5 cm. The wall thicknesses were approximately 0.5 cm on the thinner containers and approximately 1 cm on the thicker containers.

Four ThO_2 blanket configurations were tested, approximating one-half, one, two, and three blanket rows respectively (Configurations I.B, I.C, I.D and I.E). Bonner ball measurements were made on the centerline beyond all four configurations and spectrometer measurements were made beyond the two thickest configurations (I.D and I.E). The arrangements for Configurations I.B, I.C, and I.D are shown in Fig. 8.3 and the arrangement for Configuration I.E is shown in Fig. 8.4.

In Configurations I.D and I.E, energy deposition measurements within the configurations were also made. For these, thermoluminescent detectors (TLDs) were positioned in the voids between the blanket slabs. So that the measurements could be obtained at approximately 7-cm intervals along the axis of the blanket region, the order of the thick and thin ThO_2 slabs in Configuration I.D was revised before an additional thick slab was added to form Configuration I.E. And, finally, in order to increase the number of TLD measurement points even more, an additional configuration (I.F) was formed by rearranging the ThO_2 slabs of Configuration I.E. The rearranged configuration ended with a thin slab and 25 cm of iron. Adding the iron allowed a TLD measurement to be made immediately behind the blanket region, the additional iron being necessary to simulate an infinite medium.

The UO_2 slabs used for the alternate blanket configurations had been fabricated for earlier (LMFBR) experiments and they included some materials that would be foreign to the GCFR, in particular, significant amounts of aluminum and sodium. The "slabs" were iron containers (0.476-cm-thick walls) filled with UO_2 pellets stacked end to end in aluminum tubes. The tubes were positioned vertically on a triangular pitch within the containers, with the void between the tubes filled with sodium. The thickness of the slabs was 10.1 cm, with the UO_2 comprising 64.6% of the volume. The aluminum comprised 11.2 vol.% and the sodium 23.2 vol.%. The remaining 1 vol.% consisted of an argon-filled space between the UO_2 pellets and the aluminum tubing.

Only one UO_2 blanket configuration (11.A) was tested, and it consisted of three of the 10.1-cm-thick UO_2 slabs behind the spectrum modifier (see Fig. 8.5). A full complement of measurements was made beyond the configuration and TLD measurements were made within the voids between the slabs.

Following the measurements with the blanket mockups, three configurations with mockups of the GCFR inner shield were tested. As noted in Table 8.1, the CSC.1 design specified that the first row of the inner shield would consist of 1 cm of stainless steel (type 316), 12.7 cm of a mixture of graphite and B_4C (25 wt.% boron), and 1.3 cm of stainless steel. The second row, to be separated from the first by a 5.1-cm helium gap, would consist of 15 cm of the steel.

In the experiment the first row of the inner shield was simulated by filling a rectangular stainless steel container (inside thickness of 14.45 cm) with a mixture of B_4C powder and powdered graphite. Analyses for the boron content gave a spread of 26.18 to 27.15%. The stainless steel wall thicknesses were 1.40 cm on one side and 0.953 cm on the other. The second row of the inner shield was simulated with three type 304 stainless steel slabs, each about 5 cm thick.

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Fig. 8.5. Spectrum Modifier and UO_2 Radial Blanket (Configuration 11.A).

For the first inner shield configuration, the UO_2 blanket configuration was extended with a mockup of both rows of the inner shield (Configuration II.B as shown in Fig. 8.6). Next, the UO_2 blanket was replaced with the full ThO_2 blanket, and only the first row of the inner shield was used to form Configuration III.A (see Fig. 8.7). Finally, the second row of the inner shield was added, simulating the full CSC.1 blanket and inner shield in Configuration III.B. Measurements were made beyond, but not within, all these configurations.

The next three configurations in the experiment resulted from the sectional buildup of a mockup of the GCFR outer shield. Table 8.1 shows that the CSC.1 outer-shield design consisted of 5.1 cm type 316 stainless steel, 5.1 cm of the B_4C -graphite mixture, 22.6 cm of graphite, another 5.1 cm of the B_4C -graphite mixture, and a final 5.1 cm of type 316 stainless steel.

In the first outer-shield configuration in the experiment the innermost portion of the outer shield was simulated with a 4.7-cm thickness of type 304 stainless steel plus an approximately 5-cm thickness of the B_4C + graphite mixture in a stainless steel container (type 304L) whose walls were 0.95 cm thick on one side and 1.51 cm thick on the other. This configuration was designated as Configuration IV.A and represented the CSC.1 design through an outer radius of 244.4 cm (see Table 8.1). Its boundaries are shown in Fig. 8.8.

In the two following configurations (also depicted in Fig. 8.8) a 22.9-cm thickness of solid graphite was added for Configuration IV.B, and a second approximately 5-cm-thick stainless steel container filled with B_4C and graphite plus 4.5 cm of stainless steel were added for Configuration IV.C. Bonner ball measurements were made beyond all three radial shield configurations. In addition, spectral measurements were attempted behind Configuration IV.C; however, the low neutron intensity behind the configuration made the NE-213 data questionable and the hydrogen counter data unusable.

Two additional configurations were included in the series of outer radial shield tests. In one (Configuration IV.D in Fig. 8.8) a mockup of a portion of the PCRV was placed behind the radial shield with an intervening 35.6-cm void. Bonner ball measurements were made in the middle of the void and TLD measurements were made on each

Fig. 8.6. Spectrum Modifier, UO₂ Radial Blanket, and Inner Radial Shield (Configuration II.B).

Fig. 8.7. Spectrum Modifier, ThO₂ Radial Blanket, First and Second Rows of Inner Radial Shield (Configurations III.A and III.B).

Fig. 8.8. Spectrum Modifier, ThO_2 Radial Blanket, and Increasing Thicknesses of Inner and Outer Radial Shields (Configurations IV.A, IV.B, and IV.C); PCRV Added (Configuration IV.D).

side of the 15.2-cm thickness of stainless steel and against the front face of the iron lining the PCRV. In the other configuration (IV.E) the blanket region preceding the inner and outer shield mockups was removed to increase the neutron intensity beyond the configuration and allow spectral measurements to be made.

To fulfill Objective 3 as given in Section 8.1 above, measurements were also made beyond alternate inner shield configurations. It had been planned that the blanket region in these additional configurations would be comprised of the ThO_2 slabs, but calculations performed during the course of the experiment and described below have revealed large discrepancies between the calculated and measured results for the configurations in which ThO_2 was used. By contrast, the calculated and measured results for configurations in which the UO_2 slabs comprised the blanket region were in good agreement; therefore, it was decided to use UO_2 in the alternate-shield configurations. These configurations were identified as Configurations V.A - V.E and are depicted in Figs. 8.9 - 8.11.

Fig. 8.9. Spectrum Modifier, UO₂ Radial Blanket and Phases I and II of Alternate Inner Radial Shield (Configurations V.A, V.B, and V.C).

Fig. 8.10. Spectrum Modifier, UO₂ Radial Blanket, and Phase III of Alternate Inner Radial Shield (Configuration V.D).

Fig. 8.11. Spectrum Modifier, UO₂ Radial Blanker, and Final Phase of Alternate Inner Radial Shield (Configuration V.E).

Configuration V.A was actually a repeat of Configuration II.A to ensure the repeatability of the source term. Configuration V.B consisted of Configuration V.A plus 15.2 cm of graphite, and Configuration V.C consisted of Configuration V.B plus the 14.45-cm-thick B₄C-graphite slab and 4.68 cm of stainless steel. For Configuration V.D 15.6 cm of stainless steel was inserted ahead of the 4.68-cm thickness. Finally, for Configuration V.E both the 15.6 cm of stainless steel and the 15.2 cm of graphite were removed.

Bonner ball measurements (3-, 6-, and 10-in.) were made beyond each of the alternate shield configurations, neutron and gamma-ray spectral measurements were made beyond Configurations V.B and V.C, and TLD measurements were made within voids in Configurations V.C, V.D, and V.E.

The final series of measurements in the experiment was added because of differences between the measured and the first calculated spectra of neutrons beyond several configurations. The concern was that the calculated source term (as defined for the

TSR-II beam or as calculated to be emerging from the spectrum modifier) was in error. What was needed was a direct measurement of the TSR-II beam spectrum, but this was not possible since the residual gamma-ray flux would have saturated the spectrometer. Therefore, measurements of the beam were made with the NE-213 spectrometer shielded in one case by lead and in another by UO_2 . In addition, measurements were made with the spectrum modifier in position behind the lead and UO_2 .

The detailed results of all the measurements made during the experiment are presented in ref. 7. In this report selected measurements are compared with the corresponding calculated responses in Section 8.4 below.

8.3. PREANALYSIS AND EXPERIMENTAL DESIGN

The experiment described in Section 8.2 was designed on the basis of a detailed preanalysis that consisted of one-dimensional (ANISN) transport calculations for trial experimental configurations mocking up the GCFR blanket and radial shields. The preanalysis included calculations of the sensitivities of the TSF detector responses to particular nuclear reactions within the configurations, and also calculations of contribution fluxes (i.e., the fluxes of neutrons or gamma rays within the configuration that reach the detectors).

The preanalysis actually began with P_3S_8 ANISN transport calculations for the GCFR Conceptual Shielding Configuration I (CSC.I) in order to gain some insight into what the experimental configurations should represent. For these calculations CSC.I was modeled in one-dimensional cylindrical (R,θ) geometry using the material compositions and thicknesses specified for the design at the core midplane. The model covered the region from the core vertical axis to approximately 30 cm into the PCRV. The source was a distributed fission source calculated by General Atomic, and the cross sections used were from the ORNL-GCFR coupled cross-section library (51 neutron groups, 25 gamma-ray groups). The calculated results were given as neutron and gamma-ray spectra at key locations in the system, and it was these spectra that were to be reproduced in the experimental configurations.

One of the spectra to be reproduced was, of course, the spectrum of neutrons leaking from the GCFR-CSC.I core. Since the spectrum of the TSR-II collimated beam differed from

the CSC.1 spectrum, the first calculations for the experimental design consisted of one-dimensional transport calculations in slab geometry to determine how various trial materials would modify the TSR-II beam. It was found that the spectrum modifier shown in Fig. 8.2 gave a leakage spectrum that essentially matched the spectrum of neutrons from the CSC.1 core in the high-energy region (see Fig. 8.12), which was the important region.

Next, transport calculations were performed for experimental configurations consisting of the spectrum modifier plus materials mocking up the radial blanket. It was originally intended that in addition to making measurements for the ThO_2 blanket specified in CSC.1, this phase of the experiment would include a duplicate set of measurements for a UO_2 blanket. Therefore, the preanalysis was to include calculations for both types of blankets. However, when attempts were made to locate the UO_2 for the experiment, it was found that an insufficient amount of "clean" UO_2 was available. UO_3 was obtainable from the Savannah River Project, but transport calculations showed that water and nitrate contaminants in the UO_3 would produce an undesired leakage spectrum. Similarly, UO_2 "slabs" used in earlier TSF LMFBR experiments contained sodium and aluminum that would produce a deep dip in the energy spectrum over a broad sodium resonance. It was therefore decided not to perform a complete set of UO_2 measurements and instead to include a few measurements with the LMFBR slabs already on hand (see Section 8.2).

Fig. 8.12. Comparison of Neutron Spectra from GCFR-CSC.1 Core, In TSR-II Plan, and from Spectrum Modifier.

An important goal of the preanalysis was to ascertain whether analyses of the measurements made in the experiment would provide information on the nuclear reactions of particular interest. This was determined by sensitivity analyses, the results of which indicated how much a given detector response would change with a change in the density of a specific region of the configuration, thereby indicating the importance of nuclear reactions in the material(s) of that region and identifying the particular types of reactions that were important. The sensitivity analyses were performed with the SMANLAK sensitivity code, which uses fluxes calculated from both forward and adjoint ANISN calculations.

The one-dimensional models used for the sensitivity analyses of the Bonner ball responses beyond the UO_2 and ThO_2 blanket configurations are shown in Fig. 8.13 (Sketches A and B). The analysis showed that the Bonner ball responses were very sensitive to changes in the densities in the important nuclides (uranium and thorium), which indicated that reliable interpretation of the data would be possible in the final analysis. On the other hand, the responses were also quite sensitive to the oxygen in the blanket regions, which meant that the heavy metals and the oxygen both would have to be considered carefully in the final analysis of the experiment. The relative sensitivities to the heavy metals and oxygen varied markedly with the size of the Bonner ball, however, and the differences between the Bonner ball responses were expected to facilitate identification of the interactions contributing to any discrepancies that might occur between measured and calculated values.

In the sensitivity calculations for TLD responses in the blanket configurations, it was assumed that the detector was positioned between the last two blanket slabs shown in Sketches A and B of Fig. 8.13. (Note: The model actually employed for the TLD calculations included 10.1 cm of UO_2 in the spectrum modifier instead of the 9 cm of aluminum.) In contrast to the Bonner balls, the sensitivities of the TLD were not exceptionally large, but they appeared to be adequate for reliable interpretation of the results.

An important result of the TLD calculations was the discovery that 21% of the gamma rays detected in the ThO_2 configuration and 33% of those detected in the UO_2 configuration were secondary gamma rays produced in the steel on each side of the

Fig. 8.13. One-Dimensional Models Used for Preamalysis. The slab calculations had a boundary source at the left boundary and a void on the right boundary.

detector. The remaining gamma rays were produced in the thorium and uranium, with the region behind the TLD contributing approximately as much to the response as the region preceding the TLD. It was also noted that gamma rays born in locations farther than 25 cm from the TLD did not contribute to the response. So that TLD measurements could be made in which a higher percentage of the detected gamma rays would be coming from the blanket materials, one of the ThO_2 blanket rows was fabricated as two half-rows, each with thinner stainless steel walls. The TLD could then be positioned with less steel on either side.

Following the studies of the blanket configurations, calculations were performed for the inner and outer radial shields, using the model shown in Sketch C of Fig. 8.13. These calculations included sensitivity calculations for a Bonner ball positioned in the void between the outer shield and the PCRV. Again the Bonner ball sensitivities were found to be large (especially for the stainless steel and the graphite) and the variation in the relative sensitivities between the three Bonner balls was apparent.

Calculations were also performed for a fictitious total gamma-ray detector located between the outer shield and the PCRV. Of the gamma rays reaching the detector, nearly 75% were produced as secondary gamma rays in the stainless steel claddings on the shield slabs and nearly 10% in the PCRV carbon steel lining. The only other significant contribution was from the boron in the shields (13.6%). Essentially no "source" gamma rays (those entering from the blanket) were detected.

Finally, contribution fluxes were calculated for the Bonner balls and the fictitious gamma-ray detector in the gap. For these calculations the TOOTH code (a one-dimensional equivalent to the FANG code) was written for the purpose of calculating the contribution fluxes as a function of both space and energy and plotting the results on a three-dimensional surface. For the Bonner balls the results indicated the relative magnitudes and energies of the neutron fluxes in the various shield regions that would contribute to the Bonner ball responses. Figure 8.14 shows, for example, that of the neutrons passing through the inner shield, only those with high energies will reach the Bonner balls in the gap. However, because the 3-in. Bonner ball largely detects low-energy neutrons and the 10-in. ball largely detects high-energy neutrons, the two balls will "see" different neutrons from the outer shield. Also, the 3-in. ball will detect neutrons scattered back from the PCRV concrete.

