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Progress Report

Period Ending October 31, 1975

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NEUTRON PHYSICS DIVISION PROGRESS REPORT

For Period Ending October 31, 1975

F. C. Maienschein, Director

JANUARY 1976

OAK RIDGE NATIONAL LABORATORY
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INTRODUCTION AND SUMMARY

The Neutron Physics Division continues here its custom of compiling a progress report from abstracts and summaries of papers and reports published or presented at scientific meetings during the reporting period, which this time extends from August 31, 1974 to October 31, 1975. Since this policy results in a report that does not fully reflect the Division's program for the indicated period, we offer the following discussion of highlights of the Division's current activities.

Outline

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The general areas of research covered by the Division have been described in considerable detail in the introductions to the two preceding progress reports (ORNL-4902 and ORNL-4997). They consist of measurements, calculations, and evaluations of data that describe the interactions of nuclear particles with matter; the processing of such data for use in calculations of the transport of the particles through matter; the development of transport calculational methods; and the application of the data and methods to various systems having internal or external radiation sources. In addition, the Division has established information centers through which the specialized technology it develops and similar technology developed elsewhere are made available to industry, contractors, universities, and other interested groups. In the case of the nuclear data studies, much of the work is coordinated with the Cross-Section Evaluation Working Group (CSEWG), and the data are deposited in the National Neutron Cross-Section Center (NNCSC) for release through the ENDF/B files.

LMFBR PROGRAM

The major program in the Division continues to be one sponsored by the Energy Research and Development Administration's Division of Reactor Research and Development (DRRD) and designed to provide support to the nuclear design of liquid-metal fast breeder reactors (LMFBR). We are currently phasing out work for the Fast Flux Test Facility (FFTF) now under construction at the

Hanford Site near Richland, Washington, and are midstream in work for the proposed 300-MW(e) Clinch River Breeder Reactor Plant (CRBR) to be built in Oak Ridge. As would be expected, much of the technology developed for the FFTF has been applicable, with adaptations, to the CRBR. This is a progression that we expect to carry over to the first large-size breeder reactor [~1500 MW(e)]. The status of the various studies in this program is given below.

Cross-Section Studies for the Fissile and Fertile Materials

Experiments have continued at our ORELA facility (Oak Ridge Electron Linear Accelerator) to obtain accurate values of neutron cross sections and related quantities for the fertile and fissile materials. In one of the experiments now in progress, measurements have been made of ν of ^{239}Pu and ^{235}U in the region below a few electron volts. The purpose of these low-energy measurements is to tie down the normalization of upcoming measurements that will focus on the energy dependence of $\bar{\nu}(E)$ over the important energy region up to several MeV. It has already been demonstrated that reliable measurements with a low background are possible through the entire energy range. The results indicate that ν for ^{239}Pu gradually drops by 1% from 0.01 to 0.3 eV, which dependence is contrary to an assumed constancy of ν in this energy region in the current ENDF/B ^{239}Pu evaluations. This variation may explain some of the difficulties that have been encountered in interpreting ^{239}Pu data obtained in experiments using thermal spectra.

Another major experimental effort has concentrated on the determination of the capture and fission cross sections of some of the actinides with A > 239 (specifically, ^{240}Pu, ^{241}Pu, ^{241}Am, and ^{242}Pu), and this work has progressed to the point that all the data obtained thus far are in process of publication. The absorption cross-section measurements on ^{241}Am (to above 200 keV) are the first available. New fission cross-section measurements will be necessary for ^{241}Am and ^{240}Pu, and the work on ^{242}Pu is just now well under way. All these cross sections are being determined within the accuracy limits currently estimated to be required for fast reactor design.

In a third series of experiments the transmission of neutrons through ^{238}U samples having a large range of thicknesses has been measured for a flight path of 40 m, and measurements are in progress for a flight path of 150 m. The purpose of these measurements is to obtain high-accuracy data to allow a more definitive determination of the resonance parameters of ^{238}U. The initial analysis revealed a large discrepancy between the measured cross sections and those reproduced from the ENDF/B files for neutron energies between resonances. This discrepancy was removed when a multilevel formalism that took into account distant resonances was used instead of the presently employed single-level formalism. Work is under way to use the transmission data to produce a complete set of ^{238}U resonance parameters in time for the ENDF/B-V release.

In another analysis, it was demonstrated that intermediate structure exists in the capture cross section of ^{238}U in the unresolved resonance region. This structure is not taken into account in the current ENDF formats and procedures for handling evaluated data. However, a preliminary study indicates that the estimated effect of this structure on reactor calculations for the CRBR is much smaller than the effect of the uncertainty in the overall normalization of the capture cross sections.

In our continued effort on the practically important nuclide ^{238}U, we have recently initiated, with the aid of an IAEA fellow, a study of ^{238}U fission in the subthreshold region and up through the threshold region to a few MeV. Preliminary measurements using a 20-m flight path exhibit many previously unreported subthreshold fission resonances. Refined measurements at a 40-m station are currently in progress and should reveal in some detail the structure of the cross section near threshold.

Cross-Section Measurements for Low- and Medium-A Nuclides

Cross-section studies were also continued at ORELA for other nuclides included in LMFBR systems, that is, those in the structural, coolant, control, and shield materials. The specially designed scattering chamber used earlier to acquire inelastic-scattering data for the elements sodium, silicon, and iron was modified and used to obtain elastic-scattering data for these elements for neutron energies up to about 2 MeV. Reduction of the experimental data to inelastic- and elastic-scattering cross sections is essentially complete. The elastic-scattering measurements revealed details that had never been observed before but which fortunately could be checked. The method is to integrate the angular distributions of the scattered neutrons to obtain the integral elastic-scattering cross sections and compare them with total cross sections measured independently with high accuracy by transmission. Since below the threshold for inelastic scattering the capture cross sections are small, the total cross sections and the total elastic cross sections should be equivalent. It is anticipated that these new elastic-scattering data will have a significant impact on the ENDF/B-V releases for these elements.

Other measurements geared for the ENDF/B-V releases are measurements of secondary gamma-ray-production cross sections for nickel, chromium, and stainless steel. In addition, measurements are being made to improve the total and capture cross sections for iron, nickel, and chromium.

Cross-Section Evaluations and Uncertainty Studies

The Division continues to participate in CSEWG cross-section evaluations for the ENDF/B files, both for the fissile and fertile materials and for the low-A and medium-A nuclides. During the reporting period work was begun on an updated evaluation for carbon, whose elastic scattering cross sections below 2 MeV are often used as a standard. We also have the responsibility for updating the evaluations for iron and sodium.

With the advent of cross-section sensitivity analysis methods that can identify the cross sections important to a given calculation and specify the uncertainties in the calculated responses attributable to uncertainties in the cross sections (see discussion below), it has become mandatory that the ENDF/B releases include covariance files. The ENDF/B-IV libraries for nitrogen, oxygen, and sodium all contain covariance files in a format developed within the Division. In order to maintain state-of-the-art cross sections for our ORNL calculations, we have developed in-house covariance files for other major nuclides in LMFBR systems, in particular for the plutonium and uranium isotopes and also for iron.

The covariance files for the heavy elements are based on an external statistical analysis of the manner in which the various important data sets scatter about the evaluated cross sections. This analysis method fails to account for uncertainties common to the various data sets. One such source of common uncertainty, the standard cross sections used in many measurements, has been analyzed within the Division so that the necessary uncertainty file components can be produced when uncertainty files become available for the standards themselves.

Cross-Section Processing

The development of cross-section processing techniques has continued with a debugging effort on the MINX neutron code and the combination of MINX with gamma-ray modules in the AMPX system. The advantage of using MINX is that with it we can produce an ultra-fine group master cross-section library based on detailed calculations of the physics involved without considering problem-dependent factors. This master set can then be used to produce sets for different types of systems, which in turn can be collapsed to produce problem-dependent subsets with a great savings in computer time. The widespread use of a master cross-section library should result in its consistent improvement through feedback from the users.

With support from various LMFBR programs and also from ERDA's Division of Controlled Thermonuclear Research (see discussion of RSIC-CTR program below), we are using the MINX-AMPX system to develop a master coupled cross-section library consisting of 171 neutron groups and 36 gamma-ray groups for about 60 materials. The major elements for LMFBR problems have already been included and a DRRD cross-section library containing 126 neutron groups and 36 gamma-ray groups is already available and has been collapsed to a 39-16 subset for Division calculations. It is anticipated that subsets produced by this method will be compared with other sets in common use.

Subsets of the master cross-section library in a different format are used by the ORIGEN isotope generation and depletion code to calculate such quantities as fission-product concentrations, gamma-ray and beta-ray spectra, and decay heat; actinide transmutation; etc.

Core Analysis Methods

The diffusion theory code VENTURE for performing reactor core analyses in up to three dimensions has been sent to the Argonne Code Center and a report describing the code has been published. A mode of data handling was added to VENTURE to use efficiently a slow, extended core for solving large problems. Application of the code by General Electric and Westinghouse Advanced Reactors Division (WARD) from remote terminals was supported by upgrading the auxiliary codes for input-data processing, cross-section processing, and calculation of reaction rates. Extensive testing of VENTURE was done on the IBM-360/195 computer to assess calculation cost and iteration and data transfer strategies. A depletion module is being programmed to serve with VENTURE and similar neutronics codes. A new diffusion theory code is being programmed with the objectives of reducing the finite-difference approximation error and allowing complicated geometries to be described with a flexible triangular meshpoint arrangement on planes normal to the axis of a reactor core; the formulation admits application of the usual and linear finite-difference, higher-order Taylor series, and linear finite-element formulations.

Shield Analysis Methods

Work on Version IV of the two-dimensional discrete ordinates code DOT has progressed to the extent that a limited version is now operational and is being used in-house for CRBR analyses. We expect to release DOT-IV in the fall of 1976. The advantages of DOT-IV over DOT-III are that it has the capability for solving larger problem meshes with less fast memory, and it allows the radial spatial mesh and directional quadrature to vary with axial position so that the computation work can be concentrated in the most important spatial regions.

During the analysis of the FFTF we discovered that the VIP code, originally developed to link the SWANLAKE sensitivity code with DOT, could be used with DOT to identify the paths that particles follow through a system and thereby to show where additional shielding should be placed if needed. The VIP code folds together the forward and adjoint fluxes and normally integrates over zones. By modifying VIP to produce the output by spatial interval, we were able to obtain plots that showed the "channels" of particle transport through a specified *r-z* region. And since in the FFTF case the source was strongly directed toward the location of interest (upward toward the head compartment) the plot could be interpreted as showing the channels through which particles contributing to the head compartment dose reached their destination. Realizing the value of such plots, we have subsequently developed a simpler folding code called FANG which has several options, including the option of considering only those particles that actually reach the location of interest. The result is a plot that truly shows the channels through which particles travel to contribute to the response at that location.

Sensitivity Analysis Methods

The development of the cross-section sensitivity code system for reactor core studies reported on last year has been expanded to include full reactor systems. Called FORSS, it is a modular code system that uses MINX, AMPX, PUFF, SPHINX and ANISN to process cross sections from the ENDF files; DOT, ANISN, and VENTURE to perform transport or diffusion-theory calculations; and SWANLAKE to carry out sensitivity analyses. With FORSS, uncertainties in reactor performance parameters can be calculated and the components contributing to the uncertainties can be identified. As a result, data measurements that would most economically reduce those uncertainties can be selected. Basic to this capability are the cross-section covariance files (see above) that consider information from differential experiments. It now appears that techniques can be developed to incorporate information from selected integral experiments into the data base by adjusting the group cross sections and their associated covariance matrices. This potential depends on our ability to compute the sensitivity coefficients linking the observed integral parameter and the group cross section of interest. If successful, it will allow a posteriori estimates to be made of the uncertainties associated with a design problem, as well as estimates of the cross-section requirements.

Further work on FORSS will be aimed at preparing a version of the system suitable for release. In the meantime the various components to be incorporated in FORSS are being perfected and applied to a number of problems.

Core Neutronics Calculations for FFTF and CRBR

Work is essentially complete on the analytical program supporting the development of subcritical reactivity monitoring techniques for the FFTF. We completed the analysis of the engineering mockup critical ZPR-9 BOL-REF-5S experiment to test the adequacy of the proposed FFTF low level flux monitors (LLFMs). We also completed the analysis of the reverse-approach-to-critical experiment performed on the ZPR-9 to assess the usefulness of the LLFMs in determining the initial core startup reactivity. Calculations are also being performed to support the design of the CRBR LLFM, including calculations to serve as a basis for designing a series of LLFM experiments for the ZPPR-5 at Argonne National Laboratory (ANL).

Throughout FY 1975 calculations were performed to determine the sensitivity of CRBR reactor performance parameters (i.e., breeding ratio, sodium void coefficients, and control rod worths) to changes in cross sections, modeling techniques, design parameters, etc. In addition, similar calculations were performed for a critical experiment set up on the ZPR-6-7 at ANL. These calculations included a FORSS-type analysis, performed with assistance from the Georgia Institute of Technology, that showed that the cross-section sensitivity profiles for the CRBR performance parameters were similar to those for corresponding parameters calculated for the critical experiment. This led to the conclusion that clean critical experiments will be very useful in assigning uncertainties to the performance parameters calculated for specific reactors.

Beginning with FY 1976, the charter for this work was modified to cover direct analytical neutronics support for the design of the CRBR, which, among other things, is to include an analysis of bowing (fuel bending) reactivity coefficients, criticality calculations for the ex-vessel stored-fuel tank, and a scoping effort on the initial core loading problems associated with the startup of the CRBR, the last of these to be carried out in conjunction with the ORNL Instrumentation and Controls Division.

Reactor Safety Studies

In July 1975, the Division initiated a reactor safety program in which very sophisticated and detailed techniques are used to calculate k_{eff} values and material worths that are compared with values obtained by less elaborate techniques. Such comparisons will indicate the adequacy of various methods for neutronics analysis of distorted systems. Thus far the entire effort has been devoted to the determination of the reactivity associated with the collapse of bubbles produced by stainless steel vaporizing in a medium of molten fuel; both Monte Carlo and discrete ordinates techniques are being applied. We are just beginning to develop a three-dimensional quasi-static kinetics code that will be based on Monte Carlo techniques and will handle nonorthogonal coordinate geometries.

In a separate effort sponsored by the Nuclear Regulatory Commission (NRC), we are developing calculational models for analyzing a series of reactor safety experiments performed in the ORNL Reactor Division. In the experiments a capacitor is discharged through a UO_2 sample, causing the sample to fragment violently. In the first experiments the UO_2 samples will be surrounded by an inert atmosphere of argon, and in later experiments they will be surrounded by sodium. This will be a long-term effort.

Shielding Analyses for FFTF and CRBR

In the Division's LMFBR shielding program, supporting analyses on the FFTF are being phased out, and radiation shielding analyses for the CRBR are well under way. Except for two studies now in progress, all the FFTF work dating back to 1969 has been reviewed in a recently published report, including a sensitivity analysis of the reactor head which represented the first application of the VIP-SWANLAKE code linkup to a two-dimensional system. Of the current studies, one consists of a further examination of the sources contributing to the secondary gamma-ray dose above a concrete shield in the cavity surrounding the reactor vessel. The other consists of a simple-model three-dimensional calculation of radiation that streams through the reactor head via gaps around three in-vessel handling machines and then penetrates up through the thick steel operating floor.

Among the first studies for the CRBR was an analysis of the preliminary design of the system using the end-of-equilibrium cycle core, the geometry extending from the core through the reactor cavity, closure head and head compartment. Particular attention was given to the complex streaming gaps formed by the largest of three rotating plugs in the closure head. In spite of the fact that fuel will not normally be stored within the reactor vessel, as it will be in the FFTF vessel, these calculations indicated that severe streaming problems exist in the vessel support area, and several design modifications for this region were assessed. Current work is concerned with a preliminary design for the pipe chaseways leading to the intermediate heat exchanger. The complexity of the chaseways has necessitated the development of new 90° coupling calculational techniques that are yet to be tested in an experiment to be performed at the Tower Shielding Facility.

Shielding Experiments and Analyses

Several TSF experiments supporting the CRBR analyses have already been performed. As has been stated in the previous progress reports, the objective of these experiments is to test the nuclear data and methodology developed for the shielding analyses of specific systems, as well as to provide direct information for the systems. This capability has been greatly improved during the past year by the installation of a new reactor beam shield that has a larger diameter and shorter length than the shield that has been used in the past. With this new shield and an increase in the reactor power from 100 kW to 1 MW, the beam intensity is a factor of 2000 greater than that available for earlier experiments.

One of the major analytical tasks during this reporting period has been the definition of the large-beam source. Whereas the small-beam source was effectively a point source with an energy distribution, the large-beam source is a disk source with energy and angular distributions that vary as a function of the location on the disk. Thus the definition of the source could not be obtained directly from measurements. Instead it was determined through a series of calculations that were normalized to selected measurements. In the calculations an ad hoc analytic method was used to transform the angular flux from ANISN spherical-reactor calculations to a cylindrical geometry for use in DOT and MORSE calculations of the emerging beam.

The new source has made it possible to perform a very challenging experiment measuring neutron transmission through 18 in. of stainless steel followed by 15 ft of sodium and 24 in. of mild steel. A DOT-III (two-dimensional) analysis of the experiment using the 51-group LMFBR neutron cross-section set revealed difficulties in calculating the transmission of high-energy neutrons through the stainless steel zone, as well as cross-section structure and moderation effects in the sodium that had not previously been identified and are of substantial importance to the analyses of LMFBR systems. A year-long FORSS-type sensitivity study performed as a preanalysis of this experiment showed that it would be prototypic of a simplified upper axial shield model for the CRBR.

Two additional experiments were performed during the year with the small-beam source. In one, designed to validate methods used to calculate heating in the reflector-shield region of the CRBR, the heat deposition in mild steel and in laminated mild steel and stainless steel configurations was measured with CaF_2 and LiF thermoluminescent dosimeters and with an ion chamber, and gamma-ray spectra measurements were made with an NaI spectrometer. There were inconsistencies between the three sets of dosimeter measurements as large as 15%. An analysis of the experiment gave results that agreed with the CaF_2-TLD measurements within 10 to 20%. This is outside the limits of the CRBR design uncertainty criterion of less than 10%. A FORSS sensitivity analysis of the experiment is under way to determine which cross-section errors might cause the disagreement, and a new experiment is being carried out during which the accuracy of the measurements will be improved.

The second small-beam experiment was performed at the request of WARD to investigate the effect of including a large amount of mild steel reinforcement in a concrete shield. Gamma-ray transmission measurements showed that there was little difference between a concrete shield containing no rebar and a shield containing rebar both at the back and at the front. This is due to a compensating effect in which the rebar at the front increased the gamma-ray production by a relatively small amount (~10%) and that at the back provided slightly more attenuation.

The analysis of an experiment performed in FY-1974 to check the total cross sections of a number of materials was also completed this year. It resulted in severe discrepancies with the ENDF/B-IV total cross sections for chromium and nickel and indicated a difference of about 5% in the MeV range for sodium. The results for iron were in good agreement with the ENDF/B-IV values.

Waste Management Studies

In addition to the reactor and shielding studies described above, the Division is participating in a program to aid in the design of fuel-reprocessing plants. Our work in this area is supported through ORNL's Chemical Technology Division and consists of calculations with the ORIGEN isotope generation and depletion code to determine the isotopic content of spent CRBR fuel assemblies.

In another study, thus far supported by "seed" money, the ORIGEN code is being used to study reactor "burning" for converting actinide wastes into fission products that have relatively short half-lives and could be processed by conventional methods. The actinides are added to the reactor core where some will undergo fission directly and others will be transmuted into fissionable isotopes. Upon selection of a reactor type, the FORSS system will be used to determine the sensitivity of specific reactor performance parameters to the actinides and to identify the important cross sections for the actinide burning problems.

Nuclear Safety Data Base

In conjunction with the recently initiated reactor safety program (see above), the Division this year established a computerized data base for use as input in various fast reactor safety analysis codes. Called SACRD (for *Safety Analysis Computerized Reactor Data*), the data base will be designed to meet the needs of the various groups in the country who develop and use the safety codes. We are already in the process of collecting and collating data and putting it on ORNL-IBM machines. Guidance for the program will be provided by a coordinating group consisting of representatives from the major organizations involved in safety analysis, including representatives from ERDA and NRC.

RSIC-LMFBR Activities

The Radiation Shielding Information Center continues to be heavily committed to promoting communication among the various groups engaged in LMFBR shielding studies and receives direct support from DRRD for that purpose. As has been pointed out previously, its activities center on facilitating the exchange of nuclear data, transport codes, and shielding literature. It also provides problem assistance upon request.

During FY 1975, RSIC responded to an average of 12.3 requests for assistance per day, which resulted in the following daily activities: shipping 2.6 code packages and 1.3 data packages; mailing 9.1 shielding documents and 1 information packet and or document abstract; responding to 10.5 queries for problem assistance or consultation; and performing 0.4 special literature searches. The literature searches were in addition to the routine SDI service (Selective Dissemination of Information) provided to 432 individuals. (A total of 6006 SDI pieces were sent out in 13 mailings.) In addition, the RSIC newsletter was mailed each month to as many as 1436 persons, and a total of 141 persons, including 29 foreigners, visited RSIC for an orientation or to use the Center's facilities. Maintenance of the RSIC-user directory resulted in 1919 changes during the year.

While DRRD provides the major support to RSIC, it is to be noted that the Center also receives support from other agencies as described below and that the statistics quoted above cover activities in all programs in which RSIC participates.

GCFBR SHIELDING PROGRAM

Another Division program sponsored by DRRD is the shielding analysis program initiated early in 1974 for the 300-MW(e) demonstration model of the Gas Cooled Fast Breeder Reactor proposed by General Atomic Corporation (GA). Several "full-assembly" calculations have now been performed for GA's two-dimensional reference model, with the radial shields treated both as distinct regions of stainless steel and graphite and as homogenized zones. In one calculation, reviewing design options specified by GA, the radius of the massive outer radial shield was reduced by 1 ft, which, if feasible, would greatly reduce the construction costs of the plant.

Several cross-section sensitivity analyses have also been performed for the reference model. A one-dimensional SWANLAKE study was made of the high-energy flux in the core midplane at the iron liner of the PCRV (prestressed concrete reactor vessel); several two-dimensional SWANLAKE-VIP analyses were performed for the lower axial region to examine the high-energy-neutron and thermal-neutron fluxes at various positions in the PCRV liner, the gamma-ray heating in the concrete, and the gamma-ray dose at the tendon position beyond the liner. These studies have enabled us to identify neutron transmission paths to critical design areas and to estimate the effect of varying shield composition and placement. The results will be used by GA to determine a new reference radial shield design.

In addition, a prototypic experiment was performed at the Tower Shielding Facility to investigate neutron streaming through helium coolant channels in a region similar to that from the core to the grid support plate located above the core. The fuel pins were mocked up as 4-ft long aluminum-clad depleted ^{238}U oxide rods set on a triangular pitch in an air-filled aluminum can surrounded by water. Measurements beyond the array indicated a significant streaming effect in the forward angles. In order to calculate this effect accurately for the GCFBR, we must develop techniques for modeling and calculating the region in three dimensions. The TSF experiment can serve as a benchmark for testing the calculational methods as they are developed.

CTR PROGRAM

A third reactor program in which the Division is active is the Controlled Thermonuclear Reactor Program sponsored by ERDA's CTR Division. During the past year, in addition to the basic studies that have been in progress for some time, this work has included participation in a large design for a Tokamak-type Experimental Power Reactor (EPR).

Cross-Section Studies

The technology developed at ORELA under other programs to measure neutron-scattering and gamma-ray-production cross sections for neutrons incident at high energies (~0.5 to 20 MeV) has been used to provide cross sections for several nuclides of interest to the CTR program. Thus far both neutron-scattering and gamma-ray production measurements have been made for niobium, and gamma-ray production measurements have been made for molybdenum and vanadium. In addition, some of the equipment used in these experiments has been adapted to measurements of the spectra of neutrons emitted from the ORMAK Tokamak device.

In recent months preparations have been made for measuring helium-production cross sections for the nuclides in CTR systems. These cross sections are particularly important in that the helium particles are a primary causative of radiation damage to structural components. Thin foil samples will be suspended in a xenon-filled cylindrical chamber that will be exposed side-on in the ORELA neutron beam. At each end of the chamber photomultipliers sensitive to ultraviolet light will detect scintillations in the gas produced by the emission of alpha particles from the samples. The chamber and photomultipliers have already been constructed, and a system for handling the xenon gas has been designed. Since this is a new technique, extensive testing of the detection system will be required with samples whose helium-production cross sections are well known.

In our cross-section theoretical program, we have worked on extending the capability of our nuclear model calculations to include recoil energy spectra and helium-production cross sections for neutrons incident at energies up to 35 MeV. These data are needed for analyses of CTR experiments performed at the Oak Ridge Isochronous Cyclotron using the Be(d,n) reaction.

Neutronics Studies

Studies of the cross-section sensitivity of the tritium breeding ratios calculated for proposed fusion-reactor blanket designs have continued, the latest being for designs proposed by ORNL, by Los Alamos Scientific Laboratory, and by the Princeton Plasma Physics Laboratory. In addition, several other sensitivity studies have been performed, including a study of the sensitivity of the calculated D-T fusion probability to D-T cross sections for various incident deuteron energies and plasma electron temperatures and a study of the sensitivity of the calculated energy deposition and radiation damage in a toroidal field coil to iron and carbon neutron cross sections for a prototypical EPR blanket-shield-coil configuration.

A series of neutronics "scoping" calculations have also been performed as part of a large design study involving a number of disciplines and leading to a reference design for a 100- to 400-MW EPR proposed for construction at ORNL. The neutronics calculations defined the radial dependence of the nuclear heating and radiation attenuation for seven different blanket configurations, as well as the breeding ratios for those modules that contained lithium as the fertile material. Additional calculations, including radiation damage studies, were performed for the design selected as the reference. In another series of calculations for a conceptual reactor design, the effect of various blanket structural materials on several reactor performance parameters was investigated.

To facilitate the CTR neutronics calculations, we have developed an ORNL-EPR cross-section library consisting of 100 neutron groups and 21 gamma-ray groups for the most commonly used cross sections. We have also developed a primary recoil spectra data base in multigroup format which, when coupled with damage and displacement models, yields effective energy and displacement cross sections for radiation damage calculations.

Neutral Atom Transport Calculations

The neutral atom transport studies are designed to provide the energy, spatial, and angular distributions of neutral atoms within a fusion reactor plasma for use as input in plasma physics calculations. To date, these studies have been divided into two parts: studying the neutral particles escaping the plasma and interacting with the first wall; and tracking the neutrals within the plasma.

For the first-wall investigations, a Monte Carlo code was written to calculate the number, energy and direction of the particles refluxed and sputtered from the wall per incident hydrogen atom, the sputtered particles being high-Z atoms knocked from the wall material. In an attempt to reduce the number of particles entering the plasma through the wall, a honeycomb wall design was considered and was found to reduce the fraction of refluxed particles from 0.6 for a smooth wall down to 0.1 to 0.05 for a honeycomb wall. The use of the honeycomb design also significantly reduced the number of sputtered particles. On the basis of these results an experiment to test the honeycomb wall concept will be performed in the ORNL Thermonuclear Division and will be analyzed by us.

Preliminary to tracking the neutrals within the plasma, it was necessary that we develop a code to process multigroup cross sections for interactions of neutral atoms with charged particles (ionization, charge exchange, etc.). This code has been completed, and the cross sections for hydrogen atom interactions have been used in ANISN calculations of the transport of the hydrogen neutrals through a typical ORMAK plasma. The results were obtained as energy, angular, and spatial distributions of the neutrals within the plasma. An estimate of the effect of the particles refluxed from both a smooth wall and a honeycomb wall on the neutral atom distribution within the plasma was obtained by assuming wall albedos of 0, 0.6, and 1 in the ANISN calculations.

RSIC-CTR Activities

In addition to general Radiation Shielding Information Center technical support for the CTR program, efforts have been directed toward the development of a versatile multigroup master cross-section library consisting of 171 neutron groups and 36 gamma-ray groups. As pointed out above in the discussion of the LMFBR program, the development of this library is being sponsored by several programs.

The master library will retain detailed information on various reaction cross sections of importance in CTR neutronics calculations, covering about 60 materials. The processing of the cross sections into multigroup form is being carried out with ORNL's version of the MINX neutron code and the gamma-ray production and interaction modules of AMPX. To date (December 1975), 35 materials have been processed by MINX.

A data testing and verification program is planned which will involve calculations at ORNL and various other laboratories and universities involved in CTR neutronics studies. Distribution of the library and associated manipulation programs will follow. The data package will be kept open and updated by RSIC to reflect any significant feedback from users.

HIGH-ENERGY ACCELERATOR SHIELDING PROGRAM

The Division's high-energy accelerator shielding program sponsored by ERDA's Division of Physical Research continues to support the design of shields for high-energy particle accelerators. In addition, the program is assuming an increasingly important role in aiding the design of instruments used by experimentalists at the accelerators.

Transport Calculations

This year the high-energy particle transport calculations have included a series in which the high-energy code HETC was used to calculate the energy distribution of neutrons produced when 200-MeV protons are stopped in a thick water shield and the ANISN discrete ordinates code was used to calculate the transport of the neutrons throughout the shield. These, plus similar calculations for 200-MeV protons incident on aluminum targets embedded in a concrete shield, were performed to aid the design of shielding for a cyclotron under construction at Indiana University. Other neutron transport calculations, performed with ANISN and the DOT discrete ordinates code, yielded data that supported the design of shielding for a biomedical facility at the Fermi National Accelerator Laboratory and for experimental areas at the accelerator facility of the Massachusetts Institute of Technology.

The heavy ion code HIC, developed under the Division's now-defunct NASA and cancer radiotherapy programs, is also being applied in this program to predict the interactions of heavy ions with nuclei. The code is currently being used to obtain shielding information to support proposals for Phase II of ORNL's Heavy Ion Accelerator.

Other work is directed toward improving our calculational capability. In particular, the high-energy code HECC for calculating cross sections for particles having energies on the order of hundreds of GeV is undergoing further development.

Ionization Calorimeter Design

Calculations are performed with the HETC code to aid in the design of ionization calorimeters that measure the energy of various high-energy particles. The calorimeters are of two basic types: those which view the light produced in a liquid or plastic scintillator and those which collect a charge that drifts across a medium consisting of alternate layers of some dielectric material and a heavy absorbing material. Until recently iron has been the preferred absorber, but calculations now in progress are investigating the use of ^{238}U as the absorber with liquid argon as the dielectric. The advantage of the ^{238}U is that fast fissions will occur in the ^{238}U and return to the system energy that would ordinarily be lost as binding energy. As a result, the device will give a larger signal and its resolution will be improved. It is anticipated that pulses from incident hadrons will then be approximately equal to those from incident electrons.

A large number of experimental groups are now basing their calorimeter designs on our calculations—to the extent that we have become recognized as the international center for this type of work. As a result of our continued interaction with the experimentalists, much more efficient calorimeter designs are being realized than were possible with the empirical methods used in the past, and the number of instrument tests required on accelerators has been greatly reduced.

DNA PROGRAM

The Division's Defense Nuclear Agency program, which is primarily concerned with providing the nuclear data and methods needed for radiation transport analyses of systems exposed to radiations from nuclear detonations, has continued during the last year, though at a somewhat reduced level. In general, it has consisted of completing tasks initiated in earlier years.

Cross-Section Studies

The group performing cross-section measurements and evaluations under the DNA program has now assumed full responsibility for the evaluations of the nuclides included in the DNA Working Cross Section Library (see discussion of RSIC-DNA activities below). In this capacity they are working on incorporating recent measurements in the DNA evaluations. The group also works closely with CSWEG and is reporting on some of the evaluations included in the ENDF/B-IV releases. In particular, papers describing evaluations for lead and calcium are scheduled for publication in *Nuclear Data Tables* in November 1975, and February 1976, respectively. Evaluations for fluorine, iron, and silicon are also continuing, with a description of the silicon evaluation containing post-ENDF/B-IV total cross-section data based on new measurements soon to be released. In addition an evaluation of the gamma-ray-production cross sections for gold has been performed.

The experimental studies in the DNA cross-section program continue to concentrate on ORELA measurements of gamma-ray-production cross sections. Measurements were performed during this period for gold, and the results were used in the evaluation for that element. Earlier measurements for zinc, silver, fluorine and tantalum are described in reports that were issued during the year.

Experimentalists in this group also participated recently in a unique nondestructive check of a component that will be part of a nuclear weapons test. In the check, the fabricated thickness and material composition of the component were determined from time-of-flight spectral measurements of neutrons transmitted through the component.

Cross-Section Processing

The IBM-360 Phase-II version of AMPX, the system of interfaced modular codes that produced coupled neutron and gamma-ray multigroup cross sections for transport codes, has now been released through RSIC with the number of basic modules included in the system increased from six to sixteen. With aid from ORNL, Sandia Laboratories and Control Data Corporation have successfully adapted

AMPX (Phase II) to CDC-type machines, and their versions of the system will soon be made available through RSIC. This essentially concludes our development of AMPX, future work being limited to routine maintenance of the system and the addition of special modules as needed.

The few-group cross-section library processed by AMPX and tailored for DNA contractors who must use small computers has been tested and released through RSIC (see below), together with three neutron sources (14 MeV, thermonuclear, and fission spectra), several response functions, and two weighting functions.

The few-group library, which contains 37 neutron groups and 21 gamma-ray groups, was tested extensively by comparing neutron and secondary gamma-ray responses obtained with the set and with a many-group set (129 neutron groups and 43 gamma-ray groups) when they were used for calculations for an infinite-air medium and a concrete medium. It was found that for high-energy sources the few-group set with a $1/E$ weighting function yielded responses that were within 10% of those obtained with the many-group set. For lower energy sources in infinite air, the agreement was within 20%, which was outside the 10% uncertainty criterion set by DNA. When flux-weighted nitrogen and oxygen cross sections based on the 129-43 calculations for infinite air were used in the few-group set, however, the results for the low-energy sources also agreed within 10% with those obtained with the many-group set (for infinite air). It was therefore decided to include with the few-group set both the $1/E$ weighting function and the flux-weighted cross sections for nitrogen and oxygen.

Transport Methods Development

The DNA radiation transport methods development program this past year has concentrated on an improved version of the Monte Carlo multigroup code MORSE. Identified as MORSE-SGC, the improved code has a "supergrouping" feature that allows cross-section information to be rolled in and out of a problem as needed. As a result, any number of energy groups can be used even on small computers. Also, redundancy in the cross-section input has been eliminated, with the same input now being available to all modules. In addition, the cross-section input structure has been converted from a fixed form to a free form.

MORSE-SGC has now been flexibly dimensioned in the same mode as other standard codes used by the Division, and an improved combinatorial geometry module has been attached to it. Also, several common routines that heretofore had to be prepared by the user for each problem are now available as standard generalized user routines.

The major thrust of the effort now is to ensure the adaptability of the code to small computers such as those used by contractors of the military. It has already been made operable on the CDC-6400 at Kaman Nuclear.

Integral Experiments and Analyses

Two DNA-supported integral experiments to check differential cross-section data were performed at ORELA this year, one for a SiO_2 sample (simulating concrete) and another for aluminum. In both cases the spectra of secondary neutrons and gamma rays produced by neutron interactions in the sample were measured as a function of the incident neutron energy. Corresponding data obtained from earlier experiments using samples of water, nitrogen, and oxygen were also published this year.

The analyses of the experiments have concentrated on the earlier measurements. The analysis for nitrogen showed that the ENDF/B-IV evaluation is excellent for both gamma-ray production and neutron scattering except for a discrepancy in the neutron scattering that occurs at around 5 MeV. [This discovery has led to a re-examination of the nitrogen (n,α) cross section at this energy.] This analysis appears to resolve the long-term controversy on gamma-ray production in nitrogen. The major unresolved discrepancy now is a disagreement that persists between measurements made at ORNL and at Intelcom Rad Tech (IRT) in the forward direction. For our analysis we assumed that the ORNL data were correct.

The analysis for oxygen shows little difference in the results obtained with ENDF/B-III and ENDF/B-IV, both disagreeing significantly with the forward-angle measurements. An analysis of an earlier experiment for iron with ENDF/B-IV data gave good agreement, correcting a discrepancy that had occurred in the high-energy secondary neutron spectra when ENDF/B-III cross sections were employed.

Transport and Sensitivity Calculations

The benchmark-quality calculations performed last year with the DOT code for 14-MeV and fission sources in air-over-ground and air-over-seawater geometries received further analysis this year and the entire study was published. The analysis of the air-over-ground results yielded plots that show the source height and ground-range combinations for which the neutron and secondary gamma-ray doses at the air-ground interface are greater than those at equivalent ranges in infinite air.

The VIP-SWANLAKE linkup was used to perform a large-scale sensitivity analysis on the doses obtained from the DOT air-over-ground calculations. The results showed that the total calculated dose is very sensitive to nitrogen neutron cross sections for detectors both near and far above the ground and that nitrogen and silicon thermal-neutron captures produce over half of the total gamma-ray dose. It also showed that hydrogen dominates neutron transport in the ground, so much so that even relatively small changes in the amount of water in the ground produce changes in the dose which are outside the range predicted by linear perturbation theory.

RSIC-DNA Activities

During the past year approximately 25% of the basic RSIC services have been devoted to Department of Defense agencies and their contractors who perform weapons radiation transport calculations. The area in which assistance is most sought by these agencies is the area of computer technology, with frequent requests received for transport codes.

We have also been filling requests from the DNA Working Cross Section Library, which was established within the Center a few years ago and now includes 27 nuclides. All the evaluations in the Library are essentially equivalent to the ENDF/B-IV releases; however, those for nitrogen and oxygen have changes not yet incorporated in ENDF/B. Also, the evaluation for gold includes gamma-ray production data not available in ENDF/B, and the evaluation for ^{182}W has updated gamma-ray production data.

Other DNA activities during the past year include the release of the DNA few-group cross-section library (37 neutron groups, 21 gamma-ray groups) and Version II of the AMPX processing code, both of which are described in earlier paragraphs.

OTHER PROGRAMS

In addition to the major programs described above, the Division is engaged in several separate studies that are largely spin-offs of the major programs and closely associated with them, as will be apparent from the following discussion.

Fission-Product Energy Release Measurements

In a program sponsored by the Nuclear Regulatory Commission to determine the amount of heat associated with the decay of fission products, we have now made preliminary measurements of the gamma-ray decay heat following the thermal fission of ^{235}U to a moderate precision (to within 7%) and expect to perform final measurements in the next few months. Measurements of the beta-ray decay heat will follow, and the results from all the measurements will be available for safety and accident analyses for light-water reactors.

The data obtained are actually the spectra of gamma rays, from which the decay heat is deduced. Although not of specific interest to this program at this time, the spectra provide an experimental base against which calculations performed with the ORIGEN code (see above) can be checked. The comparisons will be used to aid the CSWEG fission-product subcommittee in identifying those isotopes that are important in decay heat measurements. At present the comparisons show good agreement at long times (≥300 sec) but significant disagreements at short times.

Neutron Cross-Section Standards

Work begun last year to help refine some of the standards required for fast-neutron measurements is still maintained at a low level, the first standard studied being $\bar{\nu}$ of ^{252}Cf (the average number of neutrons produced per spontaneous fission of the isotope). Since $\bar{\nu}$ of ^{252}Cf is the standard upon which nearly all measurements of $\bar{\nu}$ for fast reactors are based, it needs to be known to within 1.4%—or to within a narrower tolerance if possible.

The measurements for ^{252}Cf consist in placing a sample of the isotope at the center of a gadolinium-loaded scintillator tank and observing the number of scintillation pulses produced by the capture of neutrons emitted following a fission in the sample. The interpretation of the results depends, of course, on our knowledge of the efficiency of the detection system, which is being determined by calculations and checked by concurrent measurements in which the scintillations produced by "tagged" neutrons of known number and energies are observed. To date, only an initial set of measurements have been made.

This program is supported by ERDA's Division of Physical Research.

Transport Calculations for Picatinny Arsenal

Under the sponsorship of Picatinny Arsenal, we have initiated calculations to generate a computerized data base that will require only minimal input to determine the radiation doses produced near the air-ground interface by any weapon detonated at any burst height up to 100 m. The calculations are being performed with the two-dimensional code DOT in the adjoint mode. Thus they begin with a specified dose response function at the detector location (1 m above the ground) and determine the energy and location of the neutrons that could produce that dose. When the results are folded with the spectrum from a source of interest, doses may be obtained as a function of ground range and burst height.

This program, which is closely associated with the DNA program described above, could be expanded to include a variety of scenarios and response functions, thereby precluding the necessity for large-scale transport calculations for each situation.

RSIC-EPRI Activities

Because the unique services rendered by the Radiation Shielding Information Center (RSIC) are invaluable to the nuclear power industry, the Electric Power Research Institute (EPRI) has recently become one of the sponsors of RSIC. Under a Memorandum of Understanding between ERDA and EPRI (contract between UCND and EPRI), the nuclear power industry will receive information and advice and assistance in implementing state-of-the-art computer technology in the field of radiation analysis and protection. To ensure that the most up-to-date transport methods are available, RSIC will acquire computing technology radiation transport computer codes and data libraries of use to the nuclear power industry and will test, assemble, evaluate, and document them in order to make them generally available throughout the industry. Toward this end, RSIC is currently conducting a survey of the radiation transport information and computing technology needs of the industry, and a report giving the results of the survey will be prepared soon for use both by EPRI and by ERDA. On the basis of the survey, recommendations will be made to EPRI on areas requiring further research and development. In a longer range project, RSIC will work on defining the problem of fission-product and "crud" radiation source transport as it impacts the shielding design of a nuclear reactor plant.

BCTIC—A Biomedical Computing Technology Information Center

As a result of cooperative efforts that began in 1971 between ORNL and members of the Society of Nuclear Medicine (SNM), ERDA's Division of Biomedical and Environmental Research (DBER) has established within our Division the Biomedical Computing Technology Information Center (BCTIC). Five annual symposia on the sharing of computer programs and technology in nuclear medicine (sponsored by DBER and SNM with the Society for Computer Medicine and the HEW FDA Bureau of Radiological Health) have emphasized the need for the improvement and standardization of technology to facilitate diagnostic and therapeutic procedures in nuclear medicine clinics and laboratories. The BCTIC is chartered to collect, organize, evaluate, and disseminate information on computing technology among those engaged in biomedicine in general and those working in nuclear medicine in particular and to serve as a central location for the exchange of such information.

Although BCTIC is still in the collection phase, over 250 queries from the field have already been logged, and the second bimonthly newsletter is soon to be issued. Five large computer programs with geometric models of phantom men for calculating internal doses due to external gamma-ray sources have been collected. The contributors are the ORNL Health Physics Division (the ALGAM and INDOS programs), NASA (the CAMERA program with the CAM model), the U.S. Air Force (MEVDP), and the Central Research Institute of Physics of Hungary (DISDOS). The BCTIC is now the official disseminator of the ALGAM data base MIRD (Medical Internal Response Dose), which is a large body of evaluated data that describe gamma-ray interactions in tissue. BCTIC also has access to the major medical bibliographic data bases maintained by HEW through the Bureau of Radiological Health. In addition, specialized computer codes, including some that have models simulating body organs, are being received.

Another major facet of the BCTIC program is the preparation of a "workbook" that summarizes the computing resources and procedures utilized by clinics and laboratories throughout the world. A first draft of the workbook will be completed in January.

ORELA Improvement Program

The Neutron Physics Division and the Physics Division, who share responsibility for the operation of the ORELA facility, have submitted a proposal to ERDA's DPR Accelerator Improvement Program which, if approved, will allow the full power of the accelerator to be available with the narrow resolution required for some experiments.

The improvements will be accomplished by installing between the electron gun and the accelerator proper prebunching devices that would tend to accelerate the slower electrons and decelerate the faster electrons so that more electrons would be concentrated in a shorter burst width. In the development of the prebunching devices, efforts have been concentrated on generating appropriate high-voltage pulse wave forms in model prebuncher elements and testing them with an electron beam. The feasibility of using a series of the devices on the accelerator has been investigated by performing one-dimensional calculations that predict the behavior of the electron beam. The calculations indicate that despite large space charge effects, a shorter burst width without appreciable loss of charge can be achieved.

It is to be noted that the ORELA is the only experimental facility for which the Neutron Physics Division retains operation responsibilities. During the past year responsibilities for the operation of the Tower Shielding Reactor II and the Health Physics Research Reactor were assumed by the Operations Division. This change does not affect our experimental program at the TSF, however.

Additional information on the programmatic activities of the Neutron Physics Division is provided in the following papers, and the organizational responsibility for the various areas of research is shown in a chart presented on p. 119.

Lorraine S. Abbott

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1. NOTE ON THE CAPTURE WIDTH OF THE 6.67-eV LEVEL IN ^{238}U[1,2]

G. de Saussure R. B. Perez

At the 1975 seminar on ^{238}U resonance capture the suggestion was made that the capture width of the 6.67-eV level in ^{238}U might be appreciably smaller than the value 25.6 meV included in Versions III and IV of ENDF/B. This suggestion was based, in part, on a comparison between a Monte Carlo calculation and a measurement of the capture probability in a thick sample of ^{238}U.[3] For the level at 6.67 eV the calculation indicated a broader peak than the measurement (Fig. 1.1). In this note we present a few comments on this argument.

1. The measurement and calculation referred to (illustrated in Fig. 1.1) were not done for the purpose of determining the capture width of the 6.67-eV levels but were done to normalize a time-of-flight measurement by the saturated resonance technique. For such normalization, the only

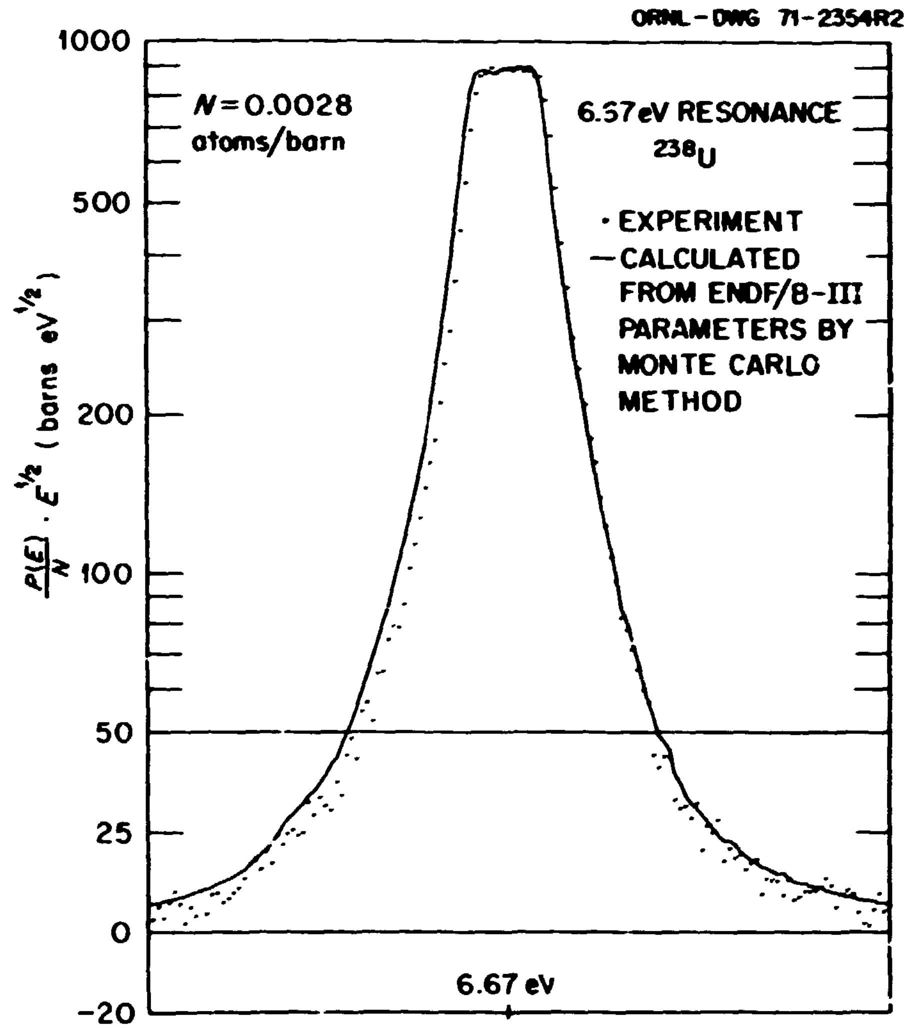

Fig. 1.1. Probability of capture in a 25-mil sample of ^{238}U at the 6.67-eV resonance. The ordinate is the probability of capture multiplied by the square root of the energy and divided by the sample thickness in atoms per barn. The calculation was done with the Monte Carlo code MULTSCA using the resonance parameters of ENDF/B-III. Note that the ordinate is linear from 20 to +50 b.eV$^{1/2}$ and logarithmic above 50 b.eV$^{1/2}$.

quantity of importance is the value of the capture probability on the "flat top" of the peak, and this quantity is insensitive to the value of the capture width.