The interpretation of the contribution fluxes for the gamma-ray detector is more complicated since it is neutron interactions that produce the detected gamma rays. Thus both neutron fluxes and gamma-ray fluxes must be considered and interrelated. Such an interrelationship is shown in Fig. 8.15. The top sketch shows the magnitudes and energies of the neutron fluxes that will produce the gamma rays reaching the detector and also shows the shield regions in which those interactions will take place. The lower sketch shows the magnitudes and energies of the resulting gamma-ray fluxes and also the shield regions in which the gamma rays are produced.

The dominant peak of the neutron flux (in the inner shield region) probably represents neutron inelastic scatterings in the steel while the lower peaks around it are probably due to the capture of intermediate-energy neutrons in the steel. The low-energy peaks on either side of the detector are due to low-energy neutron captures in the outer shield graphite and the concrete of the PCRV.

ORNL-DWG 79-14139

(a) 3-in. Bonner Ball.

ORNL-DWG 81-17263

(b) 10-in. Bonner Ball.

Fig. 8.14. Neutron Contribution Fluxes for Bonner Balls Located in Gap Between Outer Shield and PCRV.

ORNL-LWG 79-14140

(a) Neutron Contribution Flux.

ORNL-LWG 79-14141

(b) Gamma-Ray Contribution Flux.

Fig. 8.15. Contribution Fluxes for Fictitious Total Gamma-Ray Detector In Gap Between Outer Shield and PORV.

The gamma-ray fluxes show the characteristic 3-MeV minimum in the gamma-ray interaction cross section as a ridge through the shield. The 7- to 8-MeV gamma rays produced by neutron capture in the steel and 0.511-MeV annihilation gamma rays are also apparent.

In addition to the above, the preanalysis included transport calculations to resolve two problems that arose during the fabrication of components to be used in the experiment. The first one was associated with a difficulty in attaining the desired packed density of the B_4C + graphite powder to be used in some of the shield slabs. When the dimensions for the shield regions of the experimental configuration were set, it had been assumed that the density of 1.6 g/cc called for in the CSC.1 design could be achieved, and therefore the stainless steel cans fabricated to hold the powder were specified to have the same thicknesses as the corresponding CSC.1 regions. However, when the vendor attempted to fill the containers, the highest bulk density attainable was 1.0 g/cc. Transport calculations were performed to determine the effect of the reduced density, and it was found that the shape of the neutron leakage spectra from the region would be the same for both densities. It was therefore decided to use the cans with the lower density and to correct for the difference in the magnitude of the flux. As it turned out, however, ORNL experts instructed the vendor in the art of two-particle vibration packing, and a bulk density of 1.4 g/cc was eventually achieved.

The second problem had to do with a high natural background activity in the ThO_2 acquired for the experiments. A measured activity of 50 mR/h at the surface of the slabs precluded positioning the gamma-ray spectrometer close behind the slabs, while other background problems discouraged moving the spectrometer back. One alternative was to filter the background radiation through lead or iron. Since the iron cross sections were better known than the lead cross sections, and thus would introduce less uncertainty in the final analysis of the experiment, transport calculations were performed to determine the effect on the leakage spectra of using an iron filter. The results showed that the neutron spectrum suffered adverse effects and the gamma-ray spectrum was converted from a ThO_2 leakage spectrum to an iron leakage spectrum, which was unacceptable. The final solution was to move the spectrometer back and to protect it from other background effects by surrounding it with concrete housing containing a collimator.

8.4. POSTANALYSIS OF THE EXPERIMENT; COMPARISON OF CALCULATED AND MEASURED RESULTS

Table 8.2 lists a total of 24 experimental configurations. Of these, 12 representative configurations were selected for final analysis, with most of the analysis effort consisting of the usual two-dimensional DOT calculations of radiation transport through the configurations mocked up in R-Z cylindrical geometry. For the initial calculations an external boundary angular flux describing the TSR-II source was incident on one end of the cylinder, but as the configurations became larger and more complicated (including fissions in the blanket regions as well as transport), internal boundary sources were saved for use as external sources in subsequent calculations. All the DOT calculations used an S_{10} symmetric angular quadrature (70 angles), together with the GCFR 51-group neutron cross sections coupled with 25-group gamma-ray cross sections. The FALSTF code was used to determine the fluxes at detector locations beyond the configurations.

In addition to the DOT-FALSTF sequence, a series of ANISN calculations in one-dimensional slab geometry were performed to compare different cross-section data. In particular, the calculations examined the effect of substituting into the calculations the ENDF/B-V data for thorium, which were not released until after the GCFR cross-section set had been in use for some time.

8.4.1. Definition of Source

As in the analyses for the earlier experiments, the TSR-II large collimated beam used as the source in the experiment was converted to the disk boundary source required for the two-dimensional calculations (see Section 7.4.1 and Fig. 5.3). A DOT calculation was performed that included the disk source and the spectrum modifier (see Fig. 8.16), and the resulting internal boundary source at the inner surface of the spectrum modifier was then used as the initial external boundary source for the analysis.

8.4.2. Transport Calculations for Spectrum Modifier

The first transport calculations performed in the analysis of the radial shield confirmation experiment were for the spectrum modifier configuration shown in Fig. 8.2

Fig. 8.16. Two-Dimensional R-Z Model of TSR-II, Beam Collimator and Spectrum Modifier (Configuration 1.A).

(Configuration 1.A). Since this configuration was sufficiently thin for neutrons from the TSR-II beam to pass through it without undergoing a collision, calculations of the uncollided fluxes had to be performed separately, by FALSTF, and added to the DOT-FALSTF collided fluxes determined for each detector point. The total flux at each point was then converted to Bonner ball responses. The resulting ratios of the calculated and measured responses for 3-, 6-, and 10-in. Bonner balls are shown for three on-axis locations behind the spectrum modifier in Table 8.3.

Table 8.3. Calculated-to-Measured Bonner Ball Responses Beyond Spectrum Modifier (Configuration 1.A)

Distance (cm)	Measured Response (counts/min-W)			Calc./Meas.			Method Used for Uncollided Flux ^a
	3-in. BB	6-in. BB	10-in. BB	3-in. BB	6-in. BB	10-in. BB	
30.5	1.25(5) ^b	8.22(5)	3.92(5)	1.06	1.07	1.01	FALSTF
304.8	5.95(3)	3.89(4)	1.78(4)	0.94	0.93	0.93	FALSTF
396.9	3.40(3)	2.32(4)	1.08(4)	1.00	0.95	0.93	FALSTF
396.9	3.40(3)	2.32(4)	1.08(4)	1.06	1.04	1.07	Point TSR-II Source
396.9	3.40(3)	2.32(4)	1.08(4)	1.02	0.99	1.01	Shell TSR-II Source

^aAll collided fluxes calculated by DOT-FALSTF.

^bRead: 1.25×10^5 .

The underprediction of the total and high-energy neutron fluxes at the distant detector locations, as reflected in the low "Calc./Meas." values for the 6-in. and 10-in. Bonner balls respectively (lines 2 and 3 in Table 8.3), was attributed to the angular quadrature used by FALSTF being too coarse; therefore, two alternate approaches were also used to calculate the uncollided fluxes at these locations. In the first alternate approach the TSR-II was approximated as a point source and the uncollided flux was calculated from the simple relation:

$$\phi_u = (S/4\pi R^2) \exp[-(\Sigma_T X)_{Fe} - (\Sigma_T X)_{Al} - (\Sigma_T X)_{Boral}]$$

where ϕ_u is the uncollided flux, S is the TSR-II source strength, R is the distance from the TSR-II to the detector, Σ_T is the total macroscopic cross section, and X is the slab thickness. The second approach used the same relation but treated the reactor source as a shell source which required integration over the surface of the reactor. This second technique yielded the best results as shown by a comparison of the entries in the last two lines of Table 8.3.

The same three techniques were applied to the calculation of the neutron energy spectra for comparison with measurements made at the 396.9-cm location with the NE-213 spectrometer (800 keV to 15 MeV) and the hydrogen counter (50 keV to 1 MeV). The results are shown in Fig. 8.17. Again the shell source approach (labelled "sphere" in the figure) appears to give the best overall results.

8.4.3. Transport Calculations for Blanket Configurations

As pointed out in Section 8.2, blanket configurations using both ThO_2 and UO_2 were measured. Emphasis was placed on the ThO_2 measurements because of the particular interest in "alternate" fuel cycles and because the uncertainties associated with the cross sections were larger for thorium than for uranium. Also, as explained earlier, "clean" UO_2 slabs were not available for the experiment.

Calculated-to-measured ratios of Bonner ball responses beyond two of the ThO_2 blanket configurations and one UO_2 configuration are shown in Table 8.4. In general, the calculations underpredict the responses for the UO_2 configuration (II.A), particularly at

Fig. 8.17. Comparison of Calculated and Measured Neutron Spectra 396.9 cm Behind Spectrum Modifier (Configuration 1.A).

Table 8.4. Calculated-to-Measured Bonner Ball Responses Beyond Blanket Configurations

Configuration	Distance (cm)	Measured Response (counts/min-W)			Calc./Meas.		
		3-in. BB	6-in. BB	10-in. BB	3-in. BB	6-in. BB	10-in. BB
11.A (~30 cm UO ₂)	30.5	1.50(4) ^a	7.44(4)	2.44(4)	0.98	0.91	0.83
	285.4	8.06(2)	3.60(3)	1.23(3)	0.88	0.86	0.77
	304.8	7.15(2)	3.21(3)	1.09(3)	0.88	0.84	0.77
11.D (~30 cm ThO ₂)	30.5	1.42(4)	7.20(4)	2.33(4)	0.99	0.95	0.91
	304.8	5.94(2)	2.86(3)	9.90(2)	1.03	1.05	0.96
	1314.1	2.74(1)	1.37(2)	4.90(1)	1.28	1.26	1.12
11.E (~45 cm ThO ₂)	30.5	5.78(5)	1.62(4)	4.84(5)	1.02	1.03	0.99
	304.8	1.69(2)	6.83(2)	2.48(2)	1.01	1.09	0.88
	1296.4	7.83(0)	3.21(1)	1.08(1)	1.27	1.36	1.20

^aRead: 1.60×10^4 .

large distances and for high neutron energies (in the 10-in. Bonner ball range). By contrast, but with exceptions, the calculations overpredict the responses for the ThO_2 configurations (I.D and I.E), particularly at large distances. However, in light of the combined experimental and analytical uncertainties, which range as high as 10, 15, and 20% for the 3-, 6-, and 10-in. balls, respectively, the agreement is considered to be good in most cases.

Comparisons of calculated and measured neutron spectra at a distance of approximately 1300 cm beyond each of the configurations also showed relatively good agreement (see upper curves in Figs. 8.18a-c). In the case of the UO_2 configuration, the calculations underpredicted the spectrum 20 to 30% below 2 MeV, while in the cases of the ThO_2 configurations the calculations overpredicted the spectrum somewhat, more so for the thicker blanket.

Calculations of the gamma-ray spectra beyond the blanket configurations were not nearly so successful. For the UO_2 configuration (II.A) the calculated spectrum was low by about 50% at 1 MeV and by as much as a factor of 5 at 7 MeV. For the ThO_2 spectra the calculations and measurements are in fair agreement below 4 MeV, but the calculations severely underpredicted the measurements above 4 MeV. It is to be remembered, however, that the ENDF/B-IV cross sections applied in these calculations did not include thorium gamma-ray production data, and the gamma-ray production cross sections used were estimated from uranium gamma-ray production cross sections.

The inadequacy of the estimated gamma-ray production cross sections is also apparently reflected in the calculated TLD heating rates for Configuration I.E (see Table 8.5). The large underprediction of the heating rate in the first void in Configuration I.E (and also in Configuration II.A) can, of course, be attributed to the absence of a gamma-ray component in the calculated TSR-II source. But the heating rates in the subsequent voids would be primarily due to secondary gamma rays and therefore would be determined by the gamma-ray production cross sections. In the case of the UO_2 configuration the calculated heating rates for Voids 2 and 3 are within the 7 to 10% uncertainties in the experiment, indicating that the gamma-ray production and absorption cross sections for uranium are reasonable. But in the case of the ThO_2 configuration, large discrepancies appear for Voids 2, 3, and 4, the calculations overpredicting the measurements by 25 to 37%.

(a) Configuration I.D.

(b) Configuration I.E.

(c) Configuration II.A.

Fig. 8.18. Comparison of Calculated and Measured Neutron and Gamma-Ray Spectra 1300 cm Beyond Blanket Configurations.

Table 8.5. Comparison of Calculated and Measured (TLD) Heating Rates in Blanket Configurations

Configuration	Location	Heating Rate (MeV/(g-sac-cm))				
		Measured	Calculated			Calc./Meas.
			Gamma	Neutrons	Total	
II.A ^a	Void 1 ^b	9.60(5) ^c	5.05(5)	4.48(4)	5.50(5)	0.57
	Void 2	1.26(5)	1.02(5)	1.59(4)	1.18(5)	0.93
	Void 3	4.52(4)	4.17(4)	5.38(3)	4.71(4)	1.04
I.E ^d	Void 1 ^b	9.72(5)	5.00(5)	4.10(4)	5.41(5)	0.56
	Void 2	7.77(4)	8.92(4)	8.25(3)	9.75(4)	1.25
	Void 3	4.00(4)	5.08(4)	3.80(3)	5.46(4)	1.37
	Void 4	1.88(4)	2.37(4)	1.61(3)	2.53(4)	1.34

^aSee Fig. 8.5.^bFirst void following spectrum modifier.^cRead: 9.60×10^5 .^dSee Fig. 8.4.

By the time this analysis was being performed, ENDF/B-V thorium cross sections that included gamma-ray production data had been released. In order to gain some insight as to what effect the new cross sections would have on the calculated results for the blanket configurations, a technique was devised for introducing the ENDF/B-V thorium data into the Configuration I.E analysis via the use of a series of one-dimensional ANISN calculations, which are faster and more economical than the two-dimensional DOT calculations.

The first step in the technique was to perform a 76-group one-dimensional ANISN calculation for Configuration I.E with the same ENDF/B-IV cross sections that had been employed in the original two-dimensional DOT analysis. The DOT-to-ANISN ratio could then be used as a scaling factor to convert one-dimensional results to two-dimensional results. Next, the ENDF/B-V thorium data, which were in 211 groups, were collapsed to the 207 groups required by the VITAMIN C Library (171 neutron groups and 36 gamma-ray groups) and combined with the non-thorium data of the GCFR ENDF/B-IV cross-section set that had been processed into the 207-group structure. The combined data set was then used in an ANISN calculation for Configuration I.E, and the results were multiplied by the scaling factor to obtain the comparable two-dimensional results. But to make a

direct comparison with the ENDF/B-IV thorium cross sections, it was necessary to also perform a 207-group ANIS calculation with ENDF/B-IV data and multiply the results by the scaling factor. A comparison of the two scaled 207-group calculations then showed the effect of substituting ENDF/B-V thorium data for ENDF/B-IV thorium data. (Note: In this case no estimated gamma-ray production cross sections were added to the ENDF/B-IV thorium data set.)