The relative energy scales of the calculation and of the measurement shown in Fig. 1.1 were not well aligned, and the calculation was made assuming a disk of infinite radius, whereas the measurement was done with a disk 7.65 cm in diameter. Hence it seems unwise to conclude from Fig. 1.1 that the capture width of the 6.67-eV level is smaller than that used in the calculation (25.6 meV).

2. An attempt was made to see whether our previous measurements of the capture probability in the saturated 6.67-eV level could yield information of the capture width of that level, despite that the intent of the measurements was not to determine that width.

The Monte Carlo program was refined to include the effect of the finite size of the capture sample, and calculations were done with capture widths of 21, 23, and 25.6 meV for sample thicknesses of 0.002849 and 0.000789 atom/b.

The calculations were compared with a set of nine measurements recently done with those two sample sizes.[4] The experimental background could not be measured directly with sufficient precision, so the calculations were fitted to the measurement and a "free" background. The results of this comparison are illustrated in Table 1.1. In Fig. 1.2 we compare one run with calculations done with capture widths of 21 and 25.6 meV.

As shown in Table 1.1, the data from the nine saturation measurements examined seem to favor a value of 23 meV for the capture width of the 6.67-eV level in ^{238}U. But this conclusion is barely statistically significant. Furthermore, there is at least one possible experimental effect that could reduce the apparent capture in the wings of the resonances.

Neutrons with energies near 6.67 eV have a very short mean free path in the sample and hence are captured on the surface of the sample. However, neutrons in the wings of the resonance are captured nearly uniformly through the sample. Hence the capture gamma rays produced by the latter neutrons are perhaps more attenuated by the sample. We don't know how to compute reliably the magnitude of this effect in the complicated geometry of the experiment, and a realistic estimate should come from measurements with "split samples."

Table 1.1. Comparison of calculation and measurements of the capture rate in the 6.67-eV level on resonance in ^{238}U

Run number	Thickness (atoms/b)	"χ^2" of fit[a] for Γ_γ of			Background obtained by fit[a] for Γ_γ of			Estimated background
		21 meV	23 meV	25.6 meV	21 meV	23 meV	25.6 meV	
4670	0.002849	1.167	1.147	1.501	2873	2839	2795	2943 ± 200
4710	0.002849	1.466	1.248	1.782	3630	3561	3474	3745 ± 200
4730	0.002849	[illegible]	1.098	1.657	3809	3732	3633	3972 ± 200
4790	0.002849	1.318	1.168	1.760	3504	3428	3354	3639 ± 200
4900	0.000789	1.407	1.192	1.450	662	643	620	650 ± 30
5010	0.000789	1.065	0.825	0.933	937	910	876	918 ± 50
5050	0.000789	1.177	1.094	1.164	1103	1076	1039	1076 ± 50
5060	0.000789	1.146	0.982	1.234	1470	1434	139[illegible]	1444 ± 50
5070	0.000789	1.072	0.800	1.105	1205	1174	1134	1178 ± 80

[a]The experimental data were fitted to a function $A \cdot S(E,\Gamma_\gamma) + B$, where $S(E,\Gamma_\gamma)$ is the capture probability computed for a given value of Γ_γ by a Monte Carlo program. The numbers in columns 3 to 5 are proportional to the weighted sums of the residual for the fit. The numbers in columns 6 to 8 are the values of B obtained by the fits. The numbers in column 9 are estimates of B obtained independently of the fit (with a "background scaler").

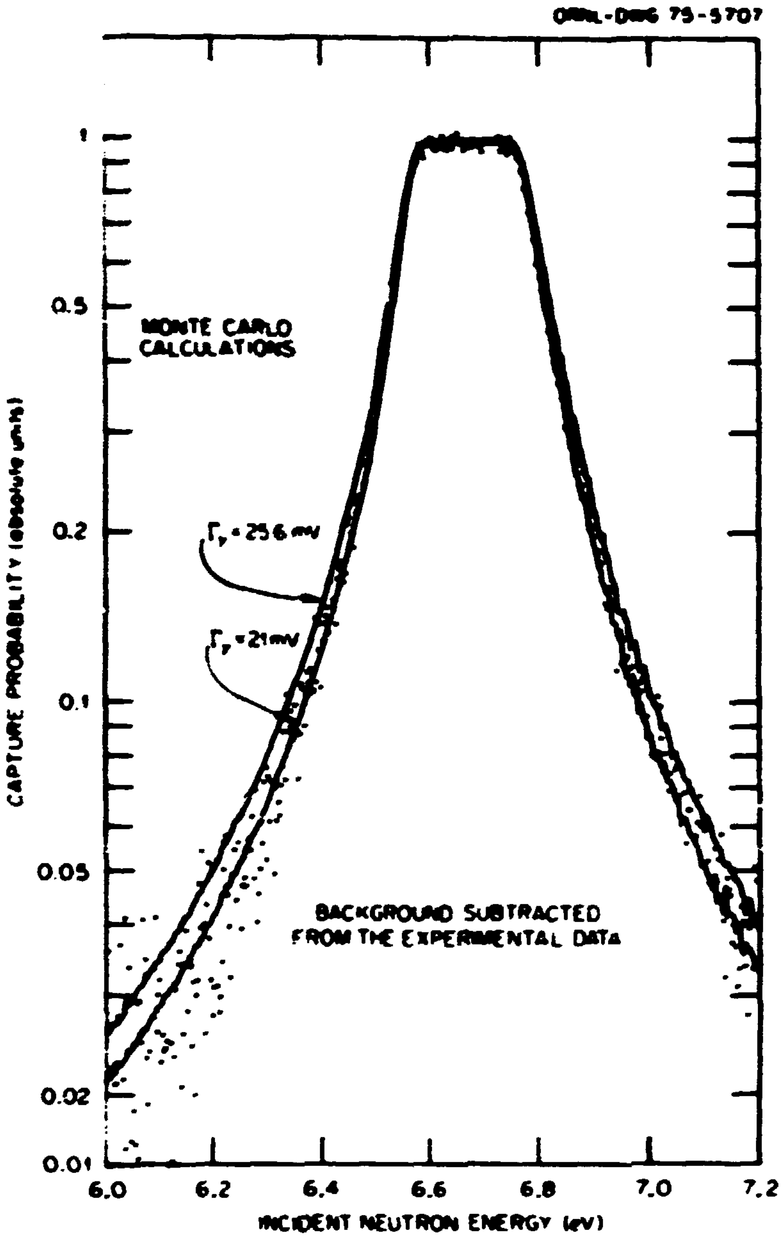

Fig. 1.2. Capture probability in the 6.67-eV level of ^{238}U for sample of 0.000045 atoms/b.

In conclusion, our previous measurements of the capture rate in thick samples of ^{238}U were not designed to determine the capture width of the 6.67-eV level and do not provide reliable information on this width. An analysis of recent measurements seems to favor a value of 23 meV for this capture width, but additional experiments especially designed for the purpose would be most desirable.

1. Paper included in the proceedings of a meeting held at the National Neutron Cross Section Center, Brookhaven National Laboratory, March 18–20, 1975: *Seminar on ^{238}U Resonance Capture*, BNL-NCS-50451, p. 151.

2. Research sponsored by ERDA's Division of Reactor Research and Development.

3. G. de Saussure et al., "Measurement of the Uranium-238 Capture Cross Section for Incident Neutron Energies up to 100 keV," *Nucl. Sci. Eng.* 51, 385 (1973).

4. By G. de Saussure, J. Halperin, N. W. Hill, R. L. Macklin, and R. B. Perez.

2. THE ^{238}U(n,γ) CROSS SECTION ABOVE THE RESONANCE REGION[1,2]

R. B. Perez R. R. Spencer G. de Saussure

A number of measurements of the ^{238}U(n,γ) cross section above the resonance region have been completed in the past few years. In the keV region these measurements suggest a considerable amount of intermediate structure.

An idea of the present status of affairs is given by inspection of Fig. 2.1 (summarizing data from refs. 3–9). The low-resolution shape of the ^{238}U capture seems to be reaching a "consistent" status with the exception of the measurements of Friesenhahn and Fricke[5,6] and two points (at around 35 and 85 keV) in the results of Moxon.[4] Large differences in normalization are still present. However, the results seem to settle above the ENDF/B-IV evaluation between the high ORELA data[3] and the low Moxon data.[4] This trend will help to obtain better agreement between the calculated and measured value of the dilute capture resonance integral. The presence of intermediate structure has the following implications:

1. It casts doubt on the validity of activation measurements and "spot" normalization procedures of the capture cross section, especially at neutron energies such as 24 and 30 keV, where large fluctuations exist.
2. One may question the validity of the present statistical representation of the unresolved region. For example, corresponding to the 24-keV iron window, there is a large enhancement of the ^{238}U

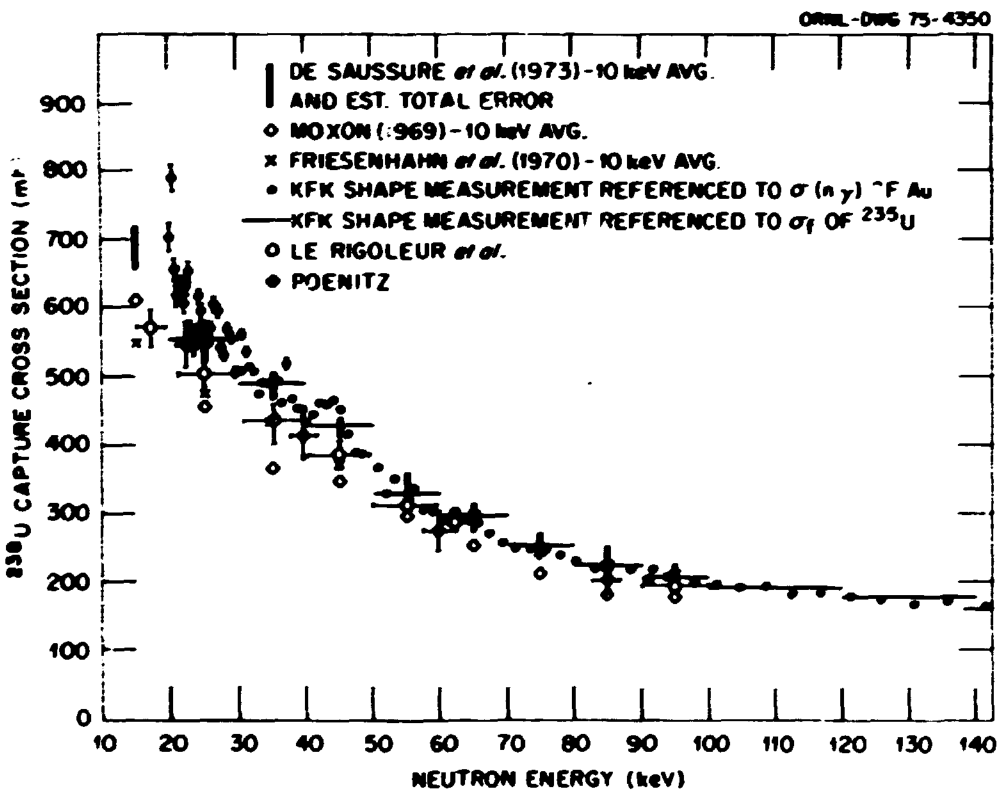

Fig. 2.1. Comparison of various sets of ^{238}U capture cross-section measurements in the 10- to 140-keV region.

capture cross section. Hence the actual detailed behavior of the intermediate structure should be included in the representation of the $^{238}U(n,\gamma)$ cross section. At ORNL we are in the process of testing the effect of these fluctuations in fast-reactor calculations.

In summary, taking into account that fast breeders are mostly lots of iron with a sprinkle of plutonium and a heavy blanket of ^{238}U for breeding, the neutron capture of this important material has to be known better than at least 3%. It is then clear that considerably more work will be needed to achieve this goal.

1. Summary of paper included in the proceedings of a meeting held at the National Neutron Cross Section Center, Brookhaven National Laboratory, March 18–20, 1975: *Seminar on ^{238}U Resonance Capture*, BNL-NCS-50451, p. 103.
2. Research sponsored by ERDA's Division on Reactor Research and Development.

3. G. de Saussure et al., "Measurement of the Uranium-238 Capture Cross Section for Incident Neutron Energies up to 100 keV," *Nucl. Sci. Eng.* 51, 385 (1973).
4. M. C. Moxon, *The Neutron Capture Cross Sections of ^{238}U in the Energy Region 0.5 to 100 keV*, AERE-R-6074 (1969).
5. S. J. Friesenhahn et al., *Neutron Capture Cross Sections on Molybdenum, Tantalum, and ^{238}U*, GA-10194 (1970).
6. M. P. Fricke et al., IAEA-CN-26 II, 265 (1970).
7. R. R. Spencer and F. Kaeppeler, "Measurement of the ^{238}U Capture Cross-Section Shape in the Neutron Energy Region 20 to 550 keV," *Bull. Am. Phys. Soc., Ser. II* 20, 161 (1975).
8. C. LeRigoleur et al., Cad ache, France, private communication.
9. W. P. Poenitz, "Measurements of the Neutron Capture Cross Sections of Gold-197 and Uranium-238 Between 20 and 3500 keV," *Nucl. Sci. Eng.* 57, 300 (1975).

3. PRECISE MEASUREMENTS AND CALCULATIONS OF ^{238}U NEUTRON TRANSMISSIONS[1,2]

D. K. Olsen — E. G. Silver
G. de Saussure — R. B. Perez

We have measured the total neutron cross section of ^{238}U below 1 keV in precise transmission experiments and have compared the results with ENDF/B-IV cross sections. Special emphasis was placed on measuring transmissions through thick samples to obtain accurate total cross sections in the potential-resonance interference regions between resonances. These total cross sections are important in computing shielded capture integrals, and it has been observed[3] that such shielded integrals are overestimated by ENDF/B-IV.

The neutron energies were determined by time of flight along a 40-m flight path at the ORELA. The detector was a 7.62-cm-diam, 1.0-mm-thick lithium-glass disk viewed edge-on by two RCA-7585 phototubes. At 20 m from the neutron source, isotopically enriched ^{238}U disks with inverse thicknesses of $1/n$ = 5405, 1603, 807, 266, 80.7, 19.2, and 5.7 b/atom were alternated in and out of the beam with 10-min cycles. At least four different transmission measurements with various combinations of cadmium, indium, cobalt, aluminum, manganese, and gold beam filters were made for each sample thickness. Blackened resonances from these filters aided in estimating the background levels.

After subtracting the background and dead-time-correcting the sample-in and sample-out time-of-flight spectra, the resulting transmissions for each sample thickness were combined and compared with a Doppler- and resolution-broadened transmission calculated from ENDF/B-IV total cross sections. In particular, Fig. 3.1 shows the experimental transmission of 50- to 300-eV neutrons through 3.62 cm ($1/n$ = 5.7 b/atom)

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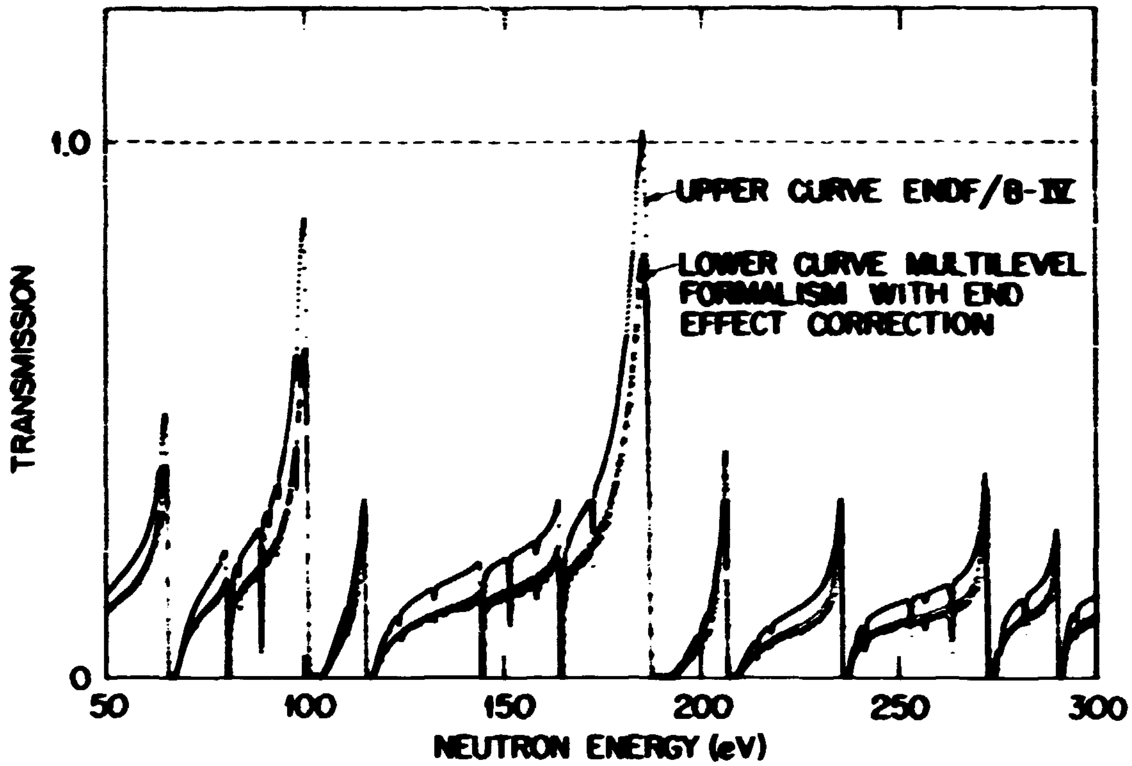

Fig. 3.1. Measured transmission of 50- to 300-eV Neutrons through 3.62 cm of ^{238}U. The upper curve is the transmission calculated from a Doppler-broadened ENDF/B-IV cross section, whereas the lower curve was obtained from a multilevel formalism.

of ^{238}U. The upper curve is the corresponding transmission calculated using the ENDF/B-IV procedure. For energies larger than 40 eV, the calculated transmission between resonances is greater than the measured transmission and, in fact, calculates to be greater than unity (negative cross section) at some interference minima in the cross section. The lower curve shows the transmission as obtained from an exact *R*-matrix calculation of the total cross section using: (1) the random phase approximation for the radiative channels, (2) the resonance parameters and scattering radius from ENDF/B-IV, and (3) the "picket fence" approach (uniform spacings and widths) for levels outside the resolved resonance region. This more exact calculation of the total cross section reproduces the measured transmission much better and does not yield negative cross sections. It is very likely that the failure of ENDF/B-IV to reproduce measured thick-sample transmissions as shown above is the cause of the overestimation of the strongly self-shielded capture resonance integrals. In particular, the capture integral from 100 to 600 eV shielded down to 10.0 b decreases by 4.6% when the multilevel formalism is used in place of ENDF/B-IV procedures.

1. Summary of paper presented at 1975 Annual Meeting of American Nuclear Society, New Orleans, Louisiana, June 8–13, 1975; published in *Trans. Am. Nucl. Soc.* 21, 505 (1975). Also summary of paper included in the proceedings of a meeting held at the National Neutron Cross Section Center, Brookhaven National Laboratory, March 18–20, 1975: *Seminar on ^{238}U Resonance Capture*," BNL-NCS-50451, p. 95.

2. Research sponsored by ERDA's Division of Reactor Research and Development.

3. J. Hardy, private communication; see also *Nucl. Sci. Eng.* 40, 101 (1970).

4. MEASUREMENT OF THE ^{238}U CAPTURE CROSS-SECTION SHAPE IN THE NEUTRON ENERGY REGION 20 TO 550 keV[1,2]

R. R. Spencer F. Kaeppeler[3]

The Karlsruhe 800-liter liquid scintillator detector and 3-MV pulsed Van de Graaff were used to measure the shape vs neutron energy of the ^{238}U capture yield relative to a gold capture sample and relative to ^{235}U fission. The resulting cross-section shape computed from the gold capture cross section is consistent with that computed from a recent evaluation of the ^{235}U fission cross section. Below 100 keV a significant intermediate structure is observed which corresponds to that in recent ORNL data.

1. Abstract of paper presented at Conference on Nuclear Cross Sections and Technology, Washington, D.C., March 3–7, 1975; *Bull. Am. Phys. Soc., Ser. II* 20(2), 161 (1975); full paper published in *Nuclear Cross Sections and Technology*, vol. II, ed. by R. A. Schrack and C. D. Bowman, NBS Special Publication 425, p. 620 (1975).

2. Work performed at Karlsruhe, West Germany; analysis and documentation at ORNL supported by ERDA's Division of Physical Research.

3. Kernforschungszentrum, Karlsruhe, West Germany.

5. INTERMEDIATE STRUCTURE IN THE ^{238}U NEUTRON CAPTURE CROSS SECTION[1,2]

R. B. Perez G. de Saussure

Recent measurements of the ^{238}U neutron capture cross section show large fluctuations in the unresolved resonance region. To test whether the observed local enhancements of the neutron capture are randomly distributed or, on the contrary, represent departures from the statistical nuclear model, the Wald-Wolfowitz runs and correlation tests were applied to the ^{238}U neutron capture data obtained at the ORELA. The Wald-Wolfowitz runs test deals with the statistic R which is the number of unbroken sequences of data points above or below a given reference line. This statistic is to be compared with the expected value of runs $E(R) \pm \sigma(R)$ arising from randomly distributed data. In the correlation test we have computed the first serial correlation coefficient of the data as well as its expected value and variance for a set of random data. In both tests one computes the probability, P, for the given statistic entity to depart from its expected value by more than F standard deviations. Both tests yield $P < 10^{-4}$ between 10 and 100 keV, confirming the presence of intermediate structure. The range of the structure far exceeds the width of the experimental resolution.

1. Abstract of paper presented at Conference on Nuclear Cross Sections and Technology, Washington, D.C., March 3–7, 1975; *Bull. Am. Phys. Soc., Ser. II* 20(2), 161 (1975); full paper published in *Nuclear Cross Sections and Technology*, vol. II, ed. by R. A. Schrack and C. D. Bowman, NBS Special Publication 425, p. 623 (1975).

2. Research sponsored by ERDA's Division of Reactor Research and Development.

6. REPRESENTATION OF NEUTRON CROSS SECTIONS IN THE UNRESOLVED RESONANCE REGION[1,2]

G. de Saussure R. B. Perez

In the unresolved resonance region, the statistical properties of neutron cross sections are usually specified by the average values and distribution functions of the resonance parameters. We discuss deficiencies of this treatment. In particular, it is based on the statistical model of nuclear reactions, a model that is inconsistent with the intermediate structure observed in many heavy nuclei. Present technology allows the measurement of cross sections with an instrumental resolution broadening which is small compared with the Doppler broadening. From the results of such measurements, the Doppler-broadened cross sections may be obtained directly by kernel broadening. We propose to replace the statistical treatment of the unresolved region by a parametric description of the actual cross sections. The use of this cross section, rather than a representation of their statistical properties, is expected to yield more accurate estimates of resonance absorption and self-shielding factors. We review here the experimental conditions required to obtain cross sections with adequate instrumental resolution, and discuss various methods of parametrization of the data.

1. Abstract of paper presented at Conference on Nuclear Cross Sections and Technology, Washington, D.C., March 3–7, 1975; *Bull. Am. Phys. Soc., Ser. II* 20(2), 151 (1975); full paper published in *Nuclear Cross Sections and Technology*, vol. I, ed. by R. A. Schrack and C. D. Bowman, NBS Special Publication 425, p. 371 (1975).
2. Research sponsored by ERDA's Division of Reactor Research and Development.

7. MEASUREMENT OF THE FISSION CROSS SECTION OF URANIUM-235 FOR INCIDENT NEUTRONS WITH ENERGIES BETWEEN 2 AND 100 keV[1,2]

R. B. Perez G. de Saussure
E. G. Silver[3] R. W. Ingle[4]
H. Weaver

The fission cross section of ^{235}U was measured for incident-neutron energies between 2 and 100 keV using the ORELA pulsed source of neutrons and the time-of-flight technique. The fission events were characterized by coincidence between the pulses of a fission chamber, placed at the center of a large scintillation tank, and gamma-ray events registered in the tank. The incident-neutron spectrum vs energy was monitored by a BF_3 ionization chamber and checked with an ^6Li-glass neutron detector. The cross sections were normalized to a value of 31,643 b-eV for the fission integral between 2 and 10 keV.

Comparison of the present ^{235}U fission cross-section results with the values from other sources shows that, in general, there is an agreement within 3 to 5% in the energy region covered by this experiment.