The neutron and gamma-ray spectra resulting from the two scaled calculations are compared with the measured spectra in Figs. 8.19a-b. The two calculated neutron spectra differ by 20 to 40% between 0.1 and 3 MeV, with the ENDF/B-V results showing the better agreement with the measurement. The differences are essentially a direct reflection of the differences between the total neutron cross sections for ENDF/B-IV and -V shown in Fig. 8.20.

The gamma-ray spectra in Fig. 8.19b show that without the estimated thorium gamma-ray production cross sections added to the ENDF/B-IV data the calculation severely underestimates the gamma-ray spectrum in all energy regions. The ENDF/B-V data significantly improve the agreement with the measurement in energy regions below 4.5 MeV, but the calculated results are still too low by a factor of 2 to 3. Above 4.5 MeV, both cross-section sets underestimate the measured spectrum by about a factor of 50. In addition, the peaks in the calculated spectra appear at a different energy than the peak in the measured spectrum.

The large discrepancy between all the calculated gamma-ray spectra and the measured spectrum for gamma-ray energies above 4.5 MeV implies a total inadequacy of the thorium cross sections for predicting the production of high-energy gamma rays. However, when the fluxes that were obtained within Configuration 1.E by the scaled ANIS calculation using the ENDF/B-V thorium data were folded with the TLD response functions, the resulting heating rates were in excellent agreement with the measured heating rates (see Table 8.6). This suggests that the problem was not within the configuration but rather beyond it — perhaps in the measurements or in the technique for projecting the calculated gamma-ray fluxes from the end of the configuration to the location of the spectrometer. These possibilities are discussed below in Section 8.4.5.

Fig. 8.19. Comparison of Neutron and Gamma-Ray Spectra Measured 1296.4 cm Beyond Blanket Configuration 1.E with Spectra Calculated with ENDF/B-IV and -V Thorium Data. (Note: All non-Th data are ENDF/B-IV in both calculations; no Th gamma-ray-production data in ENDF/B-IV Th data.)

Fig. 8.20. Comparison of ENDF/B-IV and ENDF/B-V ^{232}Th Total Neutron Cross Section in 171-Group Structure.

Table 8.6. Comparison of Measured (TLD) Heating Rates in Configuration 1.E with Heating Rates Calculated with ENDF/B-IV and -V Thorium Cross Sections

Calculation	Calc./Meas.			
	Void 1	Void 2	Void 3	Void 4
2-D 76-group calculation using ENDF/B-IV data plus estimated Th gamma-ray production data (original analysis)	0.56	1.25	1.37	1.34
Scaled 1-D 207-group calculation using ENDF/B-IV data (no Th gamma-ray production data included)	0.48	0.36	0.31	0.37
Scaled 1-D 207-group calculation using ENDF/B-V Th data with ENDF/B-IV non-Th data (includes Th gamma-ray production data)	0.54	1.01	1.08	1.08

8.4.4. Transport Calculations for Shield Configurations

As has been described in Section 8.2, the successive shield configurations tested during the experiment primarily consisted of the sectional buildup of the inner and outer radial shields specified in Table 8.1 for the CSC.1 reference design. In some cases, however, the shield regions were preceded by a UO_2 blanket rather than a ThO_2 blanket, and in other cases alternate shields were substituted for regions of the reference shield.

As noted in Table 8.1, the CSC.1 design specifies that the first row of the reference inner shield be 12.7 cm of a B_4C -graphite mixture between stainless steel walls and that the second row consist of 15 cm of stainless steel, the two rows being separated by a 5.1-cm helium-filled gap. In the experiment the inner shield was simulated by three configurations: Configurations III.A and III.B, which mocked up the design through the first and second rows, respectively, and Configuration II.B, which mocked up the design through both rows but utilized a UO_2 blanket rather than a ThO_2 blanket (see Figs. 8.7 and 8.6).

The results of Bonner ball measurements and the corresponding calculated-to-measured ratios for various detector locations behind the inner shield configurations are presented in Table 8.7. As might have been expected, the ratios are very similar to those observed for the blanket configurations; that is, for the configurations including ThO_2 , the calculations generally overpredicted the measurements, while for the configurations including UO_2 , the calculations generally underpredicted the measurements. Thus the discrepancies noted in the calculations for the blanket configurations (see Section 8.4.3) persisted through the shield configurations - to the extent that it was difficult to ascertain how accurately the transport through the shields themselves was predicted. And the same trends were apparent in the ratios of the Bonner ball responses for two alternate shield configurations (V.B and V.C), also shown in Table 8.7 (see configurations in Fig. 8.9).

Neutron spectra were measured behind the various configurations at the same location at which the most distant Bonner ball measurements were made, and an examination of the corresponding calculated spectra indicates the energy regions contributing to the

Table 8.7. Calculated-to-Measured Bonner Ball Responses Beyond Inner Shield Configurations

Configuration ^a	Distance	Measured Response (counts/min-W)			Calc./Meas.		
		3-in. BB	6-in. BB	10-in. BB	3-in. BB	6-in. BB	10-in. BB
III.A, ThO ₂ + first row of reference inner shield (15 cm boronated graphite; see Fig. 8.7)	30.5	3.23(2) ^b	1.46(3)	4.73(2)	1.57	1.43	1.29
	304.8	1.51(1)	6.95(1)	2.38(1)	1.53	1.39	1.25
	1280.6	7.62(-1)	3.47(0)	1.22(0)	1.83	1.70	1.52
III.B, ThO ₂ + both rows of reference inner shield (15 cm boronated graphite + 15 cm stainless steel; see Fig. 8.7)	30.5	4.10(1)	3.12(2)	1.00(2)	—	1.09	1.00
	304.8	3.54(0)	1.46(1)	4.68(0)	1.19	1.03	0.97
	1263.0	1.87(-1)	7.50(-1)	2.50(-1)	1.36	1.22	1.10
II.B, UO ₂ + both rows of reference inner shield (15 cm boronated graphite + 15 cm stainless steel; see Fig. 8.6)	30.5	4.53(2)	1.96(3)	7.02(2)	0.89	0.79	0.71
	304.8	2.06(1)	9.24(1)	3.40(1)	0.82	0.69	0.64
	1277.2	1.04(0)	4.88(0)	1.83(0)	0.97	0.83	0.73
V.B, UO ₂ + alternate inner shield (15 cm graphite; see Fig. 8.9)	30.5	1.22(4)	2.36(4)	6.36(3)	0.97	0.87	0.73
	304.8	5.16(2)	1.02(3)	2.81(2)	0.94	0.84	0.72
	1294.4	2.19(1)	4.37(1)	1.27(1)	1.30	1.15	0.95
V.C, UO ₂ + alternate inner shield (15 cm graphite, 15 cm boronated graphite + 4.5 cm stainless steel; see Fig. 8.9)	30.5	1.23(2)	6.90(2)	3.16(2)	0.91	0.81	0.72
	304.8	5.88(2)	3.30(1)	1.58(1)	0.82	0.75	0.69
	1270.0	3.24(-1)	1.83(0)	9.33(-1)	0.91	0.84	0.73

^aSlab thicknesses indicated are nominal thicknesses.^bReac: 3.23×10^2 .

discrepancies in the calculated Bonner ball responses. Figure 8.21a shows, for example, that for Configuration III.A the calculations overpredicted the neutron fluxes below about 2 MeV but correctly predicted those above 2 MeV. With the addition of 15 cm of stainless steel to form Configuration III.B, the calculated spectrum only slightly overpredicted the measured spectrum in the low-energy region and underpredicted it above 4 MeV (see Fig. 8.21b). For both configurations the relatively broad structure apparent in the measurements was calculated well.

For the configurations containing UO₂, the calculations yielded a slight underprediction of the neutron fluxes over the entire spectrum (see Figs. 8.23 - 8.25), but, in general, the agreement is considered to be satisfactory.

As was the case with the blanket configurations, the gamma-ray energy fluxes calculated for the spectrometer location beyond the inner shield configurations were underpredicted in the analysis, severely so for the configurations containing the ThO₂ blanket (III.A and III.B) and substantially so for the configurations containing the UO₂ blanket (II.B, V.B, and V.C) (see Figs. 8.21 - 8.25). The possible reasons for these large discrepancies are discussed in Section 8.4.5.

Fig. 8.21. Comparison of Measured and Calculated Neutron and Gamma-Ray Spectra 1280.6 cm Beyond Configuration III.A.

Fig. 8.22. Comparison of Measured and Calculated Neutron and Gamma-Ray Spectra 1263 cm Beyond Configuration III.B.

Comparisons of the calculated and measured TLD heating rates in the inner shield configurations are shown in Table 8.8. It will be noted that in the first two cases, the comparisons are made for different but similar calculations. This is because TLD measurements had not been made in the configurations that were calculated (and vice versa); therefore the calculated TLD heating rates were compared with those measured in similar configurations that had been extended with additional materials. Unfortunately the extensions, which in one case included the outer shield (Configuration IV.D), appeared to affect the measurements. For the case in which the calculations and measurements were made for the same configuration (V.C), the agreement was very good.

For investigations of the full CSC.1 shield, the reference outer shield was added behind the reference inner shield in sections. As shown in Table 8.1, the CSC.1 design specifies that the outer shield consist of 5.1 cm of type 316 stainless steel, 5.1 cm of

Fig. 8.23. Comparison of Measured and Calculated Neutron and Gamma-Ray Spectra 1277.2 cm Beyond Configuration II.B.

Fig. 8.24. Comparison of Measured and Calculated Neutron and Gamma-Ray Spectra 1294.4 cm Beyond Configuration V.B.

Fig. 8.25. Comparison of Measured and Calculated Neutron and Gamma-Ray Spectra 1270 cm Beyond Configuration V.C.

boronated graphite, 22.6 cm of stainless steel, 5.1 cm of boronated graphite, and 5.1 cm of type 316 stainless steel.

Configurations IV.A through IV.C represent the buildup of these sections, with Configuration IV.C embodying the full shield (see Fig. 8.8). In an additional configuration (IV.D) the mockup was extended into the PCRV. And, finally, in one configuration (IV.E) the ThO_2 blanket of Configuration IV.C was removed to increase the counting rates beyond the configuration and also to remove the uncertainties associated with the calculations of the blanket region.

Table 8.8. Comparison of Calculated and Measured (TLD) Heating Rates In Inner Shield Configurations

		Heating Rate [MeV/(g·min·kW)]				
Configuration	Location	Measured	Calculated			Calc./Meas.
			Gammas	Neutrons	Total	
III.B (Calc.)	Void 1 ^b Void 2	1.03(6) ^c	9.81(5)	1.06(5)	1.09(6)	1.06
V.E (Meas.)		2.08(5)	2.64(5)	1.47(4)	2.80(5)	1.34
II.B (Calc.)	Void 1 Void 2	5.60(4)	8.01(4)	2.72(3)	8.28(4)	1.48
IV.D (Meas.)		8.50(3)	9.77(3)	5.66(2)	1.03(4)	1.22
V.C	Void 1	1.98(6)	1.77(6)	1.94(5)	1.96(6)	0.99
	Void 2	1.02(6)	9.86(5)	6.6(4)	1.05(6)	1.03
	Void 3	1.35(5)	1.43(5)	2.72(3)	1.46(5)	1.08

^aTLD measurements were not made for Configurations III.B and II.B and calculations were not performed for Configurations V.E and IV.D; therefore, the calculated and measured values compared are for different but similar configurations.

^bFirst void following spectrum modifier; see void locations in Fig. 8.6 (II.B), Fig. 8.7 (III.B), Fig. 8.8 (IV.D), Fig. 8.9 (V.C), and Fig. 8.11 (V.E).

^cRead: 1.03×10^6 .

The analysis of the outer shield configurations was confined to Configurations IV.C and IV.E. Ratios of the calculated and measured Bonner ball responses for these configurations are given in Table 8.9, and the calculated and measured neutron and gamma-ray spectra beyond Configuration IV.E are presented in Fig. 8.26. The calculated and measured Bonner ball responses are in good agreement for both configurations except for the 3-in. and 6-in. detectors at the most distant locations. The calculated and measured neutron spectra for Configuration IV.E are also in good agreement except for one energy region in which it appears that one of the three hydrogen counters used in the measurements was in error. As in other cases, the calculated and measured gamma-ray spectra are in disagreement, though less so than for the earlier cases. The gamma-ray spectra have a strong similarity to the spectra given in Fig. 8.25 for Configuration V.C.

While it has been shown in the preceding discussion that in most cases the calculated and measured neutron responses for the configurations containing the shield regions are in relatively good agreement, it is also apparent that the interpretation of the calculations of the transport through the shield regions is complicated by the uncertainties in the calculated transmission through sections preceding the shield regions. Thus to better understand the attenuation characteristics of the shields

Table 8.9. Calculated-to-Measured Bonner Ball Responses
Beyond Full (Inner Plus Outer) Shield Configurations

Configuration	Location	Measured Response (counts/min-W)			Calc./Meas.		
		3-in. BB	6-in. BB	10-in. BB	3-in. BB	6-in. BB	10-in. BB
IV.C, ThO ₂ + reference	30.5	2.32(-1)*	6.54(-1)	1.86(-1)	1.08	1.06	0.98
Inner shield + reference	304.8	1.08(-2)	2.88(-2)	8.76(-3)	1.04	1.13	1.01
outer shield (see Fig. 8.8)	1210.5	5.71(-4)	1.81(-3)	6.12(-4)	1.38	1.23	1.00
IV.E, Reference inner shield	30.5	4.66(1)	1.58(2)	4.19(1)	1.15	1.01	0.93
+ reference outer shield	304.8	1.96(0)	6.78(0)	2.09(0)	1.18	1.04	0.97
(no blanket) (Configuration IV.C minus ThO ₂)	1264.9	9.84(-2)	3.43(-1)	1.16(-1)	1.46	1.29	1.10

*Read: 2.32×10^{-1} .

Fig. 8.26. Comparison of Measured and Calculated Neutron and Gamma-Ray Spectra 1264.9 cm Beyond Configuration IV.E.

themselves, an approach was used in which a Bonner ball response for a configuration with a specific shield region was "ratioed" with the response for the same configuration with the shield region removed, thereby determining the attenuation factor for that region. The ideal way to do this would be to compare the responses at a fixed and distant detector location beyond the two configurations, thereby ensuring that the same geometric attenuation would occur in both configurations and the difference in the responses would be entirely due to material attenuation within the shield region. However, difficulties encountered in calculating distant Bonner ball responses precluded the use of this technique, and, instead, the responses at a distance of 30 cm beyond each configuration were used. As a result, the attenuation factors determined include both geometric and material attenuation.

The resulting calculated and measured attenuation factors and the corresponding calculated-to-measured ratios are presented in Table 8.10 for several configurations. It is noted that the attenuation factors for 15 cm of boronated graphite are consistently underpredicted and that those for the 15 cm of stainless steel are consistently overpredicted, the overall result being that reasonably good agreement is obtained for the combined regions. When 4.5 cm of stainless steel follows the boronated graphite the agreement is also good. In fact, for nearly all the multiregion shields the calculated-to-measured ratios are considered to be good. However, markedly different attenuation factors are obtained for the full shield with and without a preceding ThO_2 blanket (as in Configuration IV.E). Without the blanket, the attenuation in the full shield was higher by factors of 3 to 6.

It was speculated that the larger attenuation factors for the full shield without a preceding blanket were due to differences in the spectra of the neutrons incident on the shield, even though, as shown in Fig. 8.27, the spectrum emerging from the ThO_2 blanket is only moderately softer than the spectrum emerging from the spectrum modifier. Still, this difference is apparently enough to make the presence of the boronated graphite in the initial section of the shield more effective. Certainly the difference is real since it appears both in the measurements and in the calculations for all three Bonner balls used.

Finally, overall, it appears that the neutron transmission through the shield configurations was well calculated.

8.4.5. Examination of Gamma-Ray Spectral Discrepancies

It has been pointed out in the previous sections that large discrepancies were observed between the measured and calculated gamma-ray spectral measurements beyond the various configurations but that the measured and calculated heating rates within the configurations were in good agreement. Therefore it appeared that the problem arose outside the configuration, either in the measurements or in projecting the calculated fluxes to the spectrometer location. In an attempt to resolve the problem, several effects that might have been contributing to the discrepancy were investigated.

Table 8.10. Attenuation Factors for Shield Sections Determined from Ratios of Bonner Ball Responses 30 cm Beyond Configurations

Shield Section(s)	Ratio	Attenuation Factors								
		3-in. BB			6-in. BB			10-in. BB		
		Calc.	Meas.	Calc./Meas.	Calc.	Meas.	Calc./Meas.	Calc.	Meas.	Calc./Meas.
15 cm B ₄ C-Graphite	I.E/III.A	7.6	11.7	0.65	8.0	11.1	0.72	7.8	10.2	0.77
15 cm 304-SS	III.A/III.B	5.4 ^a	4.3 ^a	1.27	6.1	4.7	1.31	6.1	4.7	1.29
Ref. Inner Shield (15 cm B ₄ C-Graphite + 15 cm 304-SS)	I.E/III.B	40.9 ^a	47.8 ^a	0.86	43.8	51.8	0.94	47.9	48.3	0.99
Ref. Inner Shield	II.A/II.B	38.7	35.2	1.10	43.8	37.9	1.16	41.0	34.8	1.18
15 cm Graphite	V.A/V.B	1.3	1.3	1.01	3.3	2.95	1.06	4.4	3.8	1.13
15 cm B ₄ C-Graphite + 4.5 cm 304-SS	V.B/V.C	106	99.5	1.06	36.7	34.2	1.07	20.4	20.1	1.02
Ref. Outer Shield	III.B/IV.C	373 ^a	328 ^a	1.14	491	477	1.03	549	539	1.02
Full Shield	I.A/IV.E	2.46(3) ^b	2.68(3)	0.92	5.51(3)	5.19(3)	1.06	9.09(3)	8.36(3)	1.09
Full Shield	I.E/IV.C	1.54(4)	1.63(4)	0.95	2.41(4)	2.48(4)	0.97	2.63(4)	2.60(4)	1.01

^aDue to an error in the 3-in. Bonner ball data for the 30-cm detector location beyond Configuration III.B, the 305-cm data were used to determine both the calculated and the measured attenuation factors.

^bRead: 2.46×10^3 .

Fig. 8.27. Comparison of Spectra Incident on Full Shield in Configuration IV.E (Shield Preceded by Spectrum Modifier Only) and Configuration IV.C (Shield Preceded by Spectrum Modifier Plus ThO_2 Blanket).

One effect examined was the effect of the air between the end of the configuration and the detector. When 12 m of air was included in one-dimensional calculations in slab geometry, the gamma-ray flux in most groups decreased 5 to 10% owing to attenuation of the neutron flux; however, the gamma-ray flux in the 8- to 10-MeV group increased 15 to 20%, with the maximum flux occurring approximately 5 m behind the configuration.

In another one-dimensional calculation (for Configuration II.C), the effect of using a P_8 angular expansion for the cross sections rather than a P_3 expansion was investigated and found to have no effect.

The effect of the concrete blockhouse surrounding the spectrometer was also determined in a one-dimensional calculation in which the blockhouse was modeled as an infinite annular cylinder and the source (the neutron flux calculated for the detector position beyond Configuration II.B) was placed on the axial centerline. The energy group containing the 2.2-MeV hydrogen capture gamma ray increased by 5% and other groups increased less than 1%.

A similar study of gamma-ray production within the NE-213 spectrometer itself indicated that detector-born gamma rays would increase the 1.8- to 2.3-MeV group by 10 to 15% and most other groups by less than 5%.

Fig. 8.28. Full Geometry for Spectral Measurements.

While the above effects were not of sufficient magnitude to account for the large differences in the calculated and measured gamma-ray spectra, one additional effect did appear to largely explain the problem, and this had to do with the collimation of the gamma-ray field incident on the NE-213 detector. As noted in Fig. 8.28, two large concrete collimators were used to restrict the angle of view of the spectrometer to the end of the mockup and the surrounding lithiated paraffin. When background measurements were made only the lithiated paraffin was viewed by the detector; thus subtracting the background measurement from the foreground measurement should have given a net flux coming from the mockup itself.

Throughout the analysis of the experiment an "analytic collimator" was used in the FALSTF calculations to correspond to the experimental collimator. That is, only the flux coming from the end of the mockup itself was considered. When the calculations were repeated without the analytic collimator, the calculated gamma-ray spectrum moved upward toward the measured spectrum while the calculated neutron spectrum was unaffected. A typical result (for Configuration II.B) is shown in Fig. 8.29, and the apparent reason is indicated in Fig. 8.30. That is, the gamma-ray flux appears

Fig. 8.29. Comparison of Measured Neutron and Gamma-Ray Spectra with Calculated Spectra Obtained With and Without an Analytic Collimator (Configuration II.B).

to be channeled through the lithiated paraffin, resulting in a flux exiting from the paraffin that competes with the flux exiting from the mockup itself. Since the angles defined by the analytic collimator in the analysis were precise, the observed disagreements appear to be due to the difficulty in accurately aligning the bulky collimators and shadow shields in the experiment. The overall conclusion is that an experiment designed primarily for neutron measurements may not be well-suited for gamma-ray measurements.

Fig. 8.30. Isoplot Contours for the Gamma-Ray Flux above 1 MeV Emerging from Configuration II.B.

8.5. CONCLUSIONS

By the time the analysis of the GCFR radial shield confirmation experiment had been completed, not only had several important design changes been introduced in the shield region, but work on the GCFR program itself had been suspended. Therefore the conclusions presented here are largely limited to those of generic value.

First, it was found that the neutron transmission through ThO_2 could be predicted to within the same accuracy that the experimental transmission could be measured (5 to 20%). It was also found that neutron transmission through UO_2 was somewhat underpredicted, but not by more than 30%. Comparisons of calculated and measured neutron spectra beyond the two blanket materials showed fairly good agreement; however, the calculated spectrum for ThO_2 was slightly higher than the measured spectrum. When ENDF/B-V cross sections for thorium were substituted for the ENDF/B-IV cross sections used throughout the analysis, the calculated spectrum moved closer to the measured spectrum, especially in the 0.1- to 3-MeV region.

The predicted gamma-ray heating rates for void regions distributed in the ThO_2 and UO_2 blankets agreed with the measured heating rates within the estimated uncertainties of 5 to 15%; however, the agreement for the ThO_2 blanket occurred only after the ENDF/B-V gamma-ray production cross sections for thorium were substituted into the calculation.

In contrast with the heating rates, the gamma-ray energy spectra at locations beyond the blankets were underpredicted -- severely so for the ThO_2 blanket, with no significant improvements when ENDF/B-V cross sections were used. The large underpredictions also occurred beyond the configurations that were extended with shield materials, and a subsequent investigation indicated that the experimental design was poor for gamma-ray spectral measurements outside the configuration. Specifically, it was found that a lithiated paraffin layer surrounding the configuration provided a channel for gamma rays that competed with the gamma rays emerging from the configuration proper.

With respect to the shield configurations, the calculated neutron transmission through the shields generally agreed with the measured transmission to within 15%. It was noted, however, that the attenuating characteristics of laminated shields that contained boron were dramatically affected by the spectral characteristics of the

incident flux. In particular, the attenuation factors for the full radial shield (as determined from Bonner ball responses) were higher by factors of 3 to 6 when the incident neutrons had emerged from the ThO_2 blanket rather than from the spectrum modifier simulating the GCFR core spectrum. The shield was also more effective with a preceding ThO_2 blanket than with a preceding UO_2 blanket (by 5 to 15%).

In summary, then, the GCFR radial shield confirmation experiment and its analysis have shown that a capability exists for accurately predicting (1) the neutron transmission through thick UO_2 and ThO_2 blankets and laminated shields containing steel, graphite, and boronated graphite, (2) the gamma-ray heating rates within the configurations, and (3) the neutron spectra beyond the configurations. It has also confirmed the essentiality of using the proper source spectrum in an experiment. And, finally, it has shown that if an experiment is to include gamma-ray spectral measurements outside the configuration, special additional design considerations will be necessary.

8.6. REFERENCES

1. R. E. Maerker and F. J. Muckenthaler, "Final Report on a Benchmark Experiment for Neutron Transport Through Iron and Stainless Steel," ORNL-4892 (April 1974).
2. R. E. Maerker, C. E. Clifford, and F. J. Muckenthaler, "Neutron Total Cross Section Checks for Iron, Chromium, Nickel, Stainless Steel, Sodium and Carbon," ORNL-5013 (April 1976).
3. D. T. Ingersoll and F. J. Muckenthaler, "Radial Blanket and Shield Experiment," p. 179 ff. In Gas-Cooled Fast Reactor Program Annual Progress Report for Period Ending December 31, 1978, ORNL-5560 (August 1979).
4. D. T. Ingersoll and F. J. Muckenthaler, "Radial Blanket and Shield Experiment," p. 120 ff. In Gas-Cooled Fast Reactor Program Annual Progress Report for Period Ending December 31, 1979, ORNL-5675 (November 1980).
5. D. T. Ingersoll, D. E. Bartine, and F. J. Muckenthaler, "Design of Deep-Penetration Integral Experiment for Thorium Blanket and Radial Shield Mockups," Trans. Am. Nucl. Soc. 32, 643 (1979).

6. D. T. Ingersoll, D. E. Bartine, and F. J. Muckenthaler, "GCFR Radial and Shield Experiment: Objectives, Preamalysis, and Specifications," ORNL/TM-6956 (September 1979).
7. F. J. Muckenthaler, J. L. Hull, and J. J. Manning, "GCFR Radial Blanket and Shield Experiment," ORNL/TM-7237 (December 1980).
8. D. T. Ingersoll and L. R. Williams, "Final Analysis of the GCFR Radial Blanket and Shield Integral Experiment," ORNL-5756 (April 1981).
9. D. T. Ingersoll and L. R. Williams, "Analysis of the ORNL-TSF Radial Blanket and Shield Integral Experiment," Trans. Am. Nucl. 35, 470-472 (1980).
10. D. T. Ingersoll and F. J. Muckenthaler, "Deep Penetration Integral Experiment for a Thorium Blanket Mockup," Nuclear Cross Sections for Technology 594, 122-'25 (1980); Proceedings of International Conference on Nuclear Cross Sections for Technology, University of Tennessee, October 22-26, 1979.

9. GCFR PLENUM SHIELD STUDIES: OUTER RADIAL SHIELD
HETEROGENEITY EXPERIMENT

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- 9.1. Introduction
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- 9.3. Measured Results
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9.1. INTRODUCTION

In the original schedule of GCFR-TSF shielding experiments, it was planned that the series would end with four experiments to validate the design of the system of GCFR plenum shields. However, with the curtailment of the GCFR program, only two of the four experiments were completed, one described in this chapter and another described in Chapter 10. As used in this report, the system of GCFR plenum shields includes both those shields surrounding the upper and lower plenums above and below the core (and specifically identified as plenum shields) plus several other constituent shields.

As it turned out, the two experiments most directly concerned with the shields surrounding the plenum regions were the two that were cancelled. One experiment was to have intercompared how the introduction of various types of orifices in the shields would compromise their shielding effectiveness. Because the helium flowing through the core will enter and leave through the plenum shields, they must be orificed at the location of the helium ducts, and several types of orifices were being considered (herringbone design, chevron design, etc.). The other experiment was to have investigated neutron streaming and backscattering effects within the plenums themselves.

The two experiments that were completed included one identified as the exit shield experiment, which is described in Chapter 10, and another identified as the outer radial shield heterogeneity experiment, which is the experiment described in this chapter. The outer radial shield is considered to be part of the plenum shield system because it lies between the upper and lower plenum shields and is contiguous with them. Tests of the outer shield were included in the experiment described in Chapter 8 and the heterogeneity experiment reported here is actually a refinement of that part of the earlier experiment.

As described in Chapter 8, the outer radial shield included in the CSC.1 design was specified to be a permanent shield consisting of 22.6-cm-thick blocks of graphite bounded on each side by 5.1 cm of boronated graphite and clad with 5.1 cm of stainless steel. To simplify the geometry in the experiment, the shield was mocked up with solid layers of graphite, boronated graphite, and steel. In the actual design, however, the outer radial shield consists of several interwoven pieces enclosed in stainless steel envelopes (see Fig. 9.1.). With this design the question arose as to whether excessive neutron streaming might occur between abutting pieces, and it was deemed necessary to study neutron

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Fig. 9.1. Cross Section Through GCFR Midplane. (See stepped slits through outer radial shield.)

streaming through shield heterogeneities typical of those that might be present in the actual shield. In addition, a benchmark experiment was needed to further check the analytical methods used to design the outer radial shield. The experiment described here, performed in the summer of 1980 and necessarily abbreviated because of the suspension of the GCFR program, was designed to provide the required information. The various configurations used in the experiment are described in Section 9.2, and the experimental data are presented in Section 9.3. Section 9.4 then compares the measured results with the calculated results obtained from a brief analysis. (The experiment has also been described in ref. 1.)

9.2. THE EXPERIMENT

The basic experimental configuration consisted of a 40.8-cm thickness of graphite separated by straight or stepped vertical slits with and without carbon steel linings.

The primary criterion of the design was to approximate the slit geometry and overall dimensions of the CSC-1 outer radial shield without incurring the expense that would be associated with an exact mockup. Thus the 43-cm-thick multiregion radial shield of the CSC-1 design that includes graphite, boronated graphite, and stainless steel layers was mocked up with slabs of graphite alone (to a total thickness of 40.8 cm) and the stainless steel envelopes surrounding each block of the design shield were mocked up with 5.1-cm-thick "sheets" of carbon steel lining only the slits. The graphite slabs were each 10.16 cm on a side and 121.9 cm long.

The experiment was performed in four phases. In the first (Configurations I) two stacks of graphite slabs were separated to form a vertical slit at the TSR-II beam centerline (see photograph of mockup in Fig. 9.2). The total height of the stacks was 162.6 cm, but 20.3-cm thicknesses at the top and bottom were offset, reducing the height of the slit to 121.9 cm. The beam axis was 55.9 cm from the bottom of the slit, and the slit width was varied from 0 to 10.2 cm (widths = 0, 0.635, 1.27, 2.54, 5.08, 7.62, and 10.2 cm). A sketch of the configurations through the horizontal midplane is shown in Fig. 9.3.

In the second phase of the experiment (Configurations II), the straight slit was lined with carbon steel (see Fig. 9.4), and the slits were open at the top; that is, offset graphite slabs were not used at the top and bottom of the configuration. Again the slit width was varied (widths = 0, 0.635, 1.27, and 2.54 cm).