The previously reported intermediate structure in the fission cross section in the keV region is confirmed by the results of this work.

1. Abstract of published paper: *Nucl. Sci. Eng.* 55, 203 (1974).
2. Research sponsored by ERDA's Division of Reactor Research and Development.
3. On loan to the Institute of Energy Analysis, Oak Ridge, Tennessee.
4. Instrumentation and Controls Division.

8. MEASUREMENT OF THE NEUTRON CAPTURE AND FISSION CROSS SECTIONS OF ^{239}Pu AND ^{235}U, 0.02 eV TO 200 keV, THE NEUTRON CAPTURE CROSS SECTIONS OF ^{197}Au, 10 TO 50 keV, AND NEUTRON FISSION CROSS SECTIONS OF ^{233}U, 5 TO 200 keV[1,2]

R. Gwin E. G. Silver[3]
R. W. Ingle[4] H. Weaver

The neutron absorption and fission cross sections for ^{239}Pu and ^{235}U have been measured over the neutron energy range from 0.02 eV to 200 keV. In addition, the neutron capture cross section for ^{197}Au was measured from 10 to 50 keV, and the fission cross section of ^{233}U was measured from 0.1 to 100 keV. Normalization of the ^{239}Pu and ^{235}U data was made over the energy region from 0.02 to 0.4 eV to the ENDF/B-III neutron cross sections for these isotopes, Mat 1159 and 1157 respectively. The capture cross section for ^{197}Au was normalized using the saturated resonance method for the 4.9-eV resonance. For ^{233}U fission the normalization was made using the results of Weston et al. The neutron flux was measured using the ^{10}B(n,α) reaction; the energy variation used for this reaction was that given in ENDF/B-III.

The pulsed neutron beam for these measurements was generated using the ORELA. A large liquid scintillator about 40 m from the neutron source was used to detect the prompt gamma-ray cascades resulting from neutron absorption in the sample. The time interval between the burst of neutrons and the detection of the absorption event was used to establish the neutron energy scale. The sample of the fissile isotopes was contained in multiplate (pulse) ionization chambers, and those neutron absorption events detected in coincidence with a pulse from the ionization chamber were defined as fission events.

In general, for ^{239}Pu and ^{235}U these experiments indicated lower neutron fission cross sections than those contained in ENDF/B-III for energies above 10 keV. The measured values of the ratio α, the ratio neutron captures to neutron fissions, for ^{239}Pu agree within errors with those derived from ENDF/B-III, Mat 1159. For the present measurements the uncertainty on α for ^{239}Pu is about 11% at 10 keV and increases to about 30% at 100 keV.

The experimental results for the neutron capture cross section for ^{197}Au are about 15% lower than the ENDF/B-IV values. The measured values of the ratio of the neutron fission cross section for ^{233}U to that for ^{235}U are generally higher than the ENDF/B-III values by about 5%.

1 Abstract of paper accepted for publication in *Nuclear Science and Engineering*.
2 Research sponsored by ERDA's Division of Reactor Research and Development.
3 On loan to the Institute of Energy Analysis, Oak Ridge, Tennessee.
4 Instrumentation and Controls Division.

9. A DIRECT COMPARISON OF DIFFERENT EXPERIMENTAL TECHNIQUES FOR MEASURING NEUTRON CAPTURE AND FISSION CROSS SECTIONS[1,2]

R. Gwin J. H. Todd[3]
L. W. Weston R. W. Ingle[3]
H. Weaver

Measurements of the neutron capture, σ_c, and fission cross sections, σ_f, of ^{239}Pu have been obtained at ORNL, using different detectors for both fission and capture. These experiments were normalized to the same 2200-m sec cross sections and extended into the keV region. In one experiment, fission was detected by means of the prompt neutrons, and absorption was detected using "total energy detectors" to measure the prompt gamma rays following neutron absorption. The other technique detected fission fragments to measure fission, and a large liquid scintillator detected the prompt gamma rays from absorption. Auxiliary experiments were used to investigate the neutron flux measurements and to test the fast-neutron detector with the pulse shape discrimination method for measuring fission events in ^{239}Pu. For the 28-decade energy intervals between 0.1 to 200 keV, the ratio $\langle\sigma_c\rangle/\langle\sigma_f\rangle$ obtained in the two experiments differs on the average by 6%. The cross sections are in good agreement (~2%) up to 8 keV.

1. Abstract of paper presented at Conference on Nuclear Cross Sections and Technology, Washington, D.C., March 3-7, 1975; *Bull. Am. Phys. Soc., Ser. II* 20(2), 161 (1975); full paper published in *Nuclear Cross Sections and Technology*, vol. II, ed. by R. A. Schrack and C. D. Bowman, NBS Special Publication 425, p. 627 (1975).
2. Research sponsored by ERDA's Division of Reactor Research and Development.
3. Instrumentation and Controls Division.

10. MEASUREMENTS ON THE 22-eV DOUBLET IN ^{232}Th(n,γ)[1,2]

J. Halperin[3] R. B. Perez
G. de Saussure R. L. Macklin[3]

The 22-eV doublet in the ^{232}Th(n,γ) reaction contributes about half the thin-sample resonance capture integral for ^{232}Th. A discrepancy of some 15% in the measured neutron width for the 23.439-eV resonance is apparent among several recent measurements. In view of the special interest in ^{232}Th as the fertile nuclide in the ^{233}U breeder, we have remeasured the capture cross section with a relatively thin (0.0002375 atom/b) metal sample. The measurement was carried out at the 40-m flight path of the ORELA time-of-flight facility. The capture gamma-ray cascade was measured with nonhydrogenous total gamma energy detectors. The incident neutron spectrum was measured with a thin ^{6}Li glass displaced 0.4 m forward in the beam. Since the gamma width dominates in these two resonances, the capture area predominantly provides a measurement of the neutron width. Monte Carlo calculations evaluating the capture area of the resonance at 23.439 eV (provisionally normalized to the better known 21.783-eV resonance) yield a measure of $g\Gamma_n = 3.72 \pm 0.11$ meV. This value is consistent with an unpublished ORNL value of J. A. Harvey's of $g\Gamma_n = 3.74 \pm 0.15$ meV.

1. Abstract of paper presented at Conference on Nuclear Cross Sections and Technology, Washington, D.C., March 3-7, 1975; *Bull. Am. Phys. Soc., Ser. II* 20(2), 165 (1975); full paper to be published in the proceedings of the Conference.
2. Division participation sponsored by ERDA's Division of Reactor Research and Development.
3. Physics Division.

11. MEASUREMENT OF THE NEUTRON CAPTURE CROSS SECTIONS OF THE ACTINIDES[1,2]

L. W. Weston J. H. Todd[3]

The capture cross sections of the isotopes heavier than ^{239}Pu are of great importance for the core physics, fuel management, and waste management of power reactors. Since integral measurements and total cross sections are not sufficient, a program for the measurement of these needed capture cross sections is being carried out at ORNL. Measurements have been completed on ^{240}Pu, ^{241}Pu and ^{243}Am and have been planned for ^{242}Pu and ^{237}Np. The capture gamma-ray detector used is the "total energy detector," which is a modification of the Moxon-Rae detector. Fission, when present, is detected with fast-neutron counters. Results obtained on ^{240}Pu, ^{241}Pu, and ^{243}Am extend continuously from thermal-neutron energies to 300 keV. The cross sections are normalized at thermal-neutron energies, and the neutron flux is measured relative to the ^{10}B(n,α) cross section up to 20 keV and ^{6}Li(n,α) at higher neutron energies. The accuracy of the techniques used varies with the sample but is about 8%. With such cross sections the long-range management of the actinides produced in power reactors can be planned on a more systematic basis. The basic neutron energy resolution of the system is about 0.4 nsec/m.

1. Abstract of paper presented at Conference on Nuclear Cross Sections and Technology, Washington, D.C., March 3–7, 1975; *Bull. Am. Phys. Soc., Ser. II* 20(2), 144 (1975); full paper published in *Nuclear Cross Sections and Technology*, vol. I, ed. by R. A. Schrack and C. D. Bowman, NBS Special Publication 425, p. 229 (1975).
2. Research sponsored by ERDA's Division of Reactor Research and Development.
3. Instrumentation and Controls Division.

12. NEUTRON CAPTURE CROSS SECTIONS OF THE ACTINIDES[1,2]

L. W. Weston

The long-lived actinides have assumed major importance in waste management for power reactors. A great deal of calculational effort has been applied to prediction of the buildup of these nuclides, which will be a hazard for more than 250,000 years. Additional work has been done on the feasibility of transmuting these long-lived nuclides to shorter-lived fission products by recycling them in reactors.

A major deterrent to accurately calculating quantitative actinide buildup and the feasibility of actinide transmutation has been the sparsity of cross-section measurements for the actinide isotopes, which are basically not reactor fuel. This is particularly true for fast reactors that operate with neutron energies above thermal. Fission cross-section measurements are more abundant than are capture cross-section measurements. Normally in reactor calculations the accuracy of the necessary fission cross sections is more important than capture because of multiple neutrons emitted per fission. For the case of actinide buildup and transmutation, however, the fission and capture cross sections assume more of a similar correspondence in importance.

Capture cross sections in the resonance region of neutron energies and up to a few kilovolts neutron energy can often be derived from total cross-section measurements when fission cross sections are not appreciable. These total-cross-section measurements which are simpler than direct capture measurements because of the natural radioactivity of the actinides, require about an order of magnitude less sample in the resolved resonance region of neutron energies. Above a few keV, and at lower neutron energies when fission is appreciable, direct-capture cross-section measurements assume great importance. Integral measurements are valuable but do not contain as complete or as versatile information as differential measurements.

Nuclear model calculations can be used to compute capture cross sections above a few kilovolts. The reliability of such calculations varies according to the particular case. Since direct-capture cross-section measurements cannot be made on all the actinides, particularly those with very high spontaneous fission rates, nuclear model calculations must be relied upon in many cases. For the cases where capture cross sections can be measured, they would enable refinements of the nuclear model calculations for these and other nuclides.

An example of a capture cross-section measurement[8] with an actinide is shown in Figs. 12.1 and 12.2. This measurement with ^{241}Am yields the absorption cross section that is predominately capture since fission is small. This measurement indicates that the cross section above 20 keV is appreciably greater than the nuclear model calculation represented by ENDF B-IV. The difference is important to fast reactors, since it would lead to increased ^{242}Cm production. Production of ^{242}Cm presents a fuel-handling problem because of its high spontaneous fission rate. It would also indicate greater production of the higher actinides in fast reactors. Recent nuclear model calculations[9] are more consistent with the measured data. Even at thermal neutron energies and in the resonance region (see Fig. 12.2), information is added by such a measurement. This measurement was done with the ORELA, using "total energy detectors"[10] to detect the prompt gamma rays following an absorption event in the sample.

A major difficulty in additional capture cross-section measurements on the actinides would be obtaining pure isotopic samples. Since it is not possible to completely discriminate between capture and fission events, isotopes with high spontaneous fission rates must be excluded from the samples. Some actinides for which capture cross-section measurements would be feasible if proper samples were available would be ^{237}Np, ^{243}Am, ^{242m}Am, ^{245}Cm, and ^{247}Cm. The cost of the sample preparation for the isotopes other than ^{237}Np and perhaps ^{243}Am would be rather high and might need to be justified by more refined reactor sensitivity calculations than have presently been done.

In conclusion, capture cross-section measurements could be carried out on some of the nuclides that are important to actinide waste disposal and transmutation. The required effort and expense

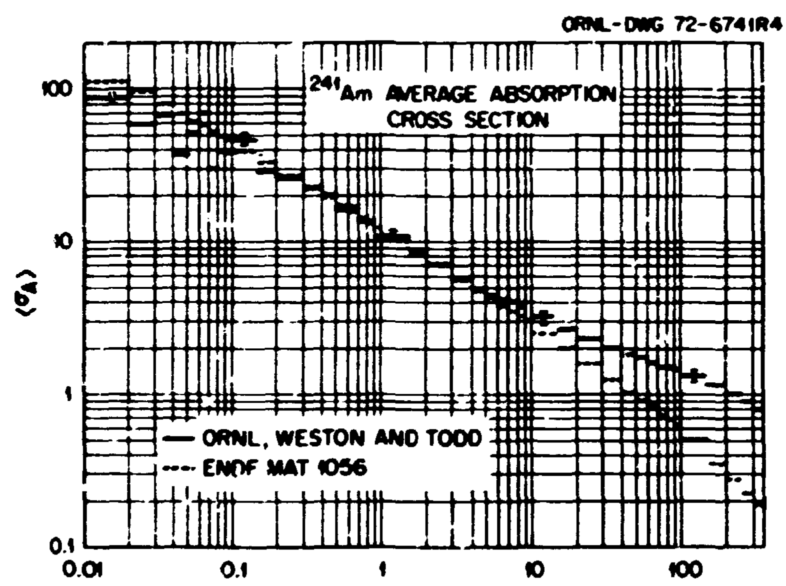

Fig. 12.1. The average absorption cross section of ^{241}Am between 0.01 and 350 keV compared to ENDF/B-IV, Mat 1056.

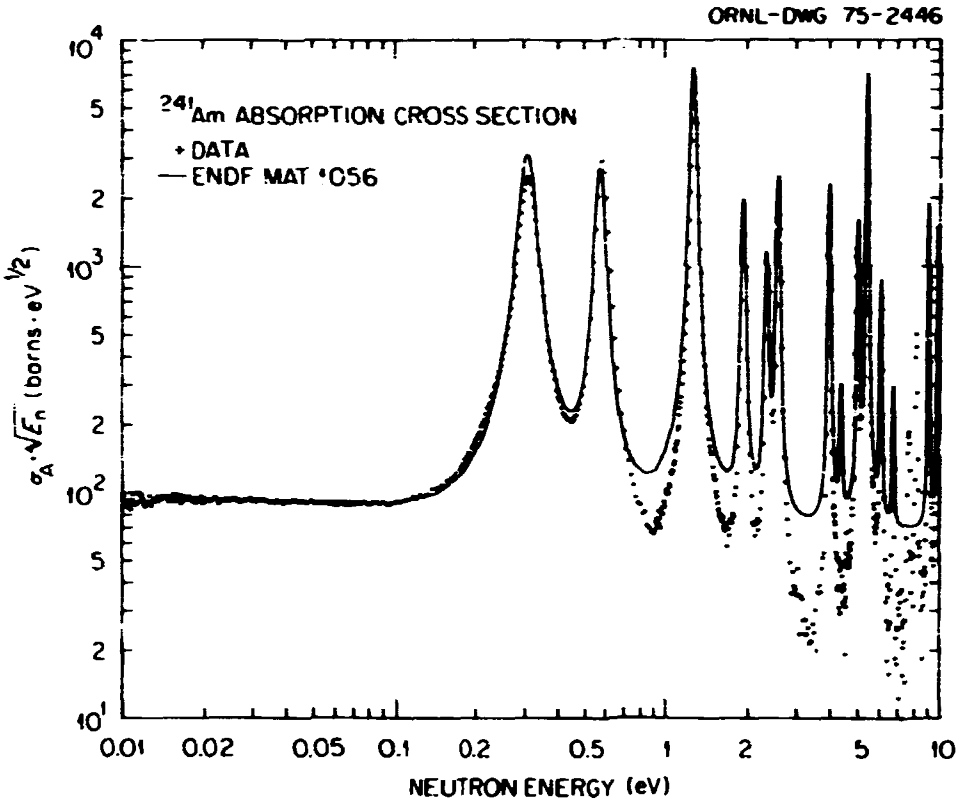

Fig. 12.2. The absorption cross section of ^{241}Am between 0.01 and 10 eV compared to ENDF/B-IV, Mat 1056.

would be rather high; however, the cost of surprises caused by misjudged cross sections could be much higher.

1. Contribution to the Advisory Group Meeting on Transactinium Isotopes Nuclear Data, Karlsruhe, Germany, November 3-7, 1975.

2. Research sponsored by ERDA's Division of Reactor Research and Development.

3. H. C. Claiborne, *Neutron-Induced Transmutation of High-Level Radioactive Waste*, ORNL-TM-3964 (December 1972).

4. H. C. Claiborne, *Effect of Actinide Removal on the Long-Term Hazard of High-Level Waste*, ORNL-TM-4724 (January 1975).

5. *High-level Radioactive Waste Management Alternatives*, BNWL-1900, vols. 1–4 (May 1974).

6. R. W. Benjamin, *Survey of Experimentally Determined Neutron Cross Sections of the Actinides*, DP-1324 (1973).

7. J. E. Lynn, *Systematics for Neutron Reactions of the Actinide Nuclei*, AERE-R-7468 (November 1974).

8. L. W. Weston and J. H. Todd, "Measurement of the Neutron Capture Cross Sections of the Actinides," *Proc. Conf. Nuclear Cross Sections and Technology*, March 3–7, 1975, Washington, D.C., to be published.

9. R. E. Schenter, Westinghouse-Hanford, private communication.

10. R. L. Macklin and J. H. Gibbons, "Capture-Cross-Section Studies for 30–220-keV Neutrons Using a New Technique," *Phys. Rev.* **159**, 1007 (1967).

13. SUBTHRESHOLD FISSION IN ^{240}Pu + n[1,2]

G. F. Auchampaugh[3] L. W. Weston

The neutron subthreshold fission cross section of ^{240}Pu has been measured from 500 to 10,000 eV using the ORELA neutron facility. The fission neutrons from the ≅10-g sample of $^{240}PuO_2$ were detected by two PSD NE-213 liquid scintillator detectors located out of the neutron beam. The sample was placed between them at a distance of 20.0 m from the neutron source. Both fission neutrons and gamma rays from capture and fission were stored simultaneously in separate regions of the data acquisition system. A total of 83 fission widths were obtained from area and shape analyses of those resonances defining the class II states at ≅782, ≅1405, ≅1936, and ≅2700 eV. The multilevel analysis of the 782-eV group required a very large fission width of 1.0 ± 0.1 eV for the 782-eV resonance. In addition, a broad background fission cross section was observed between the 791-and 810-eV resonances which could not be accounted for by the known resonances in the region. Similarly, the multilevel fits of the 1405- and 1936-eV groups required fission widths of several eV for the resonances at 1402, 1408, and 1936 eV (not seen in total). Details of the analysis will be presented.

1. Abstract of paper presented at Conference on Nuclear Cross Sections and Technology, Washington, D.C., March 3–7, 1975; *Bull. Am. Phys. Soc., Ser. II* 20(2), 158 (1975); full paper published in *Nuclear Cross Sections and Technology*, vol. II, ed. by R. A. Schrack and C. D. Bowman, NBS Special Publication 425, p. 560 (1975).
2. Division participation sponsored by ERDA's Division of Reactor Research and Development.
3. Los Alamos Scientific Laboratory.

14. PARAMETERS OF THE SUBTHRESHOLD FISSION STRUCTURE IN ^{240}Pu[1]

G. F. Auchampaugh[2] L. W. Weston[3]

The neutron subthreshold fission cross section of ^{240}Pu has been measured from 500 to 10,000 eV using the ORELA neutron facility. A total of 82 fission widths were obtained from area and shape analysis of those resonances which define the class II states at ≅782, ≅1405, ≅1936, and ≅2700 eV. The average square of the coupling matrix element for the first three class II states is 1.85 ± 1.43 eV^2 (SD). The average class II fission width is 2.5 ± 1.0 eV (SD). Approximately 22 clusters of class I resonances were observed below 10 keV, which results in a value of 450 ± 50 eV for the average class I level spacing. Assuming parabolic inner and outer barriers, the following barrier parameters were obtained:

$$V_A - B_n/\hbar\omega_A = 0.71\,^{+0.21}_{-0.09} \text{ and } V_B - B_n/\hbar\omega_B = 0.53\,^{+0.09}_{-0.06}$$

$[\,^{240}\mathrm{Pu}(n,f)\text{: } 500 \text{ to } 10{,}000\text{ eV}\text{; } \Gamma_{f\lambda}, \Gamma_{\lambda}^{II}, \langle H_{\lambda'\lambda''}^2\rangle\,]$.

1. Abstract of paper submitted for journal publication.
2. Los Alamos Scientific Laboratory.
3. Division participation sponsored by ERDA's Division of Reactor Research and Development.

15. EXPERIMENT FOR ACCURATE MEASUREMENTS OF FISSION PRODUCT ENERGY RELEASE FOR SHORT TIMES AFTER THERMAL-NEUTRON FISSION OF ^{235}U AND ^{239}Pu[1,2]

J. K. Dickens R. W. Peelle F. C. Maienschein

This memorandum describes an experimental program for definitive measurements of energy release from the gross mixture of fission products from fission of ^{235}U and ^{239}Pu by thermal neutrons. The safety program for light-water reactors, including those which will utilize recycled plutonium fuel, requires knowledge of the time-dependent heat source furnished by the beta and gamma rays from the decaying fission products. Depending on the method of uncertainty assessment, the emission rate in the critical time range about 100 sec following thermal-neutron fission of ^{235}U has an estimated standard error ranging from 8 to 15% with a probability between 1 and 5% of a deviation greater than 30% above the best estimate. The described program is designed to reduce this large uncertainty in the input heat term which would apply in a loss-of-coolant accident. A straightforward approach is outlined in which uncertainties are controlled by use of correct techniques, experimental tests to expose possible systematic errors, and replication of the trickiest parts of the task, using at least two different methods. It is anticipated that output uncertainties in the total energy release vs time of about 3% will be achieved using these approaches for the time range after fission between 2 and ~2000 sec. The upper end of the time range will be chosen such that at this time the energy release rates calculated using radiochemical data can be shown to be in detailed accord with the experimental observations. If it should become necessary to the applications, extension to the time region of a few tenths of a second could be made with somewhat increased uncertainty. Results will be obtained for both the differential and integral energy release functions. Beta- and gamma-ray energy spectra of moderate resolution will be obtained with ~7% precision. Possible follow-on extensions of the program are described.

1. Abstract of ORNL-TM-4676 (May 1975).
2. Research sponsored by the U.S. Nuclear Regulatory Commission.

16. FISSION PRODUCT BETA AND GAMMA ENERGY RELEASE QUARTERLY PROGRESS REPORT FOR JULY–SEPTEMBER 1975[1,2]

J. K. Dickens J. W. McConnell
T. A. Love J. F. Emery[3]
R. W. Peelle

Preliminary data for gamma-ray energy release from fission product decay following thermal-neutron fission of ^{235}U have been obtained and reduced for cooling times between 3 and 14,400 sec. The data were obtained as pulse-height spectra for photon energies between 0.05 and 8 MeV, using an NaI spectrometer, and were unfolded to give photon energy spectra of moderate resolution. These data are presented in graphical form for 27 combinations of irradiation, cooling, and counting times. Two irradiation times, t_i = 2.4 and 100.4 sec, were studied. The energy release values were obtained by integrating the spectra. In addition, photon yield data were obtained for $E\gamma$ between 0.025 and 0.05 MeV; they contribute insignificantly to the total energy release.

Preparations for beta-ray energy release measurements involved detailed study of the beta-ray detector's response and additional studies of electron scattering from various source configurations to arrive at an optimal design for the sample holder.

1. Abstract of ORNL-TM-5156 (in press).
2. Research Sponsored by the U.S. Nuclear Regulatory Commission.
3. Analytical Chemistry Division.

17. ORNL NEUTRON SCATTERING CROSS-SECTION MEASUREMENTS FROM 4 TO 8.5 MeV: A SUMMARY[1,2]

W. E. Kinney F. G. Perey

A program to measure neutron elastic and inelastic scattering cross sections at the ORNL Van de Graaff facility was completed with final documentation in June 1974. The data, taken over a period of five years on 26 nuclides from C to ^{238}U at incident neutron energies from 4 to 8.5 MeV, have added significantly to the store of measured neutron scattering cross sections. In many instances these are the only data on neutron inelastic scattering cross sections in this energy range. This summary is presented both to aid in evaluating the data and to catalogue them.

The data acquisition and reduction techniques are briefly described along with sources of error. The results are summarized with particular attention given to the degree of consistency and reproducibility, comparison with the results of others, and comparison with ENDF B.

1. Abstract of paper presented at Conference on Nuclear Cross Sections and Technology, Washington, D.C., March 3–7, 1975; *Bull. Am. Phys. Soc., Ser. II* 20(2), 172 (1975); full paper published in *Nuclear Cross Sections and Technology*, vol. II, ed. by R. A. Schrack and C. D. Bowman, NBS Special Publication 425, p. 883 (1975).
2. Research sponsored by ERDA's Division of Reactor Research and Development.

18. MEASUREMENT OF SECONDARY NEUTRONS AND GAMMA RAYS PRODUCED BY NEUTRON BOMBARDMENT OF WATER OVER THE INCIDENT ENERGY RANGE 1 TO 20 MeV[1,2]

G. L. Morgan

The spectra of secondary neutrons and gamma rays produced by neutron bombardment of a thick (~1 mean free path) sample of water have been measured as a function of the incident neutron energy over the range 1 to 20 MeV. Data were taken for angles of 90 and 140°. A linac (ORELA) was used as a neutron source with a 47-m flight path. Incident energy was determined by time of flight, while secondary spectra were determined by pulse-height unfolding techniques. The results of the measurements are presented in forms suitable for comparison with calculations based on the evaluated data files.

1. Abstract of ORNL-TM-5018 (August 1975).
2. Research sponsored by the Defense Nuclear Agency.

19. MEASUREMENT OF SECONDARY NEUTRONS AND GAMMA RAYS PRODUCED BY NEUTRON INTERACTIONS WITH NITROGEN AND OXYGEN OVER THE INCIDENT ENERGY RANGE 1 to 20 MeV[1,2]

G. L. Morgan

The spectra of secondary neutrons and gamma rays produced by neutron interaction in thick samples (~1 mean free path) of liquid nitrogen and liquid oxygen have been measured as a function of the incident neutron energy over the range 1 to 20 MeV. Data were taken for angles of 30, 55, 90, and 125°. A linac (ORELA) was used as a neutron source with a 47-m flight path. Incident energy was determined by time of flight, while secondary spectra were determined by pulse-height unfolding techniques. The results of the measurements are presented in forms suitable for comparison with calculations based on the evaluated data files.

1. Abstract of ORNL-TM-5023 (October 1975)
2. Research sponsored by the Defense Nuclear Agency

20. MEASUREMENT OF SECONDARY NEUTRONS AND GAMMA RAYS PRODUCED BY NEUTRON INTERACTIONS IN SILICON DIOXIDE OVER THE INCIDENT ENERGY RANGE 1 TO 20 MeV[1,2]

G. L. Morgan

The spectra of secondary neutrons and gamma rays produced by neutron interactions in a thick (≈1 mean free path) sample of silicon dioxide have been measured as a function of the incident neutron energy over the range 1 to 20 MeV. Data were taken at an angle of 90°. A linac (ORELA) was used as a neutron source with a 47-m flight path. Incident energy was determined by time of flight, while secondary spectra were determined by pulse-height unfolding techniques. The results of the measurement are presented in forms suitable for comparison with calculations based on the evaluated neutron data files.

1. Abstract of ORNL-TM-5024 (September 1975).
2. Research sponsored by the Defense Nuclear Agency.

21. MEASUREMENT OF SECONDARY NEUTRONS AND GAMMA RAYS PRODUCED BY NEUTRON INTERACTIONS IN ALUMINUM OVER THE INCIDENT ENERGY RANGE 1 TO 20 MeV[1,2]

G. L. Morgan

The spectra of secondary neutrons and gamma rays produced by neutron interaction in a thin sample (≈1/6 mean free path) of aluminum have been measured as a function of the incident neutron energy over the range 1 to 20 MeV. Data were taken at an angle of 125°. A linac (ORELA) was used as a neutron source with a 47-m flight path. Incident energy was determined by time of flight, while secondary spectra were determined by pulse-height unfolding techniques. The results of the measurements are presented in forms suitable for comparison with calculations based on the evaluated data files.

1. Abstract of ORNL-TM-5072 (in press)
2. Research sponsored by the Defense Nuclear Agency.

22. PRODUCTION OF LOW-ENERGY GAMMA RAYS BY NEUTRON INTERACTIONS WITH FLUORINE FOR INCIDENT NEUTRON ENERGIES BETWEEN 0.1 AND 20 MeV[1,2]

G. L. Morgan J. K. Dickens

Differential cross sections for the production of low-energy gamma rays (<240 keV) by neutron interactions in fluorine have been measured for neutron energies between 0.1 and 20 MeV with the

ORELA used as the neutron source. Gamma rays were detected at 92°, using an intrinsic germanium detector. Incident neutron energies were determined by time-of-flight techniques. Tables are presented for the production cross sections of three gamma rays having energies of 96, 110, and 197 keV.

1. Abstract of ORNL-TM-4823 (June 1975).
2. Research sponsored by the Defense Nuclear Agency.

23. GAMMA-RAY PRODUCTION MEASUREMENTS DUE TO INTERACTIONS OF NEUTRONS WITH ELEMENTS REQUIRED FOR NUCLEAR POWER APPLICATIONS AND DESIGN[1,2]

G. T. Chapman T. A. Love
J. K. Dickens G. L. Morgan
E. Newman[3]

For the past three years, neutron-induced gamma-ray-production cross-section measurements have been made for a variety of elements at the ORELA. A large well-shielded NaI spectrometer was used as the gamma-ray detector and the ORELA was the neutron source. The facility provides a consistent data set for neutron energies from 0.7 to 20 MeV and photon energies from 0.3 to 10.5 MeV. Typically, the samples are flat plates of the element ~0.02 atom b in thickness, although several elements studied required samples in compound form. The data are accumulated in a two-parameter-array, gamma-ray pulse height vs neutron time of flight. Data reduction was accomplished by binning in desired neutron energy groups and in fixed photon energy groups. For each neutron energy group, the data were unfolded using the FERD unfolding routine. The results are in the form of absolute differential cross sections, $d^2\sigma/d\omega\, dE$, for each photon energy bin. So far, data have been obtained for 20 elements: Li, C, N, O, F, Mg, Al, Si, Ca, Fe, Ni, Cu, Zn, Nb, Ag, Sn, Ta, W, Au, and Pb.

1. Abstract of paper presented at Conference on Nuclear Cross Sections and Technology, Washington, D.C., March 3–7, 1975; *Bull Am. Phys. Soc., Ser. II* **20**(2), 166 (1975); full paper published in *Nuclear Cross Sections and Technology*, vol. II, ed. by R. A. Schrack and C. D. Bowman, NBS Special Publication 425, p. 758 (1975).
2. Research sponsored by the Defense Nuclear Agency.
3. Physics Division.

24. NEUTRON-INDUCED GAMMA-RAY PRODUCTION IN ZINC FOR INCIDENT NEUTRONS ENERGIES OF 4.9, 5.4, AND 5.9 MeV[1,2]

J. K. Dickens

Interactions of neutrons with zinc have been studied by measuring gamma-ray-production cross sections. For a sample of natural zinc, spectra were obtained for incident mean neutron energies, E_n = 4.9, 5.4, and 5.9 MeV, with gamma-ray detector systems utilizing coaxial Ge(Li) detectors. Nearly monoenergetic neutrons were obtained from the D(*d,n*) reaction, using deuterons obtained from the (pulsed) ORNL 5-MV Van de Graaff accelerator. Time of flight was used to discriminate against pulses due to neutrons and background radiation. Gamma-ray identification was aided by obtaining spectra for samples enriched in the isotopes ^{64}Zn and ^{68}Zn, and new information on the level structure of ^{64}Zn is reported. These cross sections have been compared, where possible, with previous comparable measurements with generally satisfactorily results.

1. Abstract published paper: *Nucl. Sci. Eng.* **58**, 331 (1975).
2. Research sponsored by ERDA's Division of Reactor Research and Development.

25. GAMMA-RAY PRODUCTION DUE TO NEUTRON INTERACTIONS WITH SILVER FOR INCIDENT NEUTRON ENERGIES BETWEEN 0.3 AND 20 MeV: TABULATED DIFFERENTIAL CROSS SECTIONS[1,2]

J. K. Dickens T. A. Love G. L. Morgan

Numerical values of differential cross sections for gamma rays produced by neutron reactions with silver have been obtained for neutron energies between 0.3 and 20 MeV for $\theta_\gamma = 125°$. The $d^2\sigma/d\omega\, dE$ values were obtained using an NaI spectrometer. These data are presented as gamma-ray-production group cross-section values of $d^2\sigma/d\omega\, dE$ for $0.3 \leqslant E_\gamma \leqslant 10.5$ MeV, with gamma-ray intervals ranging from 15 keV for $E_\gamma \leqslant 0.4$ MeV to 160 keV for $E_\gamma \sim 9$ MeV. Neutron energy intervals varied from 0.10 MeV for E_n between 0.3 and 0.4 MeV to 3 MeV for E_n between 14 and 20 MeV.

1. Abstract of ORNL-TM-5081 (in press).
2. Research sponsored by the Defense Nuclear Agency.

26. THE Au($n,x\gamma$) REACTION CROSS SECTION FOR INCIDENT NEUTRON ENERGIES BETWEEN 0.2 AND 20.0 MeV[1,2]

G. L. Morgan E. Newman[3]

Differential cross sections for the neutron-induced gamma-ray production from natural gold have been measured for incident neutron energies between 0.2 and 20.0 MeV. The ORELA was used to provide the neutrons, and an NaI spectrometer was used to detect the gamma rays at 125°. The data presented are the double differential cross section, $d^2\sigma/d\Omega\, dE$, for gamma-ray energies between 0.3 and 10.6 MeV for coarse intervals in incident neutron energy. The integrated yield of gamma rays of energies greater than 300 keV and higher resolution in the neutron energy is also presented. The experimental results are compared with the Evaluated Neutron Data Library.

1. Abstract of ORNL-TM-4973 (August 1975).
2. Research sponsored by the Defense Nuclear Agency.
3. Physics Division.

27. CROSS SECTIONS FOR THE PRODUCTION OF LOW-ENERGY PHOTONS BY NEUTRON INTERACTIONS WITH FLUORINE AND TANTALUM[1,2]

J. K. Dickens G. L. Morgan F. G. Perey

A system has been developed using the ORELA as a neutron source in conjunction with a 200-mm^2 intrinsic germanium detector for measurement of low-energy photons from neutron interactions. A 47-m flight path is employed so that two-parameter data acquisition permits measurement of photon spectra as a function of incident neutron energy over the range 0.1 to 20.0 MeV.

Cross sections for the production of photons of energies 110 and 197 keV by neutron elastic scattering in fluorine have been obtained. Comparisons will be made with existing data. Cross sections for the production of a 96-keV photon from the $^{19}F(n,p)^{19}O$ reaction will demonstrate the sensitivity of the system.

Measurements will also be presented for the production of low-energy gamma rays by neutron inelastic scattering and capture in tantalum. Cross sections are also given for the production of K x rays by internal conversion of states populated in the reactions mentioned above.

1. Abstract of paper presented at Conference on Nuclear Cross Sections and Technology, Washington, D.C., March 3–7, 1975; *Bull. Am. Phys. Soc., Ser. II* 20(2), 166 (1975); full paper published in *Nuclear Cross Sections and Technology*, vol. II, ed. by R. A. Schrack and C. D. Bowman, NBS Special Publication 425, p. 762 (1975).
2. Research sponsored by Defense Nuclear Agency.

28. THE Nb($n,x\gamma$) REACTION CROSS SECTION FOR INCIDENT NEUTRON ENERGIES BETWEEN 0.65 AND 20.0 MeV[1,2]

J. K. Dickens G. L. Morgan E. Newman[3]

Differential cross sections for the neutron-induced gamma-ray production from niobium have been measured for neutron energies between 0.65 and 20 MeV. The ORELA was used to provide the neutrons, and an NaI spectrometer was used to detect the gamma rays at 90°. The data presented are the double differential cross section, $d^2\sigma/d\Omega\, dE$, for gamma-ray energies between 0.75 and 10.5 MeV and coarse intervals in the incident neutron energy. The integrated yield of gamma rays of energies greater than 0.75 MeV and higher resolution in the incident neutron energy is also presented. The experimentally determined results are compared with the Evaluated Neutron Data Library.

1. Abstract of ORNL-TM-4972 (ENDF-219) (September 1975).
2. Research sponsored by ERDA's Division of Controlled Thermonuclear Research.
3. Computer Sciences Division, UCC Nuclear Division.

29. THE Pb($n,x\gamma$) REACTION FOR INCIDENT NEUTRON ENERGIES BETWEEN 0.6 AND 20.0 MeV[1,2]

G. T. Chapman G. L. Morgan

Numerical values of the differential cross sections for gamma rays produced by neutron reactions with natural lead have been measured for neutron energies between 0.6 and 20 MeV. The data were obtained using an electron linac as a neutron source in conjunction with an NaI spectrometer located at an angle of 125°. The data consist of the doubly differential cross section, $d^2\sigma/d\Omega\, dE$, for gamma rays between 0.3 and 10.5 MeV for coarse intervals in incident neutron energy and the integrated yield of gamma rays of energy greater than 0.3 MeV with high resolution in the incident neutron energy. The measured results are compared with the current ENDF/B data.

1. Abstract of ORNL-TM-4822 (February 1975).
2. *Research sponsored by the Defense Nuclear Agency.*

30. NEUTRON RADIATIVE CAPTURE AND TOTAL CROSS SECTION OF $^{50,52,53}Cr$, $^{54,57}Fe$, AND $^{62,64}Ni$[1,2]

H. Beer[3]   R. R. Spencer

Enriched samples of $^{50,52,53}Cr$, $^{54,57}Fe$, and $^{62,64}Ni$ were measured in capture and, with the exception of ^{53}Cr and ^{57}Fe, in transmission in the energy ranges 5 to 200 and 10 to 300 keV respectively. The capture and total cross-section data analyzed provided resonance parameters for both *s*-wave and $l > 0$ wave resonances. The *s*- and *p*-wave strength functions were determined, and neutron width and level spacing distributions were established. A positive correlation was found between the radiation widths and the reduced neutron widths.

1 Abstract of published paper: *Nucl. Phys.* **A240**, 29 (1975).
2 Work performed at Karlsruhe, West Germany.
3 Kernforschungszentrum, Karlsruhe, West Germany.

31. THICK-SAMPLE TRANSMISSION MEASUREMENTS AND RESONANCE ANALYSIS OF THE TOTAL NEUTRON CROSS SECTION OF IRON[1,2]

S. Cierjacks[3]   R. Topke[3]
G. Schmalz[3]   R. R. Spencer
F. Voss[3]

The deep minima in the total neutron cross section of iron in the many-keV range have not satisfactorily been explored in the past. Better experimental results are of importance in connection with shielding applications and some background aspects of filtered beam techniques. For this reason new transmission measurements for this element were performed at the 190-m flight path of the Karlsruhe fast-neutron spectrometer, allowing for an improved resolution of 0.015 ns/m. The measurements were carried out in the energy range 0.5 to 30 MeV using two different sample thicknesses of 0.458 atom/b and of 0.109 atom/b respectively. The thick-sample results indicate that the deep *s*-wave minima are now fully explored up to the inelastic threshold at about 850 keV. From the highly resolved data, neutron resonance data were determined by a multilevel *R*-matrix fit to both transmission curves simultaneously. The results of the measurements between 0.5 and 30 MeV and the resonance parameters determined in the range between 500 and 850 keV are presented.

1 Abstract of paper presented at Conference on Nuclear Cross Sections and Technology, Washington, D.C., March 3–7, 1975; *Bull. Am. Phys. Soc., Ser. II* **20**(2), 166 (1975); full paper published in *Nuclear Cross Sections and Technology*, vol. II, ed. by R. A. Schrack and C. D. Bowman, NBS Special Publication 425, p. 754 (1975).
2 Work performed at Karlsruhe, West Germany.
3 Kernforschungszentrum, Karlsruhe, West Germany.

32. MEASUREMENT OF NEUTRON TOTAL CROSS SECTION OF ^{19}F[1,2]

C. H. Johnson[3]   D. C. Larson   J. A. Harvey[3]

We have measured the neutron total cross section of ^{19}F from 20 keV to 10 MeV by observing the transmission through teflon (CF_2) relative to a matching carbon sample. The detector was plastic

NE-110 located at the 200-m time-of-flight station of the ORELA. The burst width of the electron beam was 5 nsec. Seventeen resonances up to 1200 keV are observed: only one of these, a resonance at 309 keV, is so narrow (0.4 keV) as to be comparable to the resolution width (0.2 keV). Resonance energies, widths, and $g_J\Gamma_n$ have been extracted for the isolated levels. Below the inelastic threshold at 116 keV there are three resonances with energies and J^π values of 27 keV, 2^-; 49 keV, 1^-; and 98 keV, 1^-. Above the inelastic threshold, only lower limits on the spin J of the remaining states have been assigned from extracted values of $g_J\Gamma_n\Gamma$. No analysis has been attempted for the complex structure above 1200 keV. No evidence was found for 50 or so small peaks below 300 keV as reported by Hibdon.[4]

1. Abstract of paper presented at American Physical Society Meeting, Pittsburgh, Pennsylvania, October 31–November 2, 1974.

2. Division participation sponsored by the Defense Nuclear Agency.

3. Physics Division.

4. C. T. Hibdon, *Phys. Rev.* 133, B353 (1964).

33. DEVELOPMENT OF A TWO-STEP HAUSER-FESHBACH CODE WITH PRECOMPOUND DECAYS AND GAMMA-RAY CASCADES—A THEORETICAL TOOL FOR CROSS-SECTION EVALUATIONS[1,2]

C. Y. Fu

In order to refine our theoretical tools for meeting future data needs, particularly for cross-section evaluations aimed at fusion reactor study and weapons effects analysis, we have recently developed a two-step Hauser-Feshbach code with precompound decays and gamma-ray cascades. The Hauser-Feshbach formula for compound binary-reaction cross sections is extended to treat the emission of a second particle, assuming sequential decays without correlation. A precompound model is used when necessary to modify the energy-differential cross sections of binary reactions. Gamma-ray competition with the second outgoing particle and gamma-ray cascades are calculated separately for each initial spin and parity state to conserve angular momentum. Calculated excitation functions for ^{40}Ca(n,pn) and ^{19}F($n,n\alpha$) show variations of a factor of 3 with the level spins. Calculated ^{40}Ca ($n,np\gamma$) cross sections at 18.5 MeV show a factor-of-4 improvement, in agreement with experiment over that previously obtained with a Weisskopf model.

1. Abstract of paper presented at Conference on Nuclear Cross Sections and Technology, Washington, D.C., March 3–7, 1975; *Bull. Am. Phys. Soc., Ser. II* 20(2), 149 (1975); full paper published in *Nuclear Cross Sections and Technology*, vol. I, ed. by R. A. Schrack and C. D. Bowman, NBS Special Publication 425, p. 328 (1975).

2. Research sponsored by Defense Nuclear Agency.

34. USE OF NUCLEAR REACTION MODELS IN EVALUATING GAMMA-RAY PRODUCTION DATA[1,2]

F. G. Perey

An extensive body of neutron-induced gamma-ray production cross sections has been collected recently as a result of major advances in experimental techniques.[3] The use of nuclear-reaction theories in evaluating these results has proved surprisingly successful,[4] and the excellent agreement suggests that these reaction models can be used to generate cross sections for materials and targets for which data are not feasible or not yet available.

1. Abstract of invited paper presented at American Physical Society Meeting, Pittsburgh, Pennsylvania, October 31–November 2, 1974.

2. Research sponsored by Defense Nuclear Agency.

3. J. K. Dickens, G. L. Morgan, and F. G. Perey, "Neutron-Induced Gamma-Ray Production in Iron for the Energy Range $0.8 \leq E_n \leq 20$ MeV," *Nucl. Sci. Eng.* **50**, 311–36 (1973).

4. C. Y. Fu and D. C. Larson, private communication.

35. DESCRIPTION OF THE ENDF/B-IV SILICON EVALUATION[1,2]

D. Larson

The ENDF/B-IV version of the silicon evaluation is considered as an example of an evaluation which includes angle and energy distributions of outgoing charged particles. Relevant optical-model parameters were determined by careful fitting to available elastic-scattering data. Competition among open channels was calculated using the Hauser-Feshbach formalism, including direct reaction components where appropriate. Gilbert-Cameron level density parameters were modified to reproduce available data and also checked against predictions from shell-model calculations. Gamma-ray production for binary channels was calculated from the Hauser-Feshbach results, with the branching ratios taken from the literature and shell-model calculations. The tertiary reaction cross sections and associated gamma-ray productions were calculated using a statistical model with empirical angular momentum corrections. Energy distributions of both neutrons and charged particles are given as probability distributions and are obtained from the Hauser-Feshbach and tertiary calculations. Comparisons of the evaluation to available data will be shown.

1. Abstract of paper presented at the Conference on Nuclear Cross Sections and Technology, Washington, D.C., March 3–7, 1975; *Bull. Am. Phys. Soc., Ser. II* **20**(2), 148 (1975); full paper published in *Nuclear Cross Sections and Technology*, vol. 1, ed. by R. A. Schrack and C. D. Bowman, NBS Special Publication 425, p. 317 (1975).

2. Research sponsored by the Defense Nuclear Agency.

36. REPORT TO THE U.S. NUCLEAR DATA COMMITTEE[1,2]

F. G. Perey

This is a collection of recent, not generally publicized, cross-section work done at ORNL and recent papers which have been submitted for publication. This report is prepared for the U.S. Nuclear Data Committee and includes only that material which the author thinks might be of interest to the USNDC subcommittee members. The memo covers the period of time since the last USNDC meeting.

1. Abstract of ORNL-TM-4885 (April 1975).

2. Research sponsored by ERDA's Division of Physical Research.

37. AN EXPERIMENTAL SYSTEM FOR PROVIDING DATA TO TEST EVALUATED SECONDARY NEUTRON AND GAMMA-RAY-PRODUCTION CROSS SECTIONS OVER THE INCIDENT NEUTRON ENERGY RANGE FROM 1 TO 20 MeV[1,2]

G. L. Morgan T. A. Love F. G. Perey

A system is described which allows simultaneous measurement of secondary neutron and gamma-ray-production cross sections. Measurements can be made rapidly over wide energy ranges. An electron linac is used as a neutron source. Annular scattering samples located 47 m from the neutron source are viewed by an NE-213 scintillation counter. Multiparameter data acquisition is done by on-line computers for incident neutron energies from 1 to 20 MeV.

1. Abstract of a paper to be published in *Nucl. Instr. Methods* 127.
2. Research sponsored by the Defense Nuclear Agency.

38. HIGH-ENERGY GAMMA-RAY TRANSITIONS OF ^{56}Fe RESONANCES IN THE ENERGY RANGE 7 TO 70 keV[1,2]

H. Beer[3] R. R. Spencer F. Kaeppeler[3]

High-energy gamma-ray transitions to low-lying states in ^{57}Fe following neutron capture in ^{56}Fe were investigated for individual resonances. The Karlsruhe 3-MW pulsed Van de Graaff accelerator was used as a source of kinematically collimated neutrons from the ^{7}Li$(p,n)^7$Be reaction, and a 50-cc Ge(Li) detector served for gamma-ray detection. Neutron energies were determined in a time-of-flight experiment with a flight path of 44 cm and a time resolution of 10 nsec m. The neutron flux at a repetition rate of 2.5 MHz was sufficient to carry out the measurement with a sample consisting of only 56.5 g of natural iron. With this sample the multiple-scattering corrections are small enough, even for the broad *s*-wave resonances, to deduce quantitative values for the partial radiation widths. Five partial radiation widths of four resonances were so determined with an uncertainty of about 20 to 30%. For the ^{56}Fe *s*-wave resonance at 27.7 keV, the ^{57}Fe ground-state transition and the transition to the 14-keV first excited state were observed. Three $l > 0$ wave resonances at 34.25, 38.38, and 59.25 keV showed a strong high-energy transition to the ground state or the first excited state.

1. Abstract of paper presented at Conference on Nuclear Cross Sections and Technology, Washington, D.C., March 3–7, 1975; *Bull. Am. Phys. Soc., Ser. II* 20(2), 169 (1975); full paper published in *Nuclear Cross Sections and Technology*, vol II, ed. by R. A. Schrack and C. D. Bowman, NBS Special Publication 425, p. 816 (1975).
2. Work performed at Karlsruhe, West Germany.
3. Kernforschungszentrum, Karlsruhe, West Germany.

39. GAMMA-RAY TRANSITIONS IN ^{181}Ta OBSERVED IN ^{181}Ta$(n,n'\gamma)$ REACTIONS[1,2]

J. K. Dickens G. G. Slaughter[3]

The level structure of ^{181}Ta has been studied by observing gamma radiation due to inelastic neutron scattering by Ta. Of the 131 gamma rays (or groups of unresolved gamma rays) observed, 60 could be placed as transitions among 42 levels in $^{?}$Ta. These include 24 new levels having $E_x > 990$ keV and verification of previously reported results. Possible quantum assignments based on the Nilsson single-particle rotational bands are suggested for four of the new levels.