For Configurations III two additional straight slits were introduced into the mockup by moving each of the two sections of graphite away from the steel liners (see Figs. 9.5 and 9.6). In Configuration III.A, all three slits were 0.635 cm wide, but in Configurations III.B and III.C the outer slits were one-half the width of the center slit, which was 1.27 and 2.54 cm in III.B and III.C respectively.

Configurations IV contained a single stepped slit, which was formed by introducing small sections of graphite into the mockup as shown in Fig. 9.7. The first leg of the slit was maintained at the TSR-II beam axis and the third leg was maintained at a radial distance of 30.5 cm from the beam axis. In each configuration the length of the first

Fig. 9.2. Typical Mockup of the Graphite Shield with Single Slit: Configurations I.B, I.C, I.D, I.E, I.F, I.G.

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Fig. 9.3. Mockup of Outer Radial Shield with Single Slits: Configurations I.A - I.H.

Fig. 9.4. Mockup of Outer Radial Shield with Iron-Lined Single Slit: Configurations II.A - II.D.

Fig. 9.5. Mockup of Outer Radial Shield with Iron Liners and Three Slits: Configurations III.A - III.C.

Fig. 9.6. Typical Mockup of Graphite Shield with Iron Liners - Three Sills: Configurations III.A, III.B, III.C.

Fig. 9.7. Mockup of Outer Radial Shield with Single Unlined Stepped Sills: Configurations IV.A - IV.D.

leg was 20.3 cm minus one-half of the slit width and the length of the third leg was 20.3 cm plus one-half of the slit width. In all cases the same slit width was maintained in all three legs (0.635, 2.54, 5.08, or 10.16 cm), and, as was the case for Configurations 1, 20.3-cm thicknesses of graphite were offset at the top and bottom of the mockup.

It will be noted that each configuration is preceded by a spectrum modifier consisting of 10.2 cm of iron, 2.54 cm of boron, and 10.2 cm of UO_2 surrounded by lithiated paraffin. Preamalysis calculations had shown that this arrangement of materials would convert the energy spectrum of the TSR-II beam to a spectrum typical of that incident on the GCFR outer radial shield. (Note: The UO_2 "slab" is the same slab described on page 8-9 in Section 8.)

9.3. MEASURED RESULTS

The experiment began with measurements behind the spectrum modifier. However, because agreement between measurements and calculations for similar spectrum modifiers had repeatedly verified the analytical methods for such configurations, and because the experiment was to be abbreviated, the measurements were limited to 2-, 5-, and 10-in. Bonner ball measurements at two centerline locations behind the spectrum modifier. Measurements beyond the slit configurations were also limited to the three Bonner ball detectors plus the Hornyak button. Horizontal traverse measurements were made with the Hornyak button and the 2-in. Bonner ball directly behind all the configurations and with all the detectors at a distance of 30.5 cm beyond the configurations.

9.3.1. Measurements for Configurations 1: Single Slits

As described above, the assemblies measured in the first phase of the experiment consisted of a 40.8-cm thickness of graphite that was positioned behind a spectrum modifier and contained a single straight slit (see Figs. 9.2 and 9.3). The configurations differed from one another in the slit width, with seven different slit widths being measured. This phase of the experiment also included centerline measurements behind the spectrum modifier alone, the results of which are shown in Table 9.1.

Table 9.1. Measured Bonner Ball Responses on Axis 30.5 and 305 cm Beyond Spectrum Modifier (Configuration 1.A)

Bonner Ball	Response (counts/s-W)	
	At 30.5 cm	At 305 cm
2-in.	1.15×10^1	7.5×10^{-1}
5-in.	9.60×10^3	3.809×10^2
10-in.	4.61×10^3	1.996×10^2

Measurements along radial traverses were made behind each of the single-slit assemblies with the Hornyak button at a distance of 1.6 cm and with the 2-in. Bonner ball at a distance of 3 cm. Additional measurements with the 2-in. Bonner ball were made behind two configurations at a distance of 6 cm, and short radial traverses were made with both the 2-in. Bonner ball and the Hornyak button, as well as with the 5-in. and 10-in. Bonner balls, at a distance of 30.5 cm. The primary intent at 30.5 cm was to obtain centerline values, with the short traverses supporting the centerline measurements.

The 2-in. Bonner ball responses and the Hornyak button responses obtained directly behind the configurations are plotted in Figs. 9.8 and 9.9 respectively, and selected detector responses from all the traverses are shown in Tables 9.2 and 9.3. It will be noted that the Hornyak button data are limited to slit widths of 0 and 1.27 cm. This is because water became trapped between the graphite pieces during the experiment, negating most of the measurements, and the remaining time was insufficient to repeat all the measurements.

The curves in Fig. 9.8 all show a broad plateau across the slit, with relatively small differences between the various slit widths. Since the 2-in. Bonner ball is primarily sensitive to low-energy neutrons, this indicates that much of the low-energy flux is due to neutrons that penetrate through the graphite rather than through the slit. As would be expected, however, the Hornyak button measurements shown in Fig. 9.9 peak at the slit location, indicating considerable streaming of high-energy neutrons through the slit. The Hornyak button centerline flux for the 1.27-cm slit is higher by a factor of 2.8 than the centerline flux for the no-slit case. On the basis of the Hornyak button "wet graphite" data, which are not included here but which do not differ greatly from the "dry graphite" data (see ref. 1), it appears that the increase in the fast-neutron flux would be about a factor of 13 for the 10.2-cm slit.

9.3.2. Measurements for Configurations II: Iron-Lined Single Slits

Configurations II represented two abutting pieces of the radial shield (graphite) in stainless steel envelopes. Thus the point of interest was the impact of the iron pieces on the neutron streaming through the slit separating them. The slit widths measured were 0, 0.635, 1.27, and 2.54 cm.

Radial traverse measurements were made directly behind each of the four configurations with the 2-in. Bonner ball and with the Hornyak button. Radial traverse measurements with the Hornyak button were also made at 30.5 cm, and near-centerline measurements were made with the 2-, 5-, and 10-in. Bonner balls at 30.5 cm. The results obtained directly behind the configurations are plotted in Figs. 9.10 and 9.11, and typical responses from all the measurements are given in Tables 9.4 and 9.5.

Figure 9.10 shows a pronounced depression of the low-energy flux directly behind the iron, with an increase in the neutron transmission at the slit position occurring only for the widest slit (2.54 cm). As shown in Fig. 9.11, the high-energy flux is also depressed behind the iron relative to the flux behind the graphite; however, in all cases it peaks sharply at the position of the slit, again indicating the streaming of fast neutrons through the slits.

9.3.3. Measurements for Configurations III: Three-Slit Assemblies

The measurements behind Configurations III, which had slits between the two iron liners and also between the iron and graphite, were made at the same locations as those behind Configurations II. The results are given in Figs. 9.12 and 9.13 and in Tables 9.6 and 9.7.

The shapes and magnitudes of the curves shown in Fig. 9.12 for the 2-in. Bonner ball measurements directly behind the configurations are very similar to those shown in Fig. 9.10. That is, the low-energy flux is significantly depressed behind the iron and the values of the slight peaks at the centerline are approximately equal for the same slit widths. No peaking of the flux occurs opposite the two outer slits of Configurations III; however, the curves do rise more sharply near the iron-graphite interfaces than do the curves for Configurations II.

Table 9.2. Measured Bonner Ball Responses Along Radial Traverses Beyond Configurations I (Single Silt Through Graphite)

Bonner Ball	Distance from Axis (cm)	Response (counts/s-W) ^a						
		I.B, 0.635 cm	I.C, 1.27 cm	I.D, 2.54 cm	I.E, 5.08 cm	I.F, 7.62 cm	I.G, 10.16 cm	I.H, 0 cm
3 cm Beyond Configuration								
2-in.	20	8.03(1) ^b	7.99(1)	8.25(1)	8.52(1)	8.64(1)	8.81(1)	7.76(1)
	10	8.82(1)	8.84(1)	9.46(1)	1.02(2)	1.09(2)	1.15(2)	8.44(1)
	5	8.99(1)	9.44(1)	1.06(2)	1.29(2)	1.48(2)	1.59(2)	8.53(1)
	2.5	9.37(1)	—	—	—	—	—	—
	0	9.78(1)	1.05(2)	1.35(2)	1.64(2)	1.75(2)	1.76(2)	8.54(1)
6 cm Beyond Configuration								
2-in.	20	—	—	—	—	—	8.59(1)	7.18(1)
	10	—	—	—	—	—	1.17(2)	7.74(1)
	5	—	—	—	—	—	1.39(2)	3.01(1)
	0	—	—	—	—	—	1.48(2)	1.93(1)
30.5 cm Beyond Configuration								
2-in.	5	5.06(1)	5.07(1)	—	6.27(1)	6.53(1)	6.59(1)	4.82(1)
	2.5	5.13(1)	5.25(1)	—	6.26(1)	—	6.67(1)	—
	0	5.14(1)	5.27(1)	—	6.27(1)	6.47(1)	6.65(1)	4.89(1)
5-in.	5	6.95(2)	—	9.43(2)	1.43(3)	1.89(3)	2.31(3)	6.41(2)
	2.5	7.14(2)	—	1.04(3)	1.53(3)	2.00(3)	—	—
	0	7.20(2)	—	1.06(3)	1.58(3)	2.08(3)	2.44(3)	6.44(2)
10-in.	5	1.50(2)	1.65(2)	2.29(2)	3.93(2)	5.68(2)	7.52(2)	1.36(2)
	2.5	1.52(2)	1.69(2)	2.42(2)	4.21(2)	5.94(2)	—	—
	0	1.53(2)	1.68(2)	2.43(2)	4.37(2)	6.11(2)	8.06(2)	1.34(2)

^aResponses are averages of measurements at equidistances from the axis.^bRead: 8.06×10^1 .

Comparing the Hornyak button traverses of Fig. 9.13 with those in Fig. 9.11 shows that the high-energy centerline fluxes in Configurations II and III also are approximately the same for the same silt widths. However, in this case the presence of the outer slits results in additional neutron streaming through the configurations. (The lack of symmetry in the data for the outer voids could be due to several things: slight differences in void widths, in detector location, etc.)

Table 9.3. Measured Hornyak Button Responses Beyond Configurations I (Single Slit Through Graphite)

Distance from Axis (cm)	Dose Rate (ergs/g-s-kW) ^a	
	I.C, 1.27 cm	I.H, 0 cm
1.6 cm Beyond Configuration		
20	1.62(-2) ^b	1.25(-3)
10	1.82(-2)	1.31(-2)
5	2.00(-2)	1.35(-2)
2	2.23(-2)	—
0.5	3.67(-2)	—
0	3.81(-2)	1.36(-2)
30.5 cm Beyond Configuration		
20	9.78(-3)	—
10	1.05(-2)	9.82(-3)
5	1.11(-2)	9.92(-3)
2	1.30(-2)	—
0.5	1.58(-2)	—
0	1.67(-2)	1.01(-2)

^aResponses are averages of measurements at equidistances from the axis.^bRead: 1.62×10^{-2} .

9.3.4. Measurements for Configurations IV: Offset Slits

The measurements beyond the stepped slits of Configurations IV were made with the same detectors at the same distances beyond the experimental assemblies as in Configurations II and III. In Configurations IV, however, the radial traverses were referenced from the centerline of the exit slit and not from the centerline of the reactor beam. The results of these measurements are shown in Figs. 9.14 and 9.15 and in Tables 9.8 and 9.9.

In both Fig. 9.14 and 9.15 the peaks in the curves at the exit slit position are rather broad, indicating that neutrons are scattering against the walls of the slit and exiting in various directions. The broadening increases with increasing slit width, with the right sides of the curves falling rapidly as the distance from the exit slit increases. By contrast, the left sides of the curves tend to level off, with only the

Fig. 9.8. Measured 2-in. Bonner Ball Responses Directly Behind Single-Slit Assemblies (Configurations 1.B - 1.H).

Fig. 9.9. Measured Hornyak Button Response Directly Behind Single-Slit Assemblies (Configurations 1.C and 1.H).

Fig. 9.10. Measured 2-In. Bonner Ball Responses Directly Behind Iron-Lined Single-Slit Assemblies (Configurations 11.A - 11.D).

Fig. 9.11. Measured Hornyak Button Responses Directly Behind Iron-Lined Single-Slit Assemblies (Configurations 11.A - 11.D).

Fig. 9.12. Measured 2-in. Bonner Ball Responses Directly Behind Three-Slit Assemblies (Configurations III.B and III.C, Compared with II.D).

Fig. 9.13. Measured Hornyak Button Responses Directly Behind Three-Slit Assemblies (Configurations III.B and III.C).

Fig. 9.14. Measured 2-in. Bonner Ball Responses Directly Behind Stepped-Slit Assemblies (Configurations IV.A - IV.D).

Fig. 9.15. Measured Hornyak Button Responses Directly Behind Stepped-Slit Assemblies (Configurations IV.A - IV.D).

Table 9.4. Measured Bonner Ball Responses Beyond Configurations II
(Iron-Lined Single Slit Through Graphite)

Bonner Ball	Distance from Axis (cm)	Response (counts/s-M) ^a			
		II.A, 0.635 cm	II.B, 1.27 cm	II.C, 2.54 cm	II.D, 0 cm
3 cm Beyond Configuration					
2-in.	20	7.36(1) ^b	7.54(1)	7.31(1)	7.25(1)
	10	6.44(1)	6.69(1)	6.34(1)	6.38(1)
	5	4.92(1)	4.98(1)	4.92(1)	4.78(1)
	2	4.19(1)	4.56(1)	4.98(1)	3.85(1)
	0	4.08(1)	4.72(1)	5.35(1)	3.62(1)
30.5 cm Beyond Configuration					
2-in.	5	3.89(1)	3.90(1)	4.03(1)	3.77(1)
	2	3.89(1)	3.94(1)	4.09(1)	3.76(1)
	0	3.95(1)	4.02(1)	4.05(1)	3.73(1)
5-in.	5	7.18(2)	7.87(2)	9.46(2)	6.79(2)
	2	7.32(2)	6.24(2)	1.02(3)	6.81(2)
	0	7.32(2)	8.36(2)	1.03(3)	6.66(2)
10-in.	5	1.69(2)	1.90(2)	2.49(2)	1.53(2)
	2	1.72(2)	1.94(2)	2.58(2)	1.56(2)
	0	1.72(2)	1.94(2)	2.64(2)	1.55(2)

^aResponses are averages of measurements at equidistances from the axis.

^bRead: 7.36×10^1 .

high-energy fluxes in Fig. 9.15 showing a slight increase opposite the entry slit. This indicates that neutron interactions within the graphite negate the effect of neutrons directed outward from the entry slit.

9.4. COMPARISON OF MEASURED AND CALCULATED RESULTS

As noted earlier, the suspension of the U.S. GCFR program precluded a complete analysis of the experiment. However, some calculations were performed with both the discrete ordinates code DOT and the Monte Carlo code MORSE for selected locations of the 2- and 5-in. Bonner balls beyond the configurations. Typical results are compared with the corresponding measurements in Tables 9.10 and 9.11.