1. Abstract of ORNL-TM-[illegible] (June 1975).
2. Research sponsored by the Defense Nuclear Agency.
3. Physics Division.

40. MICROSCOPIC INTERPRETATION OF INELASTIC PROTON SCATTERING FROM ^{138}Ba AND ^{144}Sm[1,2]

D. C. Larson S. M. Austin B. H. Wildenthal

Microscopic DWBA calculations for the (p,p') reaction at $E_p = 30$ MeV have been performed for 2^+, 4^+, and 6^+ states in ^{138}Ba and ^{144}Sm. Large-basis shell-model wave functions were used to describe the nuclear states. The effective two-body interaction mediating the inelastic scattering was obtained from a recent survey of inelastic-scattering analysis. Polarization charges for the nucleons were extracted using a scheme based on the model of Atkinson and Madsen. Two sets of shell-model wave functions were employed for the ^{138}Ba calculation, and it was found that comparison of the inelastic proton scattering calculations with experiment clearly distinguished between the two sets. For the better wave functions, the polarization charges were essentially independent of state and multipole.

1. Abstract of published paper: *Phys. Rev.* C11(5), 1638 (1975).
2. Research sponsored by the National Science Foundation and performed at Michigan State University.
3. Michigan State University.

41. ELASTIC AND INELASTIC SCATTERING[1,2]

F. G. Perey

Elastic scattering is the largest of all partial cross sections when a light ion interacts with nuclei. When the energy of the incident particle is sufficiently above the Coulomb barrier, the elastic angular distribution is dominated by a diffraction-like pattern. Nearly 20 years ago it was realized that this phenomenon is due to the combination of the finite size of the nucleus and the fact that nuclear matter is "partially transparent." The optical model, which consists in replacing the many-body problem involved by a simple complex effective potential in a one-body Schroedinger equation, was developed to explain elastic scattering. Since then much work has been done on this optical model of elastic scattering, and most of it falls into two distinct categories. The first deals with attempting to compute this effective potential from considerations of the many-body problem and is highly theoretical in nature. The second is purely phenomenological and deals with the empirical determination of this optical-model potential from a fitting procedure, using experimental elastic-scattering data. Aside from its intrinsic interest, the study of the optical-model potential is motivated by the important role it plays in the interpretation of many nuclear reactions. This is due to the fact that the refraction and reflections introduced by elastic scattering, in the form of distortions of the incident and outgoing waves in a reaction, are very large and cannot be neglected.

In this chapter our aim will be to emphasize what in our opinion are current problems encountered in optical-model analyses. We discuss the theoretical framework, which is intended only to bring into focus some of the important aspects of the optical model, deal with the analysis of elastic-scattering data, and report on some systematic studies which could be helpful in the analysis of nuclear reactions. We also consider extensions of the optical model to inelastic scattering.

1. From introduction to a section included in Chapter IV.B of *Nuclear Spectroscopy and Reactions*, Part B, p. 137, Academic Press, New York, 1974.
2. Research sponsored by ERDA's Division of Physical Research.

42. SURVEY OF COMPUTER CODES WHICH PRODUCE MULTIGROUP DATA FROM ENDF/B-IV[1,2]

N. M. Greene[3]

A large fraction of today's sophisticated nuclear analysis is based on multigroup methods. Unfortunately, this approach requires that a basic nuclear data set be compromised to make ready for the multigroup calculation.

This paper contrasts the features of several code systems that produce multigroup neutron data—primarily the MC^2-2/SDX, MINX/SPHINX, and AMPX code packages. These systems are all committed to maintaining full processing capabilities with the current evaluated nuclear data files, ENDF/B.

MC^2-2/SDX and MINX/SPHINX were developed specifically for fast-reactor analysis, though both have options that can address a wider range of application. AMPX was developed primarily for shielding applications and contains techniques that have been widely used in light-water and molten-salt reactor work. AMPX also addresses the gamma-ray transport problem.

Aside from differences in their heritages, the three systems have philosophical differences. Two of the systems must preprocess an ENDF library before a multigroup library can be made. MC^2-2 calculates a very detailed weighting spectrum over which to weight basic data; the other systems generally start with a "guess" for weighting. MC^2-2/SDX and MINX/SPHINX generally use resonance treatments based on narrow resonance approximations, though other options are available. AMPX has cruder narrow resonance treatments and has relied heavily on the Nordheim treatment in the past.

This paper explores some of these differences and their implications for particular applications.

1. Abstract of invited paper presented at Conference on Nuclear Cross Sections and Technology, Washington, D.C., March 3–7, 1975; *Bull. Am. Phys. Soc., Ser. II* 20(2), 170 (1975); full paper published in *Nuclear Cross Sections and Technology, vol. II*, ed. by R. A. Schrack and C. D. Bowman, NBS Special Publication 425, p. 848 (1975).
2. Research sponsored by Defense Nuclear Agency.
3. Computer Sciences Division, UCC Nuclear Division.

43. TABULAR CROSS-SECTION FILE GENERATION AND UTILIZATION TECHNIQUES[1,2]

D. E. Cullen[3] O. Ozer[4] C. R. Weisbin

Many present applications of cross-section data requires that the data be in a tabular, pointwise form with a linear functional dependence assumed between energy points. Such files are generated from ENDF/B within a controlled error criterion. Although a tabular representation simplifies requirements for processing codes further downstream, the large amounts of data involved necessitate the use of optimization techniques during the creation and utilization of the files.

Since ENDF/B cross sections are represented either in terms of resonance parameters or as tables with possible nonlinear interpolation laws, two problems are considered during the creation of a linear file: Resonance data is processed into a tabular form with the use of an algorithm designed to ensure the creation of a minimum number of points within the controlled error criterion. The corresponding procedure for the linearization of the ENDF/B tables outside the resonance range is dependent on an analytic approach.

After a high-accuracy general-purpose tabular library has been established, individual applications will require data with considerably less stringent error requirements. For these applications the files are thinned, using techniques dependent on all points within an interval.

Finally, a paging technique designed to enable existing codes to process large data tables is described.

1. Abstract of paper presented at Conference on Nuclear Cross Sections and Technology, Washington, D.C., March 3–7, 1975; *Bull. Am. Phys. Soc., Ser. II* **20**(2), 152 (1975); full paper published in *Nuclear Cross Sections and Technology, vol. I,* ed. by R. A. Schrack and C. D. Bowman, NBS Special Publication 425, p. 419 (1975).

2. Division participation sponsored by ERDA's Division of Reactor Research and Development.

3. Lawrence Livermore Laboratory.

4. Electric Power Research Institute, Palo Alto, California.

44. PENDF: A LIBRARY OF NUCLEAR DATA FOR MONTE CARLO CALCULATIONS DERIVED FROM DATA IN THE ENDF/B FORMAT[1,2]

R. J. LaBauve[3] R. E. Seamon[3] D. R. Harris[3]
C. R. Weisbin M. E. Battat[3] P. G. Young[3]
M. M. Klein[3]

A specialized pointwise-evaluated nuclear data library in a format suited for input to continuous-energy Monte Carlo codes has been developed from the ENDF/B at the Los Alamos Scientific Laboratory. Prominent features of the data for a particular nuclide are as follows: (1) A complete description of all alterations made to the original ENDF/B data is contained in the file. (2) All resonance parameters are processed into pointwise cross-section data for a specified temperature. (3) The cross-section data for all reactions are placed on a common energy mesh sufficiently dense to allow linear-linear interpolation between points to some specified tolerance. Excess data points are thinned to a specified tolerance. (4) Angular distributions of secondary neutrons are expressed as scattering angle cosine vs probability tabulation in which the angle cosine points are given at equal probability intervals. (5) If the energy distribution of secondary neutrons is expressed as a tabulation of secondary energy vs probability in the original ENDF/B file, the secondary energy points are also recalculated at equal probability intervals as in (4). (6) All gamma-ray production cross sections and gamma-ray spectra are summed and expressed as a single isotropic, nonelastic reaction. The gamma-ray production cross sections are given on the same energy mesh as the other reactions.

1. Abstract of Los Alamos Scientific Laboratory Report LA-5687 (October 1974).

2. Work performed at Los Alamos Scientific Laboratory.

3. Los Alamos Scientific Laboratory.

45. SPECIFICATION FOR PSEUDO-COMPOSITION-INDEPENDENT FINE-GROUP AND COMPOSITION-DEPENDENT FINE- AND BROAD-GROUP LMFBR NEUTRON-GAMMA LIBRARIES AT ORNL[1,2]

C. R. Weisbin R. W. Roussin
J. E. White[3] R. Q. Wright

Specifications are presented for pseudo-composition-independent fine-group and broad-group LMFBR neutron-gamma libraries. A one-dimensional cylindrical model of the CRBRP is used to derive the fine- and broad-group composition-dependent data. Energy boundary selection was guided by sensitivity considerations and compatibility with previously existing multigroup sets. Fast-reactor data testing of the Cross Section Evaluation Working Group benchmark critical assemblies is summarized.

1. Abstract of ORNL-TM-5142 (ENDF-224) (in press).
2. Research sponsored by ERDA's Division of Reactor Research and Development.
3. Computer Sciences Division, UCC Nuclear Division.

46. COUPLED 100-GROUP NEUTRON AND 21-GROUP GAMMA-RAY CROSS SECTIONS FOR EPR CALCULATIONS[1,2]

D. M. Plaster[3] R. T. Santoro W. E. Ford III[3]

Coupled 100-group neutron and 21-group gamma-ray 800° K cross sections have been generated using the latest ENDF/B cross-section data for H, He, ^{6}Li, ^{7}Li, Be, ^{10}B, ^{11}B, C, O, F, Al, Ti, V, Cr, ^{55}Mn, Fe, ^{59}Co, Ni, Cu, Nb, and Pb. The procedures used to generate these cross sections and the organization of the cross-section tape in ANISN format are discussed.

1. Abstract of ORNL-TM-4872 (April 1975).
2. Research sponsored by ERDA's Division of Controlled Thermonuclear Research.
3. Computer Sciences Division, UCC Nuclear Division.

47. DEVELOPMENT, GENERATION, AND TESTING OF THE DCTR FINE-GROUP CROSS-SECTION LIBRARY[1,2]

R. W. Roussin J. E. White[3]
C. R. Weisbin N. M. Greene[3]
R. H. Johnson[4]

The Radiation Shielding Information Center, ORNL, collaborated with various people at the University of Wisconsin, Princeton Plasma Physics Laboratory, University of Texas, Battelle Northwest, Los Alamos Scientific Laboratory, Argonne National Laboratory, and ORNL to develop specifications for a general-purpose multigroup cross-section library to be used by contractors of ERDA's Division of Controlled Thermonuclear Research in various neutronics studies. As a consequence, a coupled neutron-gamma-ray cross-section library which utilizes the Bondarenko[5] approach for specifying temperature dependence and self-shielding was generated at ORNL. A validation effort is under way, and preliminary results of calculations have been obtained.

The neutron energy group structure is a subset of the CSEWG standard[6] and contains 171 groups spanning the energy range from 20.0 MeV to 10^{-5} eV. The weighting function contains a Maxwellian-weighted thermal group region, a $1/E$ slowing-down spectrum tied to a fission spectrum and joined at high energies to a D-T-broadened 14-MeV spectrum.

All cross sections were generated at temperatures of 300, 900, and 2100° K. The materials included are: ^{1}H, ^{2}H, ^{3}H, ^{4}He, ^{6}Li, ^{7}Li, ^{9}Be, ^{10}B, ^{11}B, ^{12}C, ^{14}N, ^{16}O, F, ^{23}Na, Mg, ^{27}Al, Si, P, S, K, Ca, Ti, V, Cr, ^{55}Mn, Fe, ^{59}Co, Ni, ^{63}Cu, ^{65}Cu, Cu, Zr, ^{93}Nb, Mo, ^{107}Ag, ^{109}Ag, Cd, Sn, ^{151}Eu, ^{152}Eu, ^{153}Eu, ^{154}Eu, ^{181}Ta, ^{182}W, ^{183}W, ^{184}W, ^{186}W, Pb, ^{232}Th, ^{233}U, ^{234}U, ^{235}U, ^{236}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, and ^{241}Am. All the cross sections were generated from ENDF/B-IV data where possible. Values of σ_0 used ranged from 0.1 to 10^5.

The cross-section library is available in AMPX interface format, which retains all the individual reaction cross sections and transfer matrices. The neutron multigroup cross sections and transfer matrices, as well as Bondarenko factors, were generated using the ORNL version of the MINX[8] cross-section processor, since it has the capability of producing output in AMPX interface format. The gamma-ray production and interaction cross sections were generated by the LAPHNCAS and SMUG modules of AMPX. Various modules of AMPX have been modified to accept the expanded AMPX interface format, which includes the Bondarenko factors. A new module, BONAMI, was written to read the AMPX interface format and to interpolate on Bondarenko factors. The library will be distributed with various retrieval modules for group collapsing and coupling and for producing working cross-section libraries. The library will also be available in CCCC format[9] as well as working formats such as ANISN.[10]

Results of the first calculation made with the Library are shown in Fig. 47.1, which compares measured and calculated neutron leakage spectra for the case of an iron spherical shell surrounding

ORNL-DWG 75-16099

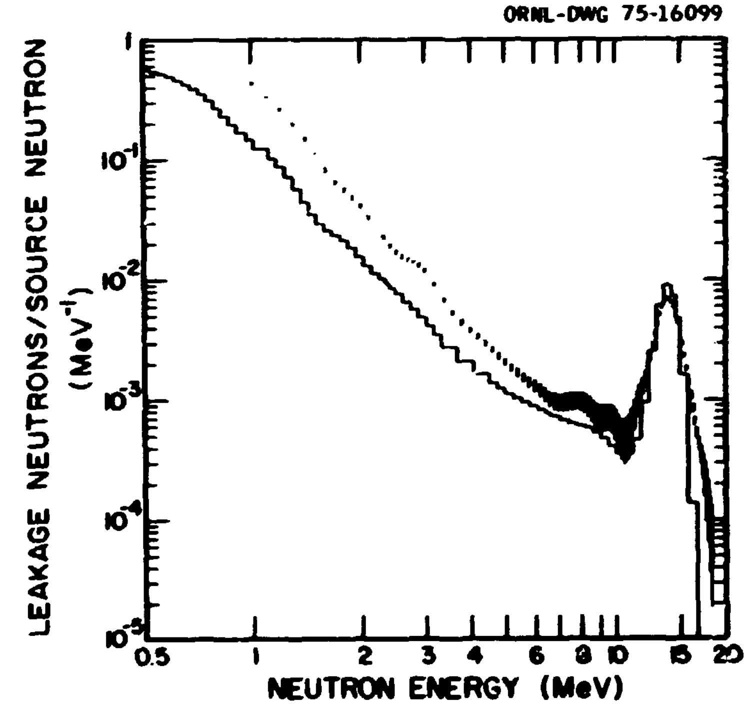

Fig. 47.1. Measured and calculated leakage spectra for a D-T source in a 30-cm thick iron spherical sphere.

a D-T source. The shell thickness is 30 cm. and its outside radius is 38.1 cm. The measurements are from an experiment performed at the University of Illinois.[11] These calculated results are consistent with similar calculations performed for the experiment using multigroup data generated with a different neutron processor.

A program of validation is under way to verify the multigroup library and the various retrieval programs which are a part of the CTR package. Several installations are performing calculations on selected integral experiments and reference CTR designs. Results will be collected, documented, and distributed with the library.

1. Summary of paper presented at 1975 Winter Meeting of the American Nuclear Society, San Francisco, California, November 16–21, 1975.
2. Research sponsored by ERDA's Division of Controlled Thermonuclear Research.
3. Computer Sciences Division, UCC Nuclear Division.
4. University of Illinois.

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6. C. R. Weisbin and R. J. LaBauve, *Specification of a Generally Useful Multigroup Structure for Neutron Transport*, LA-5277-MS (1973).
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48. COMPARISON OF DOPPLER BROADENING METHODS[1,2]

D. E. Cullen[3] R. Q. Wright[4]

C. R. Weisbin J. E. White[4]

The SIGMA1, TEMPO, and PSI-CHI methods used to Doppler-broaden neutron cross sections in a wide variety of applications have been compared to define the range of applicability and computer costs associated with each method.

Each approximation introduced by the methods has been examined to obtain universally applicable curves that *quantitatively* illustrate where each approximation is valid.

The ENDF/B-IV library has been used to demonstrate the results obtained by each method when applied to light isotopes at CTR core temperatures and fissile isotopes (^{239}Pu and ^{235}U) at fission core temperatures. The results demonstrate that compared with the exact SIGMA1 method, the light-isotope thermal ($E = KT$) cross sections obtained by the other methods differ by up to 50%. Cross sections in the resolved resonance region of ^{239}Pu and ^{235}U differ by 10%. Comparison with experimental measurements is used to illustrate that these differences lead to significant, observable macroscopic results.

1. Abstract of paper presented at Conference on Nuclear Cross Sections and Technology, Washington, D.C., March 3–7, 1975; *Bull. Am. Phys. Soc., Ser. II* 20(2), 151 (1975); full paper published in *Nuclear Cross Sections and Technology, vol. I*, ed. by R. A. Schrack and C. D. Bowman, NBS Special Publication 425, p. 392 (1975).
2. ORNL participation sponsored by ERDA's Division of Reactor Research and Development.
3. Lawrence Livermore Laboratory.
4. Computer Sciences Division, UCC Nuclear Division.

49. RECOIL SPECTRA, DAMAGE ENERGY, AND DISPLACEMENT CROSS SECTIONS[1,2]

T. A. Gabriel J. D. Amburgey[3] N. M. Greene[3]

The AMPX code system,[4] which is used at ORNL to calculate from ENDF data-coupled neutron and gamma-ray multigroup cross sections for use in transport codes, such as the MORSE, ANISN, and DOT codes,[5-7] has been modified to obtain charged-particle recoil spectra in the multigroup format. The purpose of the investigation is threefold: First, these generated spectra can be used to analyze available neutron sources to determine how well these sources simulate fusion-reactor conditions. Second, these spectra can be coupled with existing damage and displacement models[8-10] to obtain effective damage energy and displacement cross sections. Third, recoil spectra for most structural materials, along with an analyzing program, will be placed on magnetic tape so that the data will be available to users in the shielding community.

So far, elastic, inelastic resolved, inelastic unresolved, and $(n,2n)$ reactions have been considered. Several improvements have been incorporated into the present analysis. One of these is the inclusion of anisotropic inelastic neutron collisions. In previous calculations, only isotropic inelastic collisions were considered. Also, AMPX will be updated to read the charged-particle data from ENDF as the information becomes available.

The ^{27}Al primary recoil spectra, produced by neutrons with energies between 13.5 and 14.9 MeV, for elastic, first-level inelastic, and fifteenth-level inelastic neutron collisions with ^{27}Al are shown in Fig. 49.1.

1. Summary of paper presented at American Nuclear Society Meeting, New Orleans, Louisiana, June 8–13, 1975; *Trans. Am. Nucl. Soc.* **21**, 532 (1975).
2. Research sponsored by ERDA's Division of Physical Research.
3. Computer Sciences Division, UCC Nuclear Division.

ORNL-DWG 75-1126

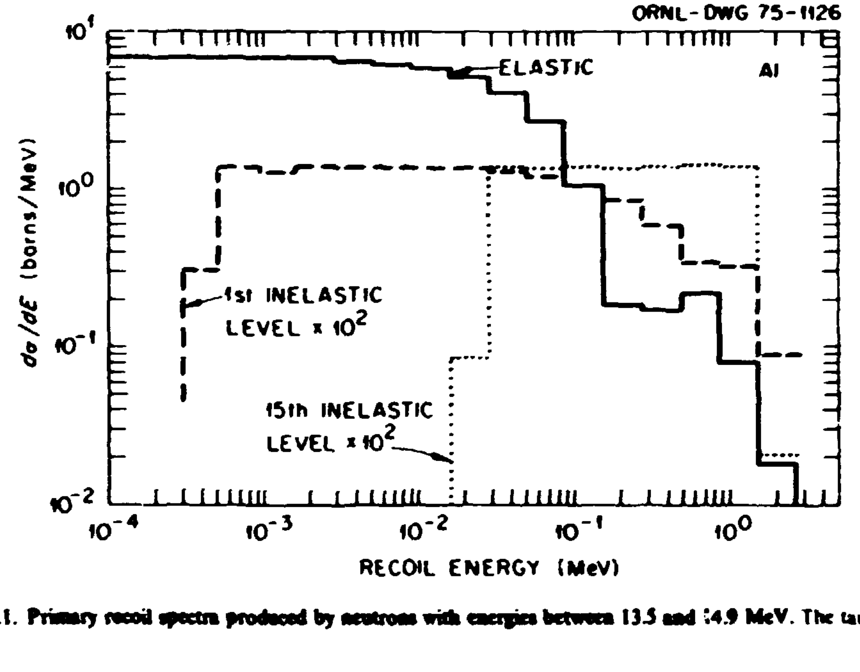

Fig. 49.1. Primary recoil spectra produced by neutrons with energies between 13.5 and 14.9 MeV. The target element is aluminum.

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6. W. W. Engle, Jr., *A Users Manual for ANISN, a One-Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering*, K-1693, Computing Technology Center, Union Carbide Corporation (1967).

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50. FICTITIOUS SCATTERING MULTIGROUP CROSS SECTIONS[1,2]

S. N. Cramer W. W. Engle, Jr.

The creation of a spatially independent total cross section for any system through which particle transport is to be calculated by Monte Carlo multigroup energy techniques can be realized by direct application of the fictitious scattering model in the structure of the multigroup cross sections. The constant total cross section in the system eliminates the need for tracking particles across material boundaries, making the mockup of a geometrically complex system economically practical. The present method represents an improvement over previous applications of the fictitious scattering model, which require additional computational steps in the solution of the transport equation.[3-7]

The total cross section of the system is modified by the addition of a fictitious nuclide scattering cross section, $\Sigma_F(\bar{r})$:

$$\Sigma_{T_g} = \Sigma_{a_g}(\bar{r}) + \Sigma_{s_g}(\bar{r}) + \Sigma_{F_g}(\bar{r}) . \tag{1}$$

Homogeneity of the system for particle transport can be created by choosing the spatial distribution of $\Sigma_{F_g}(\bar{r})$ such that Σ_{T_g} is equivalent for all mixtures in the system for a given energy group.

The physical integrity of the calculation is preserved by adjusting the scattering cross section of each mixture so that particle collisions with the fictitious nuclide result in no change in energy or direction. This condition can be illustrated by the addition of the fictitious cross section to both sides of the multigroup neutron transport equation:

$$\bar{\Omega}\cdot\nabla\phi_g(\bar{r},\bar{\Omega}) + \Sigma_{T_g}\phi_g(\bar{r},\bar{\Omega}) = S_g(\bar{r},\bar{\Omega}) + \sum_{g'}\int^{\bar{\Omega}'}\Big[\Sigma_{s_{g'\to g}}(\bar{r})f(\bar{\Omega}\cdot\bar{\Omega}')$$

$$+ \Sigma_{F_{g'\to g}}(\bar{r})\delta_{gg'}\delta(\bar{\Omega}\cdot\bar{\Omega}' - 1)\Big]\phi_{g'}(\bar{r},\bar{\Omega}')d\bar{\Omega}' , \tag{2}$$

where

$$\Sigma_{F_g} = \sum_{g'}\Sigma_{F_{g'\to g}}(\bar{r})\delta_{gg'} . \tag{3}$$

There are now two components in the scattering integral: the regular scattering function and a fictitious term that represents within-group scattering with no direction change. Using the Legendre polynomial expansion of the Dirac delta function,

$$\delta(\mu - 1) = \sum_{l=0}^{N}\frac{2l+1}{2}P_l(\mu) . \tag{4}$$

the P_N approximation of the modified scattering cross section is

$$\Sigma_{g\to g}(\bar{r},\mu)=\sum_{l=0}^{N}\frac{2l+1}{2}\left[\Sigma^{l}_{s_{g\to g'}}(\bar{r})+\Sigma_{F_{g\to g'}}(\bar{r})\delta_{gg'}\right]P_l(\mu)\,. \tag{5}$$

The bracketed quantity in Eq. (5) represents a new set of expansion coefficients which are different from the normal coefficients only for within-group scattering.

The procedure for creating the modified cross-section set is to first determine the magnitude of the fictitious cross section for each energy group in each mixture such that Σ_{T_i} is constant throughout the system. When the Σ_{F_i}'s have been determined, they are combined with the appropriate natural expansion coefficients as indicated in Eq. (5). Since the fictitious term is independent of the order of expansion, the same Σ_{F_i} is added to each $\Sigma_{s_{i\to i}}$.

Calculations have been made with the MORSE[9] Monte Carlo code, using multigroup cross-section sets produced by the methods described. Good comparison with standard methods has been obtained using an expansion order as low as P_1. By eliminating boundary crossings in the random walk and estimation processes in the code, a significant saving in computation time has been achieved for systems with many boundaries, not only over that for analog tracking, but also for the previous applications of fictitious scattering methods.

1. Summary of paper presented at the 1975 Winter Meeting of the American Nuclear Society, San Francisco, California, November 16–21, 1975.

2. Research sponsored by ERDA's Division of Reactor Research and Development.

3. F. Woodcock et al., *Proc. Conf. Application of Computing Methods to Reactor Problems*, ANL-7050 (1965).

4. N. R. Candelore and R. C. Gast, *RECAP-2: A Monte Carlo Program for Estimating Epithermal Capture Rates in Rod Arrays*, WARP-TM-427 (1964).

5. S. N. Cramer, *Monte Carlo Analysis of the Exact Geometric Mockup of ZPPR-III Assembly 48*, ORNL-TM-3596 (1971).

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7. S. N. Cramer, "Next Flight Estimation for the Fictitious Scattering Monte Carlo Method," *Trans. Am. Nucl. Soc.* **18**, 400 (1974).

8. S. N. Cramer, *Application of the Fictitious Scattering Model for Deep Penetration Monte Carlo Calculations*, ORNL-TM-4880 (to be published).

9. E. A. Straker et al., *The MORSE Code – A Multigroup Neutron and Gamma-Ray Monte Carlo Transport Code*, ORNL-4585 (1970).

51. ESTIMATED UNCERTAINTIES IN NUCLEAR DATA—AN APPROACH[1,2]

F. G. Perey

The need to communicate estimated uncertainties in evaluated nuclear data to be used in sensitivity studies has been recognized in the ENDF/B system. Starting with ENDF/B-IV, the data files contain formulated data describing the estimated covariances of the microscopic cross sections in such a form that they can be processed by computer codes to generate covariance matrices of quantities used in solving transport problems. The basic concepts behind the representation of such quantities will be described, and the work done so far in generating the information and its use in practical applications will be reviewed. Problem areas not yet addressed in ENDF/B-IV but under study will be discussed.

1. Abstract of invited paper presented at the Conference on Nuclear Cross Sections and Technology, Washington, D.C., March 3–7, 1975; *Bull. Am. Phys. Soc., Ser. II* **20**, 170 (1975); full paper published in *Nuclear Cross Sections and Technology*, vol. II, ed. by R. A. Schrack and C. D. Bowman, NBS Special Publication 425, p. 842 (1975).

2. Research sponsored by ERDA's Division of Reactor Research and Development.

52. UNCERTAINTIES AND CORRELATIONS IN EVALUATED DATA SETS INDUCED BY USE OF STANDARD CROSS SECTIONS[1,2]

R. W. Peelle

If reactor parameter uncertainties are to be obtained from sensitivity coefficients, evaluated uncertainty files must include the broadly correlated uncertainties in evaluated data sets that are induced by uncertainties in standard (n,α) cross sections (and standard instruments). This propagation problem has been analyzed using first-order error theory, assuming that the uncertainties are fully characterized for the standards and that the evaluator knows the energy-dependent weight given to each experiment in obtaining the evaluated cross sections. Three cases occur, depending on whether the standard is used at a single energy or as an energy-dependent absolute or relative "shape" standard. The analysis yields the uncertainties and correlations among evaluated cross sections, accounting for arbitrary interdependence among the standards themselves. The output uncertainty information need not refer explicitly to the input standards and can be processed into group variance-covariance matrix components in the same way as other uncertainty data. The properties of the solutions for the principal cases are demonstrated using the current ENDF/B uncertainty representations.

1. Abstract of paper presented at the Conference on Nuclear Cross Sections and Technology, Washington, D.C., March 3–7, 1975; *Bull. Am. Phys. Soc., Ser. II* **20**, 141 (1975); full paper published in *Nuclear Cross Sections and Technology, vol. I*, ed. by R. A. Schrack and C. D. Bowman, NBS Special Publication 425, p. 173 (1975).

2. Research sponsored by ERDA's Division of Reactor Research and Development.

53. TOTAL NEUTRON CROSS-SECTION CHECKS FOR IRON, CHROMIUM, NICKEL, CARBON, AND SODIUM[1,2]

R. E. Maerker F. J. Muckenthaler

More measurements have recently been performed in an uncollided geometry similar to that employed in an earlier series of "broomstick" experiments.[3] Neutron fluxes in the region 0.8 to 10 MeV were measured with an NE-213 spectrometer system[4] behind various materials placed in a collimated beam from the TSR-II at the Tower Shielding Facility. In addition, different spectrally dependent counting rates using 4-, 6-, and 10-in. Bonner balls[5] were utilized to extend the measured energy range down to 10 keV. Both types of measurements are integral in the sense that the total cross section at a given energy cannot be inferred because of poor resolution in the case of the NE-213 and spectral averaging over the entire energy range in the case of the Bonner balls. Nevertheless, these measurements serve as integral tests of the total cross section for various data sets.

Measurements were first performed behind 25 in. of carbon, since $\sigma_t(E)$ for carbon is relatively well known and can serve to check the experimental procedure. Comparison of the measured and calculated NE-213 transmitted fluxes, the latter using both versions III and IV of ENDF/B, is shown in Fig. 53.1. Although the comparisons are in good enough agreement to vindicate the experimental procedure, significant discrepancies in version IV are seen to exist in the vicinity of 2.9 and 8.3 MeV, where the measured average cross sections are respectively about 4 and 8% lower.

In general, for all the materials investigated, the calculated fluxes and Bonner ball counting rates using version IV compare better with the measurements than version III, but sizable errors

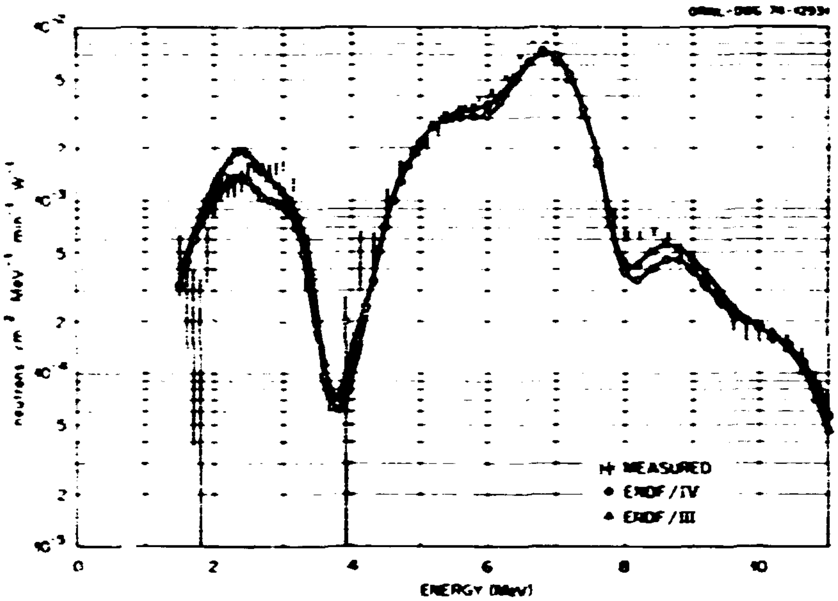

Fig. 53.1. Comparison of measured and calculated NE-213 spectra through 25 in. of carbon.

Table 53.1. Probable discrepancies in ENDF/B version IV total cross sections

Approximate energy range (MeV)	$[(\sigma_{T,IV} - \sigma_{T,meas})/\sigma_{T,meas}] \times 100^a$ Carbon (5.14)[b]	Iron (4.08)	Nickel (2.77)	Chromium (2.00)	Sodium (1.54)	304 stainless steel (2.63)
9.5–10	0			0	6	0
9–9.5	0			0	6	0
8.5–9	0			0	6	0
8–8.5	+8			0	6	0
7.5–8	0		0	0	6	0
7–7.5	0		0	0	6	0
6.5–7	+3		0	0	6	0
6–6.5	+3		0	0	6	0
5.5–6	0		0	0	6	0
5–5.5	3		0	0	6	0
4.5–5	0	+4	0	+5	6	+2
4–4.5	+4	+4	+4	+5	6	+2
3.5–4		0	+4	+5	6	0
3–3.5		0	+4	+10	10	0
2.5–3	+4	+8	+9	+13	4	+2
2–2.5	0	+3	+14	+13	4	+2
1.5–2		+3	+14	+13	4	+2
1–1.5		+7	+20	+17		+6
0.01–1		0	+13	+17	0	+4

[a]That is, a positive entry means Version IV is higher than the measurement by the percentage indicated. A zero indicates that no significant disagreement was found (i.e., ≤2%), and the absence of an entry indicates fluxes were too low to be properly interpreted.

[b]Numbers given in parentheses on this line are in units of atoms/barn.

even in version IV are indicated for chromium and nickel. Table 53.1 presents a summary of probable discrepancies in the average total cross sections for version IV as deduced from these measurements.

1. Summary of paper presented at the American Nuclear Society Meeting, New Orleans, Louisiana, June 8-13, 1975; *Trans. Am. Nucl. Soc.* 21, 530 (1975).

2. Research sponsored by ERDA's Division of Reactor Research and Development.

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54. TOTAL-CROSS-SECTION EXPERIMENTAL CHECK[1,2]

C. E. Clifford F. J. Muckenthaler P. N. Stevens[3]

This report describes a series of experiments performed at the Tower Shielding Facility to check the accuracy of the total cross sections for iron, nickel, chromium, and the elements in stainless steel. Reunfolded results of NE-213 spectral measurements made in November 1973 are presented along with the results of the May 1974 NE-213 and Bonner ball measurements. The May 1974 measurements were made because of background difficulties associated with the November 1973 results. It was found that the May 1974 experimental configuration essentially eliminated the background problems experienced with earlier configurations.

1. Abstract of ORNL-TM-4921 (May 1975).
2. Research sponsored by ERDA's Division of Reactor Research and Development.
3. The University of Tennessee.

55. ONE-DIMENSIONAL FAST-NEUTRON TRANSPORT BENCHMARK CALCULATIONS[1,2]

J. Ching[3] E. M. Oblow

A series of state-of-the-art fast-neutron transport calculations were made for several simple shield materials of interest to the Defense Nuclear Agency. The calculations should serve as benchmarks for testing newly evaluated neutron cross-section data and multigroup libraries. Results are presented for one-dimensional spherical systems of iron, air, and concrete (SiO_2). The results include detailed flux spectra, broad-group fluxes vs distance, and convergence tests for representations of the P_l scattering kernel and for space, angle, and energy grids. The methods employed in converging the solutions differ from previous benchmark calculations in that sensitivity and point-energy techniques were used to determine the energy grid for the final results.

1. Abstract of ORNL-TM-5020 (September 1975).
2. Research sponsored by the Defense Nuclear Agency.
3. Computer Sciences Division, UCC Nuclear Division.

56. BENCHMARK EXPERIMENTS FOR NUCLEAR DATA[1,2]

E. M. Bohn[3] R. J. LaBauve[4] R. E. Maerker
B. A. Magurno[5] F. J. McCrosson[6] R. E. Schenter[7]

Five general areas of application for nuclear cross sections have been identified for benchmark testing by the Cross Section Evaluation Working Group: thermal and fast reactors, shielding, dosimetry, and fission-product properties. For both thermal and fast reactors, benchmark experiments consist in measurements of integral reaction rates, reactivity coefficients, and criticality in well-defined, simple-geometry, uniform-composition assemblies. Thermal-reactor benchmark experiments are carried out in spheres of heavy-metal nitrate solutions and in H_2O- and D_2O-moderated lattices of uranium. These experiments serve as tests of moderator properties and of uranium and plutonium capture and fission cross sections. Carefully specified experiments in large fast zero-power critical assemblies (e.g., ZPR-6-7, ZPPR-2) and in small uranium and plutonium metal spheres (e.g., Godiva, Jezebel) provide integral tests primarily for uranium and plutonium capture and fission cross sections and fission neutron properties in the energy range 0.1 keV to 10 MeV. Shielding benchmark experiments are characterized by a well-defined neutron source, detectors having known absolute response functions and a simple source-detector geometry. An example of a shielding benchmark is the ORNL "broomstick" series of experiments in which uncollided neutrons emerging from thick samples placed in a collimated neutron beam were observed. These measurements serve as tests of neutron total cross sections in the range 1 to 11 MeV. Dosimetry benchmark experiments consist in integral reaction rate measurements in known neutron spectra. These experiments include measurements sensitive to low-energy neutrons [e.g., ^{10}B(n,α), ^{197}Au(n,γ) in thermal spectra] and to relatively high-energy neutrons [e.g., ^{238}U(n,f), ^{239}Pu(n,f) in fission spectra]. Benchmark experiments for fission products include integral measurements of capture rates and reactivity effects for individual fission-product isotopes in typical reactor spectra (e.g., the CFRMF and STEK facilities) and measurements of fission-product yields and decay properties, especially those decay modes relating to decay heat production. The results of all these benchmark experiments have been compared with calculations using the latest version of the Evaluated Neutron Data File, ENDF/B-IV. Better agreement between calculations and measurements was found with ENDF/B-IV as compared with ENDF/B-III. Some of the measurements are predicted well, but there are important disagreements, most notably the overprediction of ^{238}U capture in both thermal and fast reactors.

1. Abstract of paper presented at the Conference on Nuclear Cross Sections and Technology, Washington, D.C., March 3–7, 1975; *Bull. Am. Phys. Soc., Ser. II* **20**(2), 170 (1975); full paper published in *Nuclear Cross Sections and Technology*, vol. *II*, ed. by R. A. Schrack and C. D. Bowman, NBS Special Publication 425, p. 834 (1975).
2. Division participation sponsored by ERDA's Division of Reactor Research and Development.
3. Argonne National Laboratory.
4. Los Alamos Scientific Laboratory.
5. Brookhaven National Laboratory.
6. Savannah River Laboratory.
7. Hanford Engineering Development Laboratory.

57. SDT11. THE ORNL BENCHMARK EXPERIMENT FOR NEUTRON TRANSPORT THROUGH IRON AND STAINLESS STEEL, PART I[1,2]

R. E. Maerker

The first part of an experiment concerning deep neutron penetration in iron and stainless steel is described, and experimental results in a format for the Cross Section Evaluation Working Group shielding integral data testing are presented. These results provide a basis for verification of the accuracy of iron and stainless steel cross sections used in transport calculations. The experiment was performed at the Tower Shielding Facility of ORNL and included measurements of both the neutron fluence and the neutron spectra behind slabs of iron up to 3 ft thick and of stainless steel up to 1 ft thick.

1. Abstract of ORNL-TM-4222 (ENDF-188) (September 1974).
2. Research sponsored by ERDA's Division of Reactor Research and Development.

58. SHIELDING BENCHMARK PROBLEMS[1,2]

Benchmark Problems Group of the ANS-6
Standards Committee, American Nuclear Society

G. L. Simmons, Editor

The Benchmark Problems Group of the American Nuclear Society Shielding Standards Subcommittee (ANS-6) has selected several "benchmark" problems for testing computational methods of radiation transport. Problems are described and solution data are presented. It is anticipated that publication of these problems will serve to focus attention so that careful work will produce solutions that are representative of the state of the art. The problems will also serve to specify standard configurations so that meaningful comparisons can be made between different calculational methods and between experimental and calculational results.

Supplement 1 provided an additional solution to a problem issued previously (neutron spectrum in graphite) and presented a new problem (nucleon-meson cascade in iron).

Supplement 2 provides a calculational problem (polyethylene slab) that can be treated in one or two dimensions, and an experimental problem (deep neutron penetration in sodium) with calculational solutions.

1. Abstract of ORNL-RSIC-25 (ANS-SD-9), Supplement 2 (July 1974)
2. Work partially supported by the Defense Nuclear Agency.

59. SENSITIVITY THEORY FROM A DIFFERENTIAL VIEWPOINT[1,2]

E. M. Oblow

An alternate derivation of sensitivity functions in terms of differential calculus is presented to overcome some common misconceptions about their method of calculation and accuracy. No reference is made in this approach to perturbations or variations in defining sensitivity functions so that they can be seen to be "exact" differential quantities. The differential method also should allow sensitivity methods to be applied more easily to simple model problems in shielding and core physics, since they can now be analyzed in terms of simple derivatives.

1 Abstract of a paper accepted for publication as a Technical Note in *Nuclear Science and Engineering*

2 Research sponsored by ERDA's Division of Reactor Research and Development

60. SENSITIVITY ANALYSIS DEVELOPMENT AND APPLICATIONS PROGRAM AT ORNL[1,2]

E. M. Oblow

The cross-section sensitivity analysis program at ORNL is reviewed with emphasis on present computer code capabilities and past successful applications in the radiation shielding area. The FORSS sensitivity code system is discussed in regard to objectives, methodology, and code specifications. Examples of past shielding applications of FORSS emphasize the success of fine energy grid sensitivity studies and group structure selection, the use of evaluated error files in problem uncertainty estimation, two-dimensional shield sensitivity analysis and integral experiment design for fast reactors, data studies for the LMFBR program related to sodium and iron evaluations, and iron data problems in CTR shield design. Conclusions are drawn about the adequacy of present ENDF B data files for sodium and iron and the general applicability of sensitivity studies in future shield design and analysis.

1 Abstract of paper presented at the Specialists' Meeting on Sensitivity Studies and Shielding Benchmarks, jointly sponsored by the Nuclear Energy Agency of the Organization for Economic Cooperation and Development and the International Atomic Energy Agency, Paris, France, October 7–10, 1975; also an abstract of ORNL-TM-5176 (to be published)

2 Research sponsored jointly by ERDA's Division of Reactor Research and Development and the Defense Nuclear Agency

61. THE FORSS SENSITIVITY ANALYSIS CODE SYSTEM[1,2]

C. R. Weisbin E. M. Oblow
F. R. Mynatt G. F. Flanagan

The measurement, evaluation, and processing of radiation interaction data together with experimental and analytic data testing efforts have been major activities of the reactor physics community for several decades. Implicit in such a data acquisition program is the goal of generating "good" data for the analysis and design of reactors and radiation shields. While a major concern in the past was simply the acquisition of data in areas where none existed, more recent efforts have focused on questions about the quality or "goodness" of the data.

The ORNL sensitivity code system, FORSS, is being written with the following four major objectives:

1. To determine the nuclear data-related uncertainties for specified performance parameters in radiation shielding and reactor design problems from input basic cross sections and their uncertainties and integral experiments and their uncertainties.
2. To determine the impact of alternative data sets or differences in various elements of a data field on shield and reactor performance.

3. To provide quantitative guidance to the data acquisition, evaluation, and processing code development programs in the form of estimated accuracy requirements for nuclear data to meet design constraints and error margins.
4. To provide qualitative and quantitative understanding of the relative importance of the basic physical processes involved in any analytical or experimental problem so that conclusions reached in one area can be extrapolated into many others (e.g., the extrapolation of results in critical experiments to commercial reactor design).

Solution of the sensitivity analysis problem is based upon classical linear perturbation theory and generalizations to it developed in recent years.[3-8] Assuming the validity of first-order perturbation theory, FORSS is used to generate sensitivity profiles (P_Σ) to estimate the effect on some shield or reactor performance parameter (R) of finite changes in any or all members of the

$$\frac{\delta R}{R} \cong P_\Sigma \cdot \frac{\delta \Sigma}{\Sigma} \, . \tag{1}$$

The expectation value of the variance in the performance parameter (V_R) is computed according to

$$V_R = P_\Sigma \cdot B \cdot \widetilde{P}_\Sigma \, , \tag{2}$$

where B is the covariance matrix representing the variance and correlations in the basic nuclear data, and P_Σ . $\widetilde{P}_\Sigma$ are the sensitivity profile and its transpose. Integral experimental information is used as a conditional probability distribution leading to improved estimates for performance parameter uncertainties consistent with both differential and integral data.[9]

In an attempt to provide quantitative guidance for the generation of cross-section request lists, the cost function $C(V_{\Sigma_1}, V_{\Sigma_2}, \ldots, V_{\Sigma_3})$ associated with reducing the uncertainties in any basic data (V_{Σ_i}) below current levels is formulated, and it is this cost which is minimized in identifying the "best" combination of uncertainties to be achieved by subsequent measurement.[10] To employ available computer methods, the problem has been posed in terms of a variational functional which is to be minimized. Thus we attempt to:

$$\text{minimize } F = C(V_{\Sigma_1}, V_{\Sigma_2}, \ldots\ldots, V_{\Sigma_3}) - \sum_{i=1}^{I} k_i [(P_\Sigma B \widetilde{P}_\Sigma) - a_i] \, , \tag{3}$$

where the k_i's are Lagrange multipliers, the V_{Σ_j}'s are the unknown data uncertainties, the a_i's are the performance parameter constraints, and B, the covariance matrix, is assumed to be a known function of the V_{Σ_j}'s. The solution to Eq. (3) requires the minimization of a nonlinear functional subject to quadratic constraints which is handled by linearization and iterative procedures.

We have applied the FORSS system to several radiation shielding problems including neutron-gamma radiation transport in air.[3,11] The most recent advance has been in the definition of an "uncertainty profile[12] presented in Fig. 61.1. Illustrated is the contribution to the integrated flux at 2000 m of uncertainties in group cross sections of nitrogen and oxygen and the effect of correlations in energy and reaction type. Energy correlations in the (n,α) covariance description are

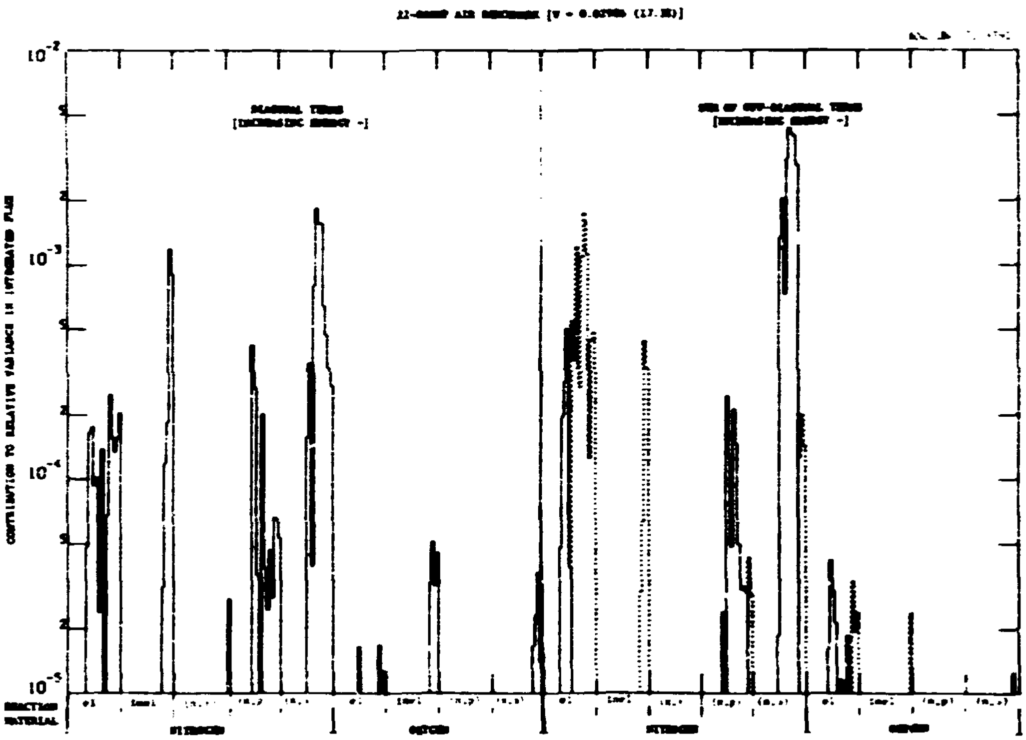

Fig. 61.1. Uncertainty profile for air transport benchmark problem.

seen to make as significant a contribution as does the variance in the group containing the nitrogen window.

1. Abstract of a paper presented at the 1975 Winter Meeting of the American Nuclear Society, San Francisco, California, November 16–21, 1975.

2. Research sponsored by ERDA's Division of Reactor Research and Development.

3. D. E. Bartine, E. M. Oblow, and F. R. Mynatt, "Radiation Transport Cross-Section Sensitivity Analysis – A General Approach Illustrated for a Thermonuclear Source in Air," *Nucl. Sci. Eng.* **55**, 147–67 (1974).