Table 9.5. Measured Hornyak Button Responses Beyond Configurations II (Iron-Lined Single Slit Through Graphite)

Distance from Axis (cm)	Dose Rate (ergs/g-s-kW) ^a			
	II.A, 0.635 cm	II.B, 1.27 cm	II.C, 2.54 cm	II.D, 0 cm
1.6 cm Beyond Configuration				
20	1.19(-2) ^b	1.20(-2)	1.15(-2)	1.21(-2)
10	1.06(-2)	1.10(-2)	1.12(-2)	1.07(-2)
5	1.01(-2)	1.09(-2)	1.15(-2)	9.88(-3)
2.5	9.33(-3)	1.10(-2)	—	—
0.5	1.09(-2)	2.10(-2)	3.63(-2)	—
0	1.43(-2)	2.24(-2)	3.67(-2)	8.03(-3)
30.5 cm Beyond Configuration				
20	6.64(-3)	6.86(-3)	6.83(-3)	6.78(-3)
10	6.85(-3)	7.00(-3)	—	7.14(-3)
5	7.03(-3)	7.55(-3)	8.33(-3)	7.35(-3)
2.5	7.21(-3)	—	—	—
0.5	8.89(-3)	—	—	—
0	9.31(-3)	1.24(-2)	1.69(-2)	7.28(-3)

^aResponses are averages of measurements at equidistances from the axis.

^bRead 1.19×10^{-2} .

The DOT results and the measurements agree within 20%, with some notable exceptions; in particular, the calculations yield significantly higher results for the iron-lined straight slit (Configuration II). In general, the MORSE results and the measured results are in poor agreement, but the curtailment of the GCFR program did not permit a detailed investigation of the reasons why.

9.5. REFERENCE

1. F. J. Muckenthaler, J. L. Hull, and J. J. Manning, "GCFR Plenum Shield Design - Shield Heterogeneity Experiment," ORNL/TM-7714 (May 1981).

Table 9.6. Measured Bonner Ball Responses Beyond Configurations III
(Three-Slit Assemblies)

Bonner Ball	Distance from Axis (cm)	Response (counts/s-M) ^a		
		III.B, 0.635 cm, 1.27 cm, 0.635 cm		III.C, 1.27 cm, 2.54 cm, 1.27 cm
3 cm Beyond Configuration				
2-in.	20	7.35(1) ^b		7.61(1)
	10	6.48(1)		6.97(1)
	5	5.20(1)		5.78(1)
	2	4.37(1)		5.34(1)
	0	4.44(1)		5.78(1)
30.5 cm Beyond Configuration				
2-in.	5	3.94(1)		4.37(1)
	2	4.02(1)		4.41(1)
	0	4.03(1)		—
5-in.	5	—		1.16(3)
	2.5	—		1.20(3)
	0	8.81(2)		1.18(3)
10-in.	5	2.07(2)		3.08(2)
	2.5	—		3.16(2)
	0	2.13(2)		3.17(2)

^aResponses are averages of measurements at equidistances from the axis.

^bRead: 7.35×10^1 .

Table 9.7. Measured Hornyak Button Responses Beyond Configurations III
(Three-Slit Assemblies)

Distance from Axis (cm)	Dose Rate (ergs/g·s·kW) ^a	
	III.B, 0.635 cm, 1.27 cm, 0.635 cm	III.C, 1.27 cm, 2.54 cm, 1.27 cm
1.6 cm Beyond Configuration		
20	1.30(-2) ^b	1.33(-2)
10	1.33(-2)	1.58(-2)
5	1.32(-2)	1.68(-2)
2	1.31(-2)	2.06(-2)
0.5	2.45(-2)	4.28(-2)
0	2.58(-2)	4.31(-2)
30.5 cm Beyond Configuration		
20	6.85(-3)	8.99(-3)
10	7.36(-3)	1.10(-2)
5	8.20(-3)	1.27(-2)
2	9.61(-3)	1.92(-2)
0.5	1.22(-2)	—
0	1.24(-2)	2.16(-2)

^aResponses are averages of measurements at equidistances from the axis.

^bRead: 1.30×10^{-2} .

Table 9.8. Measured Bonner Ball Responses Beyond Configurations IV
(Stepped Slit Assemblies)

Bonner Ball	Distance from Exit Slit Axis (cm)	Response (counts/s-M)				
		I.H, ^a 0 cm	IV.A, 0.635 cm	IV.B, 2.54 cm	IV.C, 5.08 cm	IV.D, 10.2 cm
3 cm Beyond Configuration						
2-in.	20N	4.35(1) ^b	4.34(1)	4.58(1)	5.02(1)	6.04(1)
	10N	5.70(1)	5.81(1)	6.55(1)	7.63(1)	9.99(1)
	5N	6.31(1)	6.45(1)	8.08(1)	1.10(2)	1.53(2)
	2N	—	7.11(1)	1.01(2)	1.44(2)	1.67(2)
	0	6.75(1)	7.63(1)	1.13(2)	1.54(2)	1.73(2)
	2S	—	7.57(1)	1.05(2)	1.45(2)	1.68(2)
	5S	7.43(1)	7.62(1)	9.21(1)	1.15(2)	1.56(2)
	10S	7.79(1)	7.99(1)	—	1.04(2)	—
	20S	8.42(1)	8.44(1)	—	—	1.10(2)
30.5 cm Beyond Configuration						
2-in.	5N	3.81(1)	3.95(1)	4.80(1)	5.80(1)	—
	2.5N	—	4.12(1)	—	5.97(1)	—
	2N	3.89(1)	—	—	—	—
	0	4.03(1)	4.27(1)	5.15(1)	6.11(1)	6.91(1)
	2S	4.09(1)	—	—	—	—
	2.5S	—	4.36(1)	—	6.23(1)	—
	5S	4.27(1)	4.47(1)	5.23(1)	6.34(1)	—
5-in.	5N	5.13(2)	5.33(2)	7.23(2)	1.04(3)	—
	2.5N	—	5.79(2)	—	1.11(3)	—
	2N	5.19(2)	—	—	—	—
	0	5.42(2)	5.92(2)	7.89(2)	1.09(3)	1.85(3)
	2S	5.51(2)	—	—	—	—
	2.5S	—	6.11(2)	—	—	—
	5S	5.59(2)	6.01(2)	8.06(2)	1.16(3)	—
10-in.	5N	1.05(2)	1.13(2)	1.54(2)	2.02(2)	—
	2.5N	—	1.19(2)	—	2.18(2)	—
	2N	1.11(2)	—	—	—	—
	0	1.12(2)	1.21(2)	1.68(2)	2.30(2)	4.75(2)
	2S	1.15(2)	—	—	—	—
	2.5S	—	1.24(2)	—	2.39(2)	—
	5S	1.20(2)	1.27(2)	1.73(2)	2.45(2)	—

^aSingle-slit configuration with 0-cm slit width; measurements referenced from centerline of exit slit in Configurations IV.

^bRead: 4.35×10^1 .

Table 9.9. Measured Hornyak Button Responses Beyond Configurations IV (Stepped Slit Assemblies)

Distance from Exit Slit Axis (cm)	Dose Rate (ergs/g.s.kW)			
	IV.A, 0.635 cm	IV.B, 2.54 cm	IV.C, 5.08 cm	IV.D, 11.2 cm
1.6 cm Beyond Configuration				
20N	9.06(-3) ^a	9.17(-3)	1.07(-2)	1.44(-2)
10N	1.25(-2)	1.37(-2)	1.88(-2)	3.67(-2)
5N	1.46(-2)	1.81(-2)	3.03(-2)	8.42(-2)
2N	1.62(-2)	2.36(-2)	4.83(-2)	8.72(-2)
0	1.82(-2)	2.86(-2)	4.78(-2)	8.44(-2)
2S	1.66(-2)	2.39(-2)	4.19(-2)	7.78(-2)
5S	—	2.06(-2)	2.81(-2)	6.08(-2)
10S	—	—	2.64(-2)	3.64(-2)
20S	—	2.19(-2)	—	—
30.5 cm Beyond Configuration				
20N	6.61(-3)	7.56(-3)	1.03(-2)	—
10N	8.17(-3)	9.67(-3)	1.51(-2)	2.78(-2) ^b
5N	8.69(-3)	1.12(-2)	1.71(-2)	2.97(-2) ^c
2.5N	—	1.19(-2)	1.80(-2)	3.03(-2) ^d
2N	—	—	—	2.97(-2)
0	9.64(-3)	1.21(-2)	1.83(-2)	2.92(-2)
2.5S	—	1.23(-2)	1.80(-2)	2.86(-2) ^d
5S	9.78(-3)	1.20(-2)	1.76(-2)	2.74(-2) ^c
10S	1.02(-2)	1.20(-2)	1.64(-2)	—
20S	1.08(-2)	1.23(-2)	1.60(-2)	2.08(-2) ^e

^aRead: 9.06×10^{-3} .^bAt 9 cm.^cAt 6 cm.^dAt 3 cm.^eAt 21 cm.

Table 9.10. Comparison of DOT-Calculated and Measured Bonner Ball Responses 30.5 cm Beyond Various Slit Configurations

Detector Distance from Slit Centerline* (cm)	Detector Response (counts/s.W)					
	2-in. Bonner Ball			5-in. Bonner Ball		
	Calc.	Meas.	Calc./Meas.	Calc.	Meas.	Calc./Meas.
Configuration I.E, Single Slit (5.08 cm Wide)						
0.0	65.0	62.7	1.04	1400	1580	0.88
2.5	68.0	62.6	1.09	1600	1530	1.05
5.0	80.0	62.7	1.27	2000	1430	1.40
Configuration II.C, Iron-Lined Single Slit (2.54 cm Wide)						
0.0	43.0	40.5	1.06	1065	1030	1.03
2.0	45.0	40.9	1.10	1090	1020	1.07
5.0	46.4	40.3	1.15	1290	946	1.36
Configuration III.B, Three Slits (Widths = 0.635 cm, 1.27 cm, 0.635 cm)						
0.0	47.0	40.3	1.17	1000	881	1.14
2.0	45.0	40.2	1.12			
3.0				980	856	1.14
5.0	45.0	39.4	1.14			
Configuration IV.C, Offset Slit (5.08 cm Wide)						
0.0	65.0	61.1	1.06	1167	1090	1.07
2.5	69.2	62.3	1.11	1283	1110	1.16
5.0	75.0	63.4	1.18	1467	1160	1.26

*For Configurations I.E, II.C, and III.B, slit centerline is in plane of reactor beam; for Configuration IV.C, the slit centerline is that of the exit slit, which is 30.5 cm from beam centerline.

Table 9.11. Comparison of MORSE-Calculated and Measured Bonner Ball Responses 30.5 cm Beyond Various Slit Configurations

Detector Distance from Slit Centerline* (cm)	Detector Response (counts/s-#)					
	2-in. Bonner Ball			5-in. Bonner Ball		
	Calc.	Meas.	Calc./Meas.	Calc.	Meas.	Calc./Meas.
Configuration I.E, Single Slit (5.08 cm Wide)						
0.0	64.2	62.7	1.02	1988	1580	1.26
2.5	65.2	62.6	1.05	2012	1530	1.32
5.0	65.9	62.7	1.05	1752	1430	1.23
Configuration II.C, Iron-Lined Single Slit (2.54 cm Wide)						
0.0	9.74	40.5	0.242	878	1030	0.852
2.0		40.9		732	1020	0.718
5.0	5.61	40.3	0.139	506	946	0.536
Configuration III.B, Three Slits (Widths = 0.635 cm, 1.27 cm, 0.635 cm)						
0.0	21.0	40.3	0.52	669	881	0.76
2.0	17.3	40.2	0.43			
3.0				425	856	0.50
5.0	15.6	39.4	0.39			
Configuration IV.C, Offset Slit (5.08 cm Wide)						
0.0	90.7	61.1	1.48	1199	1090	1.10
2.5	89.2	62.3	1.43	1208	1110	1.09
5.0	89.9	63.4	1.42	1203	1160	1.04

*For Configurations I.E, II.C, and III.B, slit centerline is in plane of reactor beam; for Configuration IV.C, the slit centerline is that of the exit slit, which is 30.5 cm from beam centerline.

10. GCFR PLENUM SHIELD STUDIES: EXIT SHIELD EXPERIMENT

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10.1. INTRODUCTION

The sixth and final experiment performed at the TSF as part of the five-year Integral shield testing program for the GCFR was for that portion of the plenum shield identified as the "exit shield," so called because it would provide an exit path for the helium coolant from the core into the plenum, while at the same time providing protection for the components within the plenum. By the time this experiment was performed, in 1980, the upflow GCFR design had been decided upon, and therefore the exit shield was to be at the top of the core and the grid-plate shield described in Chapter 7 was to be at the bottom of the core. As a result, the components of concern were those in the upper plenum.

Like the grid-plate shield, the exit shield was designed as an integral part of the fuel element, the grid-plate shield being incorporated into one end of the element and the exit shield being incorporated into the opposite end. Both shields consisted of B_4C , but whereas the grid-plate shield was concentrated along the walls of the fuel element, leaving an axial coolant path, the exit shield was concentrated on the fuel element axis, leaving a surrounding coolant path (see Fig. 2.3 in Chapter 2).

The purpose of the TSF experiment was to determine the effectiveness of the exit shield in reducing the radiation damage to components in the upper plenum and, through analysis of the experiment, to provide a test of the calculational methods used to develop the design. In addition, the effect of a control rod penetration through the shield was studied. The experiment and analysis are summarized below, both having been presented in detail in refs. 1 and 2.

10.2. THE EXPERIMENT

Up to a point, the experimental design for the exit shield experiment was based on the design described in Chapter 7 for the grid-plate shield experiment. As was the case in the grid-plate shield experiment, it was important in the exit shield experiment that the spectrum of neutrons incident on the shield mockup represent the neutrons that would be emerging from the end of the GCFR core and axial blanket. Although the grid-plate shield and the exit shield would be at opposite ends of the core, the spectra of neutrons incident on the shields could be expected to be the same and therefore the mockup of the

experimental source could be the same in the two cases. Thus the first configuration tested in the exit shield experiment (shown in Fig. 10.1) was basically the same as the fuel pin configurations used in the grid-plate shield experiment; that is, it used the same fuel pin subassemblies detailed in Figs. 7.2 - 7.5 of Chapter 7. The major difference was that in the grid-plate shield experiment both the number of pins and the spacing between fuel subassemblies were varied, while in the exit shield experiment only the 91-pin subassemblies were used and the spacing between the subassemblies was maintained at 0.635 cm ($\pm 5\%$) throughout.

As shown in Table 10.1 the mockup with the fuel-pin subassemblies (identified as Configuration 1.A) was the first of 10 configurations included in the exit shield experiment. Since the fuel pin subassemblies had been thoroughly studied in the grid-plate shield experiment, in this experiment the measurements beyond Configuration 1.A were limited to those required to ensure that the magnitude and energy spectrum of the neutron flux were well known.

Fig. 10.1. Mockup of Shadow Shield and Fuel Pin Subassemblies for Exit Shield Experiment (Configuration 1.A).

Table 10.1. Summary of Configurations in GCFR Exit Shield Experiment

Configuration	Description	Measurements ^a
I.A ^b	7 Fuel pin subassemblies	BB, HB, S
II.A ^b	7 Fuel pin subassemblies + seven exit shield subassemblies	BB, HB, S
III.A	6 Fuel pin subassemblies + 6 exit shield subassemblies + unshielded control rod subassembly (without control rod mockup)	BB, HB
III.B	Conf. III.A + fully inserted control rod mockup	BB, HB
III.C ^b	Conf. III.A + 50% inserted control rod mockup	BB, HB
III.D	Conf. III.A + fully withdrawn control rod mockup	BB, HB
IV.A	6 Fuel pin subassemblies + 6 exit shield subassemblies + shielded control rod subassembly (without control rod mockup)	BB, HB
IV.B	Conf. IV.A + fully inserted control rod mockup	BB, HB
IV.C	Conf. IV.A + 50% inserted control rod mockup	BB, HB
IV.D	Conf. IV.A + fully withdrawn control rod mockup	BB, HB

^aBB = Bonner balls; HB = Hornyak button; S = spectrometers.

^bConfigurations for which analysis was performed.

For the second configuration (II.A) seven exit shield subassemblies were added behind the fuel subassemblies. The design of the exit shield subassemblies is shown in Figs. 10.2 and 10.3. Centered on the axis of each subassembly were two B_4C cones of unequal length that were joined at their large ends. The cones were surrounded by inner cylindrical liners, with the liner around the larger cone having a constant inside diameter and the liner around the smaller cone having the same taper as the cone itself. The outside wall of each subassembly was hexagonal in cross section with essentially the same dimensions as the fuel pin subassemblies, and the space between the liner and the outside wall was filled with B_4C that varied in density from 1.34 to 1.37 g/cc. The density of the B_4C in the larger cone varied from 1.28 to 1.32 g/cc, and that in the smaller cone from 1.29 to 1.32 g/cc. (Note: So that the exit shield subassemblies would match the fuel pin subassemblies, they were scaled down to 77% of the volume of the shields designed for the GCFR, except for the iron wall thicknesses, which were the same as in the fuel pin subassemblies.)

Fig. 10.2. Schematic of Exit Shield Subassembly.

Fig. 10.3. Schematic of B_4C Cones In Exit Shield Subassembly.

When inserted into the experimental configuration, the ends of the exit shield subassemblies containing the smaller cones were positioned against the ends of the fuel subassemblies. In all cases the axes of the fuel subassemblies were aligned with the axes of the shield subassemblies so that the neutron streaming paths between the subassemblies would extend uninterrupted through the fuel and shield sections. A sketch of the resulting configuration (II.A), which is surrounded by concrete lined with 0.95-cm-thick iron, is shown in Fig. 10.4.

For the remaining configurations in the experiment the central fuel and exit shield subassemblies were replaced with a control rod subassembly of the same outside dimensions. A sketch of the subassembly housing is shown in Fig. 10.5. As noted, the outer wall was hexagonal in cross section while the inner liner was cylindrical. The housing was fabricated in two sections, one 82 cm long, and the other 128.5 cm long. In the shorter section the space between the outside wall and the liner was always empty, while in the longer section the space was sometimes filled with B_4C at a density of 1.39 g/cc. (Actually, two long sections were fabricated, one with and one without the B_4C filling.)

The first experimental configuration containing the control rod subassembly housing, but not the mockup of the control rod itself, was identified as Configuration III.A. In this configuration the "unshielded" housing was used; that is, the long section did not contain B_4C between the outer wall and the liner. A sketch of the configuration is presented in Fig. 10.6 and end views are shown in Figs. 10.7 and 10.8. The aim was to maintain a constant spacing of 0.65 cm between subassemblies, but as noted in Fig. 10.7, this was not accomplished. Also, because of the compact spacing of the subassemblies, a relatively large void existed between the array and the iron liner of the surrounding concrete.

Configurations III.B, III.C, and III.D consisted of Configuration III.A with the control rod mockup positioned at various locations within the control rod subassembly. The mockup itself consisted of two sections, each 60 cm long and 15.9 cm in diameter and each filled with B_4C at a density of 1.32 g/cc (see bottom sketch in Fig. 10.9). To the control rod was attached a control-rod follower, also depicted in Fig. 10.9 and also fabricated in two sections and filled with B_4C (at densities of 1.30 and 1.31 g/cc in the smaller and larger section, respectively). With the control rod and control rod follower each divided into sections, various combinations of the two could be used without any portion extending out from the end of the configuration.

Fig. 10.4. Experimental Configuration Consisting of Seven Fuel Pin Subassemblies and Seven Exit Shield Subassemblies (Configuration II.A).

Fig. 10.5. Sketch of Control Rod Subassembly Housing.

Fig. 10.6. Experimental Configuration Consisting of Six Fuel Pin Subassemblies and Six Exit Shield Subassemblies with Unshielded Central Control Rod Subassembly Housing.

Fig. 10.7. Measured Void Spacings Between Six Exit Shield Subassemblies and Central Control Rod Subassembly Housing.

Fig. 10.8. Mockup of Exit Shield (Six Subassemblies) with Central Control Rod Subassembly Housing.

Fig. 10.9. Mockup of Control Rod (Bottom Sketch) and Control Rod Follower (Top Sketches).

Configurations IV.A - IV.D were identical with Configurations III.A - III.D except that the larger section of the control rod subassembly housing was shielded; that is, the section containing B_4C between the outer wall and the liner was used.

During the experiment neutron flux measurements were made beyond each configuration with Bonner balls and the Hornyak button. For Configurations III.A and IV.A, Bonner ball and Hornyak button measurements were also made along the axis of the control rod subassembly (control rod mockup absent) from immediately behind the concrete shadow shield to beyond the configuration. In addition, neutron spectral measurements were made beyond Configurations I.A and II.A with the NE-213 spectrometer and the hydrogen counters.

10.3. ANALYSIS OF THE EXPERIMENT; COMPARISON OF CALCULATED AND MEASURED RESULTS

Because the U.S. GCFR program was suspended during the course of this experiment and closeout of the program was to be expedited, the analysis of the experiment was limited to three of the measured configurations: I.A, which was the fuel pin array; II.A, which contained the full exit shield of seven subassemblies; and III.C, in which the control rod mockup was partially inserted in the subassembly housing containing no B_4C . In view of this limitation, interpretation of the experimental data for the other configurations is largely based on comparisons with the data for the analyzed configurations.

10.3.1. Definition of Source

As was the case for the grid-plate shield experiment, the radiation source for the exit shield configurations consisted almost entirely of neutrons born from fissions in the fuel pins simulating the GCFR core and blanket, the fissions being induced by TSR-II neutrons that scattered around the shadow shield shown in Fig. 10.1. Since the technique for defining the source in the analysis was the same as that described in Section 7.4.1 of Chapter 7, it will not be discussed here. Suffice it to say that the final step of the technique was a DOT calculation from which an internal boundary source was output in the plane including the inner surface (reactor side) of the shadow shield, and this source was then used as the external input source for the analysis. As the calculation

progressed through the configuration, other internal boundary sources were output and similarly used as input sources for succeeding steps.

The initial source calculation used the 51-group GCFR cross sections with P_3 expansion and the 150-angle quadrature set.

10.3.2. Transport Calculations for Fuel-Pin Configuration

In the experiment 2- and 5-in. Bonner ball measurements and Hornyak button measurements were made 30.5 cm beyond the fuel-pin configuration (1.A) and within its horizontal midplane. At 155.4 cm beyond the configuration, on-axis measurements were made with the 2-, 5-, and 10-in. Bonner balls and with the NE-213 spectrometer and hydrogen counters.

The first calculations for Configuration 1.A were DOT calculations using the cylindrical annular-ring model shown in Fig. 10.10, the standard GCFR 51-group P_3 cross sections, and a 96-angle (symmetric S_{12}) quadrature which it was thought would be a good compromise between the 70-angle and 150-angle quadratures used in the grid-plate shield analysis (see Chapter 7). Following the calculation for the model in Fig. 10.10, the boundary flux at the end of the configuration was extracted from the output flux tape (via a special computer program) for use as a boundary source in a followup DOT calculation for a single void zone that encompassed the detector locations close to the configuration (within 40 cm). For larger distances, the FALSTF code was used to transport the flux to the detector locations.

The material compositions assumed for the calculation are given in Table 10.2. In the initial calculations, Concrete Mix #1 was used for the concrete regions shown in the model. This mix is a long-standing analysis of concrete blocks routinely used in TSF experiments and represents the composition assumed for the analysis of the grid-plate shield experiment. However, when the resulting DOT neutron fluxes 30.5 cm beyond the configuration were folded with the response functions for the 2- and 5-in. Bonner balls, the calculated responses were significantly higher than the measured responses,

Fig. 10.10. Two-Dimensional Cylindrical Model Used for DOT Calculations of Configuration 1.A. Model had 64 axial intervals and 60 radial intervals.

Table 10.2. Material Atom Densities Used in Analysis of Exit Shield Experiment

	Material Composition [atoms/(barn·cm ²)]						
	Carbon Steel	B ₄ C ^a	Fuel Pins	End Plug	Spacer ^b Plates	Concrete ^c Mix #1	Concrete ^d Mix #2
H	---	---	---	---	---	8.88(-3)	1.067(-2)
B-10	---	1.095(-2)	---	---	---	---	---
B-11	---	4.437(-2)	---	---	---	---	---
C	9.18(-4) ^d	1.496(-2)	---	---	5.53(-4)	7.97(-3)	1.093(-2)
O	---	---	3.394(-2)	---	---	4.202(-2)	4.654(-2)
Na	---	---	---	---	---	2.73(-5)	4.618(-6)
Mg	---	---	---	---	---	1.44(-3)	4.823(-3)
Al	---	---	1.447(-2)	6.025(-2)	2.40(-2)	4.14(-4)	3.658(-4)
Si	---	---	---	---	---	3.83(-3)	2.682(-3)
S	---	---	---	---	---	1.015(-4)	3.376(-5)
K	---	---	---	---	---	2.34(-3)	1.504(-4)
Ca	---	---	---	---	---	1.0(-2)	8.498(-3)
Fe	8.372(-2)	---	---	---	5.04(-2)	2.64(-4)	1.684(-4)
NI	---	---	---	---	---	---	---
U-235	---	---	3.42(-4)	---	---	---	---
U-238	---	---	1.663(-2)	---	---	---	---

^aMaterial density of 1.3 g/cc.

^bFor 91 pins per assembly.

^cAnalysis of standard TSF concrete; assumed for grid-plate shield experiment calculations.

^dAnalysis of concrete used in the experimental configuration.

especially for detector locations opposite the concrete (see Fig. 10.11). Since a study performed during the analysis of the grid-plate experiment had indicated the importance of using the correct composition of concrete in the calculations, the calculations were repeated with Concrete Mix #2, which is a composition based on a chemical analysis of the concrete actually used in the experiment. The second mix yielded results for the 2-in. Bonner ball that were in excellent agreement with the measurements for locations opposite the concrete but were somewhat lower than the measurements for locations opposite the fuel pins. The results for the 5-in. Bonner ball also were slightly better with the second mix; therefore for all subsequent calculations Concrete Mix #2 was assumed.

While it is not obvious from Fig. 10.11, subsequent calculations for the exit shield configuration (see Section 10.3.3) showed that the 96-angle quadrature was inadequate; therefore all the foregoing calculations were repeated with the 150-angle quadrature. The results, presented in Fig. 10.12, show closer agreement of the measured and calculated 2-in. Bonner ball responses for locations opposite the fuel pins, the maximum discrepancy of 8.4% being the same order of magnitude as the combined experimental and analytical uncertainties. In the case of the 5-in. Bonner ball, the calculated responses continue to overpredict the measured responses, as much as 15 to 18% opposite the fuel pins. However, this overprediction may be due to an inadequacy of the biased quadrature in the single-void zone, since a FALSTF calculation of the on-axis flux 30.5 cm beyond the configuration resulted in a 5% underprediction of the Bonner ball response. (Note: The periodic fluctuations in the calculations shown in Fig. 10.12 result from the annular-ring model used to represent the pin array.)

A comparison of the calculated and measured Hornyak button responses 30.5 cm beyond Configuration 1.A is shown in Fig. 10.13. Here the calculated responses are normalized to the measured responses on the centerline. As was repeatedly recognized during the grid-plate shield experiment, an unresolved and unexplained normalization problem exists with the Hornyak button response used in the analysis and the magnitude of the calculated response is obviously incorrect. Figure 10.13 indicates, however, that the relative spatial distribution of the response is correctly predicted.

The neutron spectrum calculated for an on-axis location 155.4 cm beyond the configuration is compared with the measured spectrum in Fig. 10.14. Unfortunately, within the time limits of the analysis it was not possible to investigate the reasons for the calculation underpredicting the spectrum.

Fig. 10.11. Comparison of Measured and Calculated Bonner Ball Responses 30.5 cm Beyond Fuel-Pin Configuration 1.A (96-Angle Quadrature Used in Analysis).

Fig. 10.12. Comparison of Measured and Calculated Bonner Ball Responses 30.5 cm Beyond Fuel-Pin Configuration 1.A (150-Angle Quadrature Used In Analysis).

Fig. 10.13. Comparison of Measured and Calculated Hornyak Button Responses 30.5 cm Beyond Fuel Pin Configuration I.A. Calculation normalized to measurement on axis (see text).

Fig. 10.14. Comparison of Measured and Calculated Neutron Energy Spectra 155.4 cm Beyond Fuel-Pin Configuration I.A.

Finally, the Bonner ball measurements made at the 155.4-cm location are given in Table 10.3. These responses were not calculated in the analysis.

Table 10.3. Measured Bonner Ball Responses on Axis 155.4 cm Beyond Fuel Pin Configuration 1.A.

Bonner Ball	Response (Counts/s·W)
2-in.	2.84×10^{-1}
5-in.	2.14×10^1
10-in.	1.19×10^1

10.3.3. Transport Calculations for Exit Shield Configuration

As described in Section 10.2 and shown in Figs. 10.2 - 10.4, the exit shield configuration (identified as Configuration 11.A) consisted of seven hexagonally shaped subassemblies, each containing two B_4C -filled cones interfaced at their bases and surrounded by helium coolant passageways. The subassemblies were positioned in the experimental assembly behind the fuel subassemblies and aligned with them.

In the experiment, radial traverse measurements were made with 2- and 5-in. Bonner balls at a distance of 39.8 cm beyond the configuration and within its horizontal midplane. In addition, centerline measurements were made with the Hornyak button, the 10-in. Bonner ball, the NE-213 spectrometer, and the hydrogen counters at a distance of 39.8 cm beyond the configuration. Finally, radial traverse measurements were made with the Hornyak button close behind the configuration (at 1.9 cm).

Because of the importance of this configuration, the analysis used both the discrete ordinates code DOT and the Monte Carlo code MORSE. The cylindrical R-Z model for the DOT calculations is shown in Fig. 10.15. As noted, the interfaced cone-shaped shields in the central subassembly were modeled as stepped cylinders, while the shields in the six outer subassemblies were modeled as cylindrical annuli with smaller dimensions. These calculations continued to use the GCFR 51-group P_3 cross sections, and, following an

Fig. 10.15. Two-Dimensional Cylindrical Model Used for DOT Calculations of Configuration II.A. Model had 60 axial intervals, 60 radial intervals in fuel region, and 62 radial intervals in shield region.

unsuccessful attempt to use a symmetric S_{12} (96-angle) angular quadrature, they also continued to use the biased 150-angle quadrature that had been applied in the analyses of the earlier experiments.

The external boundary source for the Configuration II.A calculation was an internal boundary source that had been taken from the previous Configuration I.A calculation at a distance of 90 cm into the fuel region. As was the case for Configuration I.A, the fluxes emerging from the end of the configuration were transported to the detector locations by performing a followup DOT calculation for a single void, with an output flux tape from the preceding DOT run used as a boundary source.

Comparisons of calculated and measured detector responses are shown in Fig. 10.16 for the Bonner ball radial traverses at 39.8 cm and in Fig. 10.17 for the Hornyak button radial traverse at 1.9 cm. It is clear from these comparisons that the streaming characteristics of the exit shield are more properly calculated with the 150-angle quadrature set than with the 96-angle set. Though slightly misplaced owing to modeling

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Fig. 10.16. Comparison of Measured and DOT-Calculated Bonner Ball Responses 39.8 cm Beyond Exit Shield Configuration 11.A.

Fig. 10.17. Comparison of Measured and DOT-Calculated Hornyak Button Responses 1.9 cm Behind Exit Shield Configuration II.A. [Calculations normalized to measurement on x-axis (see text in Section 10.3.2).]

limitations, the peaks at the streaming gaps are well predicted with the 150-angle quadrature for the 2-in. Bonner ball and the Hornyak button. They are somewhat overpredicted for the 5-in. Bonner ball, but that is because the assumption of a point detector breaks down for the larger diameter ball. Smoothing the calculation with the proper weighting function would broaden and lower the peaks.

The prominent peaks in the calculated responses along the axis of the exit shield are also apparently due to modeling limitations. While smearing the cone-shaped shields of the six outer subassemblies into a single annular zone may have properly predicted the total neutron transmission through the shield, it appeared to have reduced the calculated effectiveness of the shields. This explanation is supported by the DOT neutron isoflux contours shown in Fig. 10.18, which clearly show greater streaming around the outer shields than around the center shield. (The modeling problem probably would not appear in a full reactor analysis since in that case a single-cell calculation with reflective boundary conditions would probably be performed.)

Fig. 10.18. Isoflux Contours for Total Neutron Flux in Exit Shield Region of Configuration II.A.

The Bonner ball and Hornyak button responses were also calculated by the Monte Carlo method. Except for the fact that the thin iron fins that supported the central cone section of the shields were neglected, the model for these calculations was nearly an exact mockup of a 1/12th segment of the exit shield region, with reflective boundary conditions used to simulate the entire geometry. The source was the boundary flux at the fuel-shield interface as generated in the DOT calculation and prepared for the MORSE code by the DOMINO code.

Comparisons of the MORSE-calculated responses with the measured Bonner ball and Hornyak button responses are shown in Figs. 10.19 and 10.20 respectively. The agreement is obviously poor, and because of the curtailment of the program the reasons could not be thoroughly investigated.

Finally, the DOT-calculated neutron energy spectrum for the on-axis location 39.8 cm beyond the configuration is compared with the measured spectrum in Fig. 10.21, and the measured Bonner ball responses at the same location, as well as the measured Hornyak button response, are given in Table 10.4. The overprediction of the calculated spectrum at some energies, as much as a factor of 2, may be attributable to the increased calculated transmission through the outer shield subassemblies that was discussed above.

Fig. 10.19. Comparison of Measured and MORSE-Calculated Bonner Ball Responses 39.8 cm Beyond Exit Shield Configuration II.A.

Fig. 10.20. Comparison of Measured and MORSE-Calculated Hornyak Button Responses 1.9 cm Behind Exit Shield Configuration II.A.

Fig. 10.21. Comparison of Measured and DOT-Calculated Neutron Energy Spectra on Axis 39.8 cm Beyond Exit Shield Configuration II.A.

Table 10.4. Measured Hornyak Button and Bonner Ball Responses on Axis 39.8 cm Beyond Exit Shield Configuration 11.A

Detector	Response
Hornyak button	3.36×10^{-5} ergs/g-s-kW
2-in. Bonner ball	2.06×10^{-3} counts/s-W
5-in. Bonner ball	6.04×10^{-1} counts/s-W
10-in. Bonner ball	3.92×10^{-1} counts/s-W

Since measurements were not made at the same location beyond Configurations 1.A and 11.A, it was not possible to make direct comparisons of the effectiveness of the exit shield. However, a comparison of the 5-in. Bonner ball centerline measurements made 30.5 cm beyond Configuration 1.A (281 cm from TSR-11 center) with those made 39.8 cm beyond Configuration 11.A (375.1 cm from TSR-11 center) indicated that the total neutron flux measurements for Configuration 11.A were lower by a factor of approximately 3×10^2 . Thus, even with the Configuration 11.A fluxes increased to correct for the increased geometric attenuation, the exit shield is shown to be highly effective in attenuating the flux emerging from the core-blanket region to an acceptable level in the upper plenum.

10.3.4. Transport Calculations (or Measurements) for Exit Shield and Control Rod Configurations

As has been described in Section 10.2, for the remaining eight configurations in the experiment (Configurations III.A - III.D and IV.A - IV.D), the central fuel subassembly and the central exit shield subassembly were removed and replaced with a control rod subassembly that extended through the entire length of the configuration. The eight configurations differed in the position of the control rod mockup within the subassembly housing and in the shielding provided in the housing. Because of time limitations, only one of the eight configurations (III.C) was included in the analysis, and it was selected as being representative of the series. In the absence of calculations for the other configurations, the unanalyzed experimental data are included in this section, together with some interpretation based on intercomparisons of the various configurations.

Details of the control rod subassembly housing (no control rod mockup included) are presented in Fig. 10.5, and its position in the experimental assembly is shown in

Fig. 10.6. End views of the assembly are seen in Figs. 10.7 and 10.8. The specific configuration depicted in Fig. 10.6 is Configuration III.A, in which the subassembly housing was unshielded. Configuration IV.A was identical to Configuration III.A except that the subassembly housing was shielded; that is, B_4C filled the region between the lining and the wall in the longer section of the housing (the section in Fig. 10.6 extending from approximately midway from the fuel section on through the exit shield section).

In the experiment three positions of the reactor control rod were mocked up within both the unshielded and the shielded control rod subassemblies: the fully inserted position; the partially withdrawn position (50% inserted); and the fully withdrawn position. The sections of the control rod and control rod follower sketched in Fig. 10.9 were combined as required to simulate these positions.

Figures 10.22, 10.23 and 10.24 show the resulting configurations for the unshielded control rod subassembly. In the configuration representing the fully inserted position (III.B), one 60-cm section of the control rod mockup was inserted to within 7 cm of the concrete shadow shield and was followed by both sections of the control rod follower. For Configuration III.C, both sections of the control rod mockup were used, followed by one section of the control rod follower. And for Configuration III.D, the control rod follower mockup was removed and the two control rod mockup sections were withdrawn to a position flush with the outer face of the configuration. The corresponding configurations for the shielded control rod subassembly (Configurations IV.A - IV.D) were identical except that the outer section of the subassembly housing contained B_4C as noted above.

The first measurements with the control rod subassembly consisted of axial traverses within both the unshielded and the shielded subassembly housing with both the 2-in. Bonner ball detector and the Hornyak button detector. The results are shown in Fig. 10.25. Over the first 100 cm the slopes of the four curves are essentially the same, but as they approach the exit shield region the slopes of the two Bonner ball curves drop faster than those of the Hornyak button curves. In particular, the slope for the Bonner ball curve for Configuration IV.A drops rapidly as the detector enters the section of the subassembly housing containing B_4C between the inner liner and the outer wall.

Fig. 10.22. Experimental Configuration Consisting of Exit Shield and Unshielded Control Rod Subassembly with Control Rod Fully Inserted (Configuration III.B). Note: Configuration IV.B was identical except that right-hand section of Control Rod Subassembly housing was shielded.

Fig. 10.23. Experimental Configuration Consisting of Exit Shield and Unshielded Control Rod Subassembly with Control Rod Partially Withdrawn (Configuration III.C). Note: Configuration IV.C was identical except that right-hand section of Control Rod Subassembly housing was shielded.

Fig. 10.24. Experimental Configuration Consisting of Exit Shield and Unshielded Control Rod Subassembly with Control Rod Fully Withdrawn (Configuration III.D). Note: Configuration IV.D was identical; except that right-hand section of Control Rod Subassembly housing was shielded.

Fig. 10.25. 2-in. Bonner Ball and Hornyak Button Axial Traverses Within Control Rod Subassembly: Unshielded (Configuration III.A) and Shielded (Configuration IV.A).

In studying Configurations III and IV, the primary concerns were whether the presence of the control rod subassembly compromised the exit shield and whether the shielding in the subassembly housing itself was effective. The Hornyak button responses from radial traverses immediately behind Configurations III.A - III.D, which did not have the shielding, are shown in Fig. 10.26. The highest dose rate occurs, of course, for the case in which the control rod mockup is absent (III.A). The next highest is for the fully inserted position (III.B). The dose rate then drops a factor of 3 as the control rod mockup moves to the partially withdrawn position (III.C) and an additional factor of 5 as it moves to the fully withdrawn position (III.D). Simultaneously, the dose rates behind the adjacent exit shields drop a factor of 2 with each step in the control rod withdrawal. On an absolute basis, the dose rate behind the configuration with the partially withdrawn control rod (III.C), which presumably would be the most likely position of the control rod in the GCFR, is about the same as that beyond the solid exit shield configuration (II.A) (see Fig. 10.17).

Similar Hornyak button responses for Configurations IV.A - IV.D presented in Fig. 10.27 show that on the average the dose rates are reduced by a factor of about 2 when the B_4C shield is included in the control rod subassembly housing. The largest reduction is for the partially withdrawn position of the control rod mockup (a factor of 3) and the smallest is for the fully withdrawn position (a factor of 1.3). Obviously, including the B_4C would be worth the extra expense and effort.

Finally, measured 2- and 5-in. Bonner ball responses along radial traverses 39.2 cm beyond the configurations are shown in Figs. 10.28 and 10.29 for the unshielded and shielded subassemblies, respectively, and the same trends seen in the Hornyak button traverses are apparent. Centerline measurements from these traverses, plus centerline measurements at the same location with the 10-in. Bonner ball and the Hornyak button, are presented in Tables 10.5 and 10.6.

The analysis performed for Configuration III.C was performed with the DOT code using the cylindrical R-Z model shown in Fig. 10.30. As noted above, in this configuration a 120-cm length of the cylindrical control rod mockup was inserted to within 7 cm of the concrete shadow shield and was followed by a control rod follower mockup. These sections were represented in the model as steel cylinders, and, as before, other regions of the configuration were represented as cylindrical annuli.

Fig. 10.26. Measured Hornyak Button Responses 1.9 cm Behind Configurations with Exit Shield and Unshielded Control Rod Subassembly (Configurations III.A-III.D).

Fig. 10.27. Measured Hornyak Button Responses 1.9 cm Behind Configurations with Exit Shield and Shielded Control Rod Subassemblies (Configurations IV.A-IV.D).

(a) 2-In. Bonner Ball

(b) 5-In. Bonner Ball

FIG. 10.28. Measured Bonner Ball Responses 39.8 cm Beyond Configurations with Exit Shield and Unshielded Control Rod Subassembly (Configurations III.A-III.D).

(a) 2-In. Bonner Ball

(b) 3-In. Bonner Ball

Fig. 10.29. Measured Bonner Ball Responses 39.8 cm Beyond Configurations with Exit Shield and Shielded Control Rod Subassembly (Configurations IV.A-IV.D).

Table 10.5. Measured Hornyak Button and Bonner Ball Responses on Axis 39.8 cm Beyond Exit Shield/Control Rod Configurations III.A - III.D

Configuration	Hornyak Button Dose Rate (ergs/g-s-kW)	Bonner Ball Response (counts/-s-W)		
		2-In. BB	5-In. BB	10-In. BB
III.A	2.17(-3)*	3.20(-1)	2.08(1)	1.18(1)
III.B	3.33(-4)	2.09(-2)	6.01(0)	3.22(0)
III.C	1.16(-4)	4.02(-3)	1.95(0)	1.13(0)
III.D	2.29(-5)	1.93(-3)	4.00(-1)	4.60(-1)

*Read: 2.17×10^{-3} .

Table 10.6. Measured Hornyak Button and Bonner Ball Responses on Axis 39.8 cm Beyond Exit Shield/Control Rod Configurations IV.A - IV.D

Configuration	Hornyak Button Dose Rate (ergs/g-s-kW)	Bonner Ball Response (counts/-s-W)		
		2-In. BB	5-In. BB	10-In. BB
IV.A	1.64(-3)*	2.98(-1)	1.69(1)	9.42(0)
IV.B	1.81(-4)	1.15(-2)	4.01(0)	2.05(0)
IV.C	3.83(-5)	1.88(-3)	7.04(-1)	4.73(-1)
IV.D	1.67(-5)	1.38(-3)	2.28(-1)	2.01(-1)

*Read: 1.64×10^{-3} .

Fig. 10.30. Two-Dimensional Model Used for DOT Calculations of Configuration III.C. Model had 98 axial intervals, 60 radial intervals in the fuel region, and 63 radial intervals in the shield region.

Since both the fuel region and the exit shield region were modified in Configuration III.C, it was necessary to include the full configuration in the calculation and to use the original boundary source. Again the calculation was performed with the 51-group GCFR P_3 cross sections and the 150-angle biased quadrature set.

The resulting calculated Hornyak button and Bonner ball responses beyond Configuration III.C are compared with the measured responses in Figs. 10.31 and 10.32 respectively. Overall the agreement is excellent, with the same exceptions as has been discussed earlier. That is, deficiencies in the calculational model introduced peaks in the calculated curve. Also, it should be remembered that the calculated Hornyak button data are normalized to the measured data at the centerline.

Fig. 10.31. Comparison of Measured and Calculated Hornyak Button Responses 1.9 cm Behind Exit Shield and Unshielded Control Rod Subassembly with Control Rod Partially Withdrawn (Configuration III.C).

Fig. 10.32. Comparison of Measured and Calculated Bonner Ball Responses 39.8 cm Beyond Exit Shield and Unshielded Control Rod Subassembly with Control Rod Partially Withdrawn (Configuration III.C).

10.4. CONCLUSIONS

This experiment together with its abbreviated analysis has shown that the GCFR exit shield would be highly effective in attenuating the total flux directed upward from the core and blanket before it reaches the plenum region, and, moreover, that the shield would not be compromised by control rod penetrations. In fact, a comparison of the measured neutron fluxes emerging from an exit shield configuration with and without a mockup of a control rod subassembly in the center of the shield shows that when B_4C shielding was included in the control rod subassembly housing, the flux directly behind the control rod subassembly was lower than that directly behind an exit shield subassembly in the same location for most positions of a control rod mockup within the subassembly. Furthermore, the neutron fluxes emerging from the adjacent exit shields were also reduced.

Analysis of a few of the configurations with the discrete ordinates code DOT resulted in good agreement between the measured and calculated results, indicating that the analytical method is adequate for use as a design method in the exit shield region of the GCFR.

10.5. REFERENCES

1. F. J. Muckenthaler, J. L. Hull, and J. J. Manning, "GCFR Plenum Shield Design - Exit Shield Experiment," ORNL/TM-7715 (May, 1981).
2. D. T. Ingersoll and S. N. Cramer, "Final Analysis of the GCFR Exit Shield Integral Experiment," ORNL/TM-7839 (July, 1981).