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9. A. Gandini, *Nuclear Data and Integral Measurements Correlation for Fast Reactors. Part I: Statistical Formulation*, CNEN-RT FI(73)5 (February 1973).

10. Yu. G. Bobkov, L. T. Pyatnitskoya, and L. N. Usachev, *Planning of Neutron Data Experiments and Evaluations for Reactors*, INDC(CCP)-46 L (December 1974).

11. C. R. Weisbin, E. M. Oblow, J. Ching, J. E. White, R. Q. Wright, and J. Drischler, *Cross Section and Method Uncertainties: The Application of Sensitivity Analysis to Study Their Relationship in Calculational Benchmark Problems*, ORNL-TM-[illegible]47 (ENDF-218) (to be published); see, also, paper 63, this report.

12. J. D. Burns and C. R. Weisbin, *DISECT, A Generalized Uncertainty Analysis Program*, ORNL-TM-5002 (to be published).

62. APPLICATION OF THE FORSS SENSITIVITY CODE SYSTEM TO FAST-REACTOR ANALYSIS[1,2]

C. R. Weisbin E. M. Oblow F. R. Mynatt

The FORSS Sensitivity Analysis Code System is described in terms of its objectives and present capabilities. An example is made of a problem specified by the Processing Methods Testing Subcommittee of the Code Evaluation Working Group, that is, the determination of integral parameters, sensitivities to cross-section data, methods and data uncertainties, and required cross-section accuracies for an infinite medium of ZPR 6 7 core composition.

1. Abstract of talk presented at Meeting of the Cross-Section Evaluation Working Group, Brookhaven National Laboratory, October 22, 1975.

2. Research sponsored by ERDA's Division of Reactor Research and Development.

63. CROSS-SECTION AND METHOD UNCERTAINTIES: THE APPLICATION OF SENSITIVITY ANALYSIS TO STUDY THEIR RELATIONSHIP IN CALCULATIONAL BENCHMARK PROBLEMS[1,2]

C. R. Weisbin E. M. Oblow J. Ching[3]
J. E. White[4] R. Q. Wright[4] J. Drischler

The complete ORNL cross-section sensitivity and uncertainty analysis system (FORSS)[5] has been applied to compute the overall uncertainty and the contribution of individual reaction components to the total uncertainty in several calculational benchmark transport problems. Only relatively simplified geometries were considered, so that uncertainties due to spatial modeling were considerably reduced. Evaluated uncertainties and their correlations in energy and reaction type taken from ENDF/B-IV where applicable (e.g., N and O for an air-transport case studied) were processed into the multigroup formalism[6] and folded with sensitivity profiles[7] to estimate the effects of uncertainties in nuclear data. The correlations between energy groups, reaction types, materials, etc., led to a large covariance matrix (e.g., ~1.5 million entries for the 267-group air problem) which required the development of elaborate data handling schemes. Differences due to divergent cross-section processing techniques[8] were treated, within linear perturbation theory, by considering the altered multigroup sets as perturbations. The point-energy discrete-ordinates sensitivity technique[9] was used to ensure proper convergence of the calculations with respect to energy grid selection. A 267-point energy grid was required in the air problem to assure 5% convergence in the 2000-m boundary fluence. The methods considered were Monte Carlo and moments (where applicable), as well as discrete ordinates, to assess the effects of using different transport methods.

1. Abstract of invited paper presented at Conference on Nuclear Cross Sections and Technology, Washington, D.C., March 3–7, 1975; *Bull. Am. Phys. Soc., Ser. II* **20**(2), 169 (1975); and abstract of ORNL-TM-4847 (ENDF-218) (August 1975); full paper published in *Nuclear Cross Sections and Technology*, vol. II, ed. by R. A. Schrack and C. D. Bowman, NBS Special Publication 425, p. 825 (1975).

2. Research jointly sponsored by Defense Nuclear Agency and ERDA's Division of Reactor Research and Development.

3. University of Tennessee; now with Energy, Inc., Idaho Falls, Idaho.

4. Computer Sciences Division, UCC Nuclear Division.

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6. F. G. Perey, G. de Saussure, and R. B. Perez, "Estimated Data Covariance Files for Evaluated Cross Sections Examples for ^{235}U and ^{238}U," *Advanced Reactors: Physics, Design, and Economics*, Proceedings of the International Conference held at Atlanta, Georgia, September 8–11, 1974, p. 578, Pergamon, New York, 1975.

7. D. E. Bartine, E. M. Oblow, and F. R. Mynatt, "Radiation Transport Cross-Section Sensitivity Analysis - A General Approach Illustrated for a Thermonuclear Source in Air," *Nucl. Sci. Eng.* 55, 147–67 (1974).

8. C. R. Weisbin, N. M. Greene, H. Henryson II, R. J. LaBauve, C. Durston, D. E. Cullen, and R. B. Kidman, "Fast-Reactor Cross-Section Processing Codes– Is There a Dollars Worth of Difference Between Them?" *Advanced Reactors: Physics, Design, and Economics*, op. cit., p. 468.

9. E. M. Oblow, J. Ching, and J. Drischler, "Selection of Group Energy Boundaries Using Sensitivity Theory," *Trans. Am. Nucl. Soc.* 17, 545 (1973).

64. PERTURBATION THEORY AND SENSITIVITY ANALYSIS FOR TWO-DIMENSIONAL SHIELDING CALCULATIONS[1,2]

R. L. Childs[3]   D. E. Bartine   W. W. Engle, Jr.

The purpose of this work is to describe the development and implementation of perturbation theory methods for two-dimensional shielding problems. Cross-section sensitivity results are obtained by calculating the effect of varying each individual cross-section parameter.

Changes in a response, R, can be calculated using the perturbation theory relationship[4]

$$\Delta R \approx \iiint \phi^*(\bar{r},E,\bar{\Omega}) \left[\iint \delta\Sigma_s(\bar{r},E' \to E,\bar{\Omega}' \to \bar{\Omega}) \times \phi(\bar{r},E',\bar{\Omega}')\,dE'\,d\bar{\Omega}' - \delta\Sigma_t(\bar{r},E)\phi(\bar{r},E,\bar{\Omega}) \right] dE\,d\bar{\Omega}\,d\bar{r} \ , \quad (1)$$

where the perturbation is represented by the changes in the total and scattering cross sections, and ϕ and ϕ^* are the forward and adjoint fluxes. This paper will deal with the evaluation of the above equation with forward and adjoint fluxes calculated using the discrete-ordinates computer program DOT.[5] The spatial and energy integrations are easily evaluated by replacing the integrals with summations over energy groups and a space mesh. The angular integrations are more complicated and will be considered in more detail.

The scattering term in Eq. (1) requires the evaluation of an integral of the form

$$\int_{4\pi} \phi^*(\Omega) \int_{4\pi} \phi(\Omega')\sigma(\Omega \cdot \Omega')\,d\Omega'\,d\Omega \ .$$

The cross section may be expanded in a Legendre series as

$$\sigma(\bar{\Omega} \cdot \Omega') = \frac{1}{4\pi} \sum_{l=0}^{\infty} \sigma_l P_l(\Omega \cdot \Omega') \ . \quad (2)$$

where $\sigma_l = (2l + 1) \int_{4\pi} \sigma(\Omega \cdot \Omega') P_l(\Omega \cdot \Omega')\,d\Omega$. Similarly, the angular fluxes may be expanded in a spherical harmonic series as

$$\phi(\Omega) = \sum_{l=0}^{\infty} \sum_{m=0}^{l} \frac{2l+1}{4\pi} f_{l,m} A_l^m(\Omega) \ . \quad (3)$$

$$\phi^*(\bar{\Omega}) = \sum_{l=0}^{\infty} \sum_{m=0}^{l} (-1)^l (2l+1) j^*_{l,m} A_l^m(\bar{\Omega}) \ . \tag{4}$$

where

$$j_{l,m} = \int_{4\pi} \phi(\bar{\Omega}) A_l^m(\bar{\Omega}) d\bar{\Omega} \ ,$$

$$j^*_{l,m} = \frac{1}{4\pi} \int_{4\pi} \phi^*(-\bar{\Omega}) A_l^m(\bar{\Omega}) d\bar{\Omega} \ ,$$

$$A_l^m(\bar{\Omega}) = \left(\frac{2}{1+\delta_{0,m}} \frac{(l-m)!}{(l+m)!} \right)^{1/2} P_l^m(\eta) \cos m\omega \ ,$$

$$\int_{4\pi} d\bar{\Omega} = \int_0^{2\pi} \int_{-1}^{1} d\eta \, d\omega \ ,$$

$$A_l^m(-\bar{\Omega}) = (-1)^l A_l^m(\bar{\Omega}) \ .$$

The form of the above equations was chosen to conform with the equations programmed in DOT. The $(-1)^l$ and $-\bar{\Omega}$ factors arise because the adjoint problem is solved for $\phi^*(-\bar{\Omega})$.

The addition theorem for legendre polynomials is (ignoring terms that vanish under integration)

$$P_l(\bar{\Omega}\cdot\bar{\Omega}') = \sum_{m=0}^{l} A_l^m(\bar{\Omega}) A_l^m(\bar{\Omega}') \ . \tag{5}$$

Substitution of Eqs. (2)–(5) into the desired integral and use of the orthogonality relationship

$$\int_{4\pi} A_l^m(\bar{\Omega}) A_{l'}^{m'}(\bar{\Omega}) d\bar{\Omega} = \frac{4\pi}{2l+1} \delta_{l,l'} \delta_{m,m'} \ . \tag{6}$$

yields the final result:

$$\int_{4\pi} \phi^*(\bar{\Omega}) \int_{4\pi} \phi(\bar{\Omega}') \sigma(\bar{\Omega}\cdot\bar{\Omega}') d\bar{\Omega}' \, d\bar{\Omega} = \sum_{l=0}^{\infty} \sum_{m=0}^{l} (-1)^l \sigma_l j_{l,m} j^*_{l,m} \tag{7}$$

The total cross-section term in Eq. (1) requires the evaluation of an integral of the form

$$\int_{4\pi} \phi(\Omega) \phi^*(\bar{\Omega}) d\Omega \ .$$

This integral can be evaluated using the angular fluxes from the discrete-ordinates calculation as has been done in one-dimensional calculations.[6,7] However, because of the tremendous quantity of numbers to be stored in two-dimensional problems (70 million for the example which follows), it is highly desirable to avoid storing the angular flux if possible. A way to avoid the angular flux may be found by substituting Eqs. (3) and (4) into the desired integral and using Eq. (6) to obtain

$$\int_{4\pi} \phi(\vec{\Omega})\phi^{*}(\vec{\Omega})d\vec{\Omega} = \sum_{l=0}^{\infty} \sum_{m=0}^{l} (-1)^{l}(2l+1)\phi_{l,m}\phi^{*}_{l,m} \; . \tag{8}$$

Thus, Eq. (1) can be evaluated using only the flux moments $\phi_{l,m}$ and $\phi^{*}_{l,m}$ provided the series in Eqs. (7) and (8) can be truncated after a few terms without serious error. Present experience indicates that a third-order expansion (through $l = 3$) is sufficient for many problems.

The approach outlined above has been implemented using the computer programs VIP[8] and SWANLAKE.[9] The method was tested for a deep-penetration sodium problem and for an infinite-air calculation (including a first-collision source term) by comparing with one-dimensional SWANLAKE calculations. Agreement was satisfactory in both cases.

The first realistic two-dimensional sensitivity analysis performed in this study was for the calculation of neutron and gamma-ray doses at the maintenance deck of the Fast Test Reactor.[10] The DOT calculations were performed with a 21-18 coupled neutron-gamma-ray cross-section set, a 166-angle quadrature set suitable for streaming in annular gaps, and P_3 scattering.

Results obtained from the sensitivity analysis include the sensitivity of neutron and gamma-ray dose to changes in cross sections as a function of position, energy, and element. For this problem, the important gamma-ray sources are iron (n,γ) reactions (primarily in the stainless steel head), which are responsible for 55% of the gamma-ray dose, and boron (n,γ) reactions (in a borated polyethylene layer under the maintenance deck), which contribute 28% of the gamma-ray dose. A more detailed study indicates that 31% of the gamma-ray dose is due to gamma-ray production in the 7- to 8-MeV group, which contains strong iron (n,γ) production lines at 7.63 and 7.65 MeV. The gamma-ray dose due to boron production is the result of the (n,γ) 0.48-MeV line.

This work demonstrates the feasibility of performing sensitivity studies and perturbation calculations for two-dimensional shielding problems. The FTR example shows the usefulness of calculations of this type for realistic shielding applications.

1. Summary of paper presented at American Nuclear Society Meeting, New Orleans, Louisiana, June 8–13, 1975; *Trans. Am. Nucl. Soc.* 21, 543 (1975).

2. Research sponsored by ERDA's Division of Reactor Research and Development.

3. Computer Sciences Division, UCC Nuclear Division.

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8. R. L. Childs, *VIP, A Computer Program Using Two-Dimensional Discrete Ordinates Transport Calculations for Cross-Section Sensitivity Analysis*, UCCND-CSD-1 (to be published).

9. D. E. Bartine, F. R. Mynatt, and E. M. Oblow, *SWANLAKE, A Computer Code Utilizing ANISN Transport Calculations for Sensitivity Analysis*, ORNL-TM-3809 (1973).

10. F. R. Mynatt et al., *Fast Reactor Analytical Shielding Progress Report for May and June 1973*, ORNL-TM-4401 (1974).

65. DEVELOPMENT OF RADIATION SHIELDING STANDARDS IN THE AMERICAN NUCLEAR SOCIETY[1,2]

D. K. Trubey[3]

The Standards Committee of the American Nuclear Society (ANS), as a standards-writing organization-member of the American National Standards Institute (ANSI), has a subcommittee, ANS-6, whose charge is to establish standards in connection with radiation shields, to provide shielding information to other standards writing groups, and to prepare recommended sets of shielding data and test problems. This subcommittee, now composed of eight working groups, was established in 1964.[4] The goals and accomplishments of the working groups to date are briefly described below.

ANS-6.1: Shielding cross-section standards. Because of the changing state of knowledge of neutron cross sections, no standard is possible with anything near the time constant of the usual standard. At present, it is better to think of recommended cross-section sets as *reference* sets rather than standards. Since the Evaluated Nuclear Data File (ENDF) represents current reference data, no action in this area seems to be required. The group did recommend in 1970 that the photon interaction cross-section set evaluated by the Lawrence Livermore Laboratory be accepted as reference data and be distributed with the ENDF neutron data. These data are now included in the ENDF B-IV library.

Current projects include standard flux-to-dose-rate conversion factors and reference-coupled neutron-gamma-ray multigroup cross sections for the elements normally found in concrete.

ANS-6.2: Benchmark problems. The primary objective of the benchmark problems effort is to compile in convenient form a limited number of well-documented problems in radiation transport which will be useful in testing computational methods used in shielding.

Four problem solutions were published in 1969 in looseleaf form[5] by the Radiation Shielding Information Center. Revisions were issued in 1970 and 1974. The work slowed for a period, but now additional problems are being developed, including several for typical reactor configurations.

ANS-6.3: Shield performance evaluation. The initial goal of this group was attained in 1972 with the publication of ANSI N18.9-1972.[6] The current goal of the committee is to completely rewrite it to make it more applicable to current practices in the nuclear power industry.

ANS-6.4: Shield materials. Initial efforts involving materials per se, for example, lead as a shielding material, have failed. The current work has produced the draft of a standard on the "Analysis and Design of Concrete Radiation Shielding for Nuclear Plants." The draft was being revised in mid-1975.

ANS-6.5: Shielding nomenclature. The goal is the preparation of a list of shielding terms and definitions. A draft glossary was expected to be available for review in mid-1975.

ANS-6.6: Calculation and measurement of direct and scattered radiation from nuclear power plants. Organized in late 1973, the group expects to define calculational requirements and measurement techniques for estimates of exposures near nuclear power plants due to direct and scattered radiation from contained sources on site. Nitrogen-16 gamma rays are a prime consideration.

ANS-6.7: Radiation zoning for design of nuclear power plants. This group was organized in early 1974 to standardize radiation zone designations and related design considerations.

ANS-6.8: Location and range of detection of area and process fluid radiation monitoring systems for nuclear steam generating plants. Organized in early 1975, the group plans to establish requirements for personnel and environmental monitoring of radiation with regard to personnel areas, process lines, and effluent discharges.

1. Summary of paper presented at 1975 Winter Meeting of the American Nuclear Society, San Francisco, California, November 16–21, 1975; also summary of ORNL-RSIC-39 (October 1975).

2. ORNL effort jointly sponsored by ERDA's Division of Reactor Research and Development, the Defense Nuclear Agency, and the Electric Power Research Institute.

3. Chairman of Subcommittee ANS-6, Shielding.

4. H. C. Claiborne, D. S. Dudziak, and N. M. Schaeffer, *Activities of the ANS-6 Subcommittee on Radiation Shielding Standards*, ORNL-TM-3271 (1970).

5. A. E. Profio, Editor, *Shielding Benchmark Problems*, ORNL-RSIC-25 (ANS-SD-9) (Suppl. 1, 1970; Suppl. 2, 1974).

6. *Program for Testing Biological Shielding in Nuclear Power Plants*, ANSI N18.9-1972 (1972), American Nuclear Society, Hinsdale, Ill.

66. RADIATION SHIELDING INFORMATION CENTER DATA ACTIVITIES[1,2]

R. W. Roussin Betty F. Maskewitz D. K. Trubey

Activities developed at the RSIC play an important role in the utilization of nuclear cross sections in various radiation transport applications and help improve the general utility of the national ENDF/B effort. Since 1968 the Center's activities have included collecting, packaging, analyzing, and distributing computer-readable data relevant to fission and fusion reactor, weapons, and space radiation transport. A description of this Data Library Collection will be given, outlining the various processed data sets included therein and demonstrating how data generated for one application can sometimes be utilized in other areas. The role that the RSIC has played in maintaining and distributing the Defense Nuclear Agency (DNA) Evaluated Cross Section Library will be discussed, with emphasis on how that effort has served to augment, support, and improve the ENDF/B system. Also discussed is the distribution of processed data sets which are generated for specific applications for DNA contractors. Similar efforts (generating and distributing processed data sets and maintaining and distributing an evaluated library) planned for the CTR neutronics community will be described.

1. Abstract of paper presented at Conference on Nuclear Cross Sections and Technology, Washington, D.C., March 3–7, 1975; full paper published in *Nuclear Cross Sections and Technology*, vol. 1, ed. by R. A. Schrack and C. D. Bowman, NBS Special Publication 425, p. 363 (1975).

2. Research jointly sponsored by ERDA's Division of Reactor Research and Development and the Defense Nuclear Agency.

67. PROCEEDINGS OF FIFTH SYMPOSIUM ON SHARING OF COMPUTER PROGRAMS AND TECHNOLOGY IN NUCLEAR MEDICINE[1]

Compiled by

W. J. McClain[2] Betty F. Maskewitz

"Computer Applications in Nuclear Medicine" was the topic of the Fifth Symposium on Sharing of Computer Programs and Technology in Nuclear Medicine held on January 15–16, 1975, in Salt Lake City, Utah. This Fifth Symposium was cosponsored by the Society of Nuclear Medicine, the Oak Ridge National Laboratory, and the Bureau of Radiological Health. Participating in the meeting were 104 attendees representing 81 nuclear medicine facilities and laboratories throughout the United States and four foreign countries (Canada, France, Germany, and Israel). The symposium papers are being published as the result of a continuing policy of the sponsors to make available information which is likely to advance the state of the art in an important technical field.

1. From the preface of CONF-750124 (1975).
2. Computer Sciences Division, UCC Nuclear Division.

68. BIOMEDICAL COMPUTING TECHNOLOGY INFORMATION CENTER (BCTIC), A FOCUS FOR THE SHARING OF COMPUTER PROGRAMS AND TECHNOLOGY IN BIOMEDICINE[1,2]

Betty F. Maskewitz W. J. McClain[3] R. L. Henne[3]

The BCTIC has been established at the Oak Ridge National Laboratory by the U. S. Energy Research and Development Administration Division of Biomedical and Environmental Research as a national technology resource. The Center provides a coordinating focus for the interchange of information on computing technology, both hardware and software, among participating laboratories, hospitals, and medical clinics.

The goal of the Center will be more effective use of computers in biomedicine with consequent cost savings by (1) providing technology transfer and services, thereby improving the interchange of information among research and medical groups; (2) providing the media (seminars, conferences, and newsletters) for collaboration among members of the user community, focusing attention on outstanding problems of mutual concern; (3) advancing the technology in the subject software through activities which implement the "open code" concept; and (4) providing training in the implementation of specialized software. Through the proposed Center the benefits of computer-assisted technology can be more effectively conveyed to patients of clinicians untrained in the use of computers in clinical medicine.

1. Abstract of paper presented at Fifth National Conference of Society of Computer Medicine, Chicago, Illinois, November 12–14, 1975.

2. Research sponsored by ERDA's Division of Biomedical and Environmental Research, the Society of Nuclear Medicine, the Bureau of Radiological Health (HEW PHS FDA), and the Society for Computer Medicine.

3. Computer Sciences Division, UCC Nuclear Division.

69. SHARING OF COMPUTER PROGRAMS AND TECHNOLOGY IN NUCLEAR MEDICINE[1,2]

Betty F. Maskewitz W. J. McClain[3] R. L. Henne[3]

The expense of developing and maintaining special software and interfaces has been prohibitive to many clinical installations desiring to perform computer-assisted studies. Also, existing developmental groups have been economically unable to adequately transfer their work to such smaller installations. In response to this need and through the efforts of the Society of Nuclear Medicine Computer Committee, the ERDA has established the Biomedical Computing Technology

Information Center at ORNL. The BCTIC collects, organizes, evaluates, and disseminates information on computing technology pertinent to biomedicine in general and nuclear medicine in particular, providing the needed routes of communication between installations, and serving as a clearinghouse for the exchange of biomedical computing software, data, and interface designs. BCTIC services are available to its sponsors and their contractors and to any individual or group willing to participate in mutual exchange.

1 Abstract of paper presented at 16th Annual Meeting of the Society of Nuclear Medicine, Southeastern Chapter, Atlanta, Georgia, October 22-25, 1975

2 Research sponsored by ERDA's Division of Biomedical and Environmental Research, the Society of Nuclear Medicine, the Bureau of Radiological Health (HEW PHS FDA), and the Society for Computer Medicine.

3 Computer Sciences Division, UCC Nuclear Division

70. THE DIGITAL COMPUTER IN NUCLEAR MEDICINE: WHERE DO WE STAND?[1,2]

Betty F. Maskewitz W. J. McClain[3] R. L. Henne[3]

The use of the digital computer in clinical nuclear medicine has been met by varying degrees of acceptance. Recognized in many fields as a necessary instrument, the computer provides a means of acquiring, storing, and manipulating large blocks of data without human intervention. This, on the surface, seems to be adequate reason for utilizing the computer in nuclear medicine clinics. A conflict has arisen, however, for two reasons: (1) the inability of processing algorithms and display media to provide greatly enhanced diagnostic aids in static images; and (2) the costs of establishing, programming, and maintaining computer-based systems.

The advent of dynamic function studies (which can be properly done only with the computer and a display device to provide parametric curves required for diagnosis), coupled with the constant decrease in minicomputer cost over the past five years, has resulted in an increased acceptance of the computer. A recent samping of nuclear medicine computer users at 79 different sites showed a total of 466 separate clinical studies under way in which computers are being used.

Total computer-based systems, with peripheral resources and software necessary for direct application to nuclear medicine, have been developed and are commercially available. Many laboratories and hospitals have developed computer systems and software of their own. However, development costs for software and special interfaces continue to be a serious obstacle to the small clinic or hospital. The transfer of technology from the large developmental groups has been limited to publication in journals, general technical meetings, and personal contact. Such sources are insufficient to permit others to apply the technology without considerable expense.

The need for a facility to encourage technology transfer has long been recognized. In early 1971, members of the Society of Nuclear Medicine (SNM) consulted ORNL staff members to determine how the Society and the Laboratory could jointly attack the problems of information exchange. Since that time, five symposia have been held,[4-8] and much interest has been generated within SNM and the Society of Computer Medicine. This interest resulted in the organization of the SNM Computer Committee, which has as its main objective the fostering of the use of computer-based systems to facilitate diagnostic and therapeutic procedures in nuclear medicine clinics and laboratories.

Through these efforts the BCTIC has been established at ORNL by the Division of Biomedical and Environmental Research of the Energy Research and Development Administration to collect, organize, evaluate, and disseminate information on computing technology. The BCTIC will initially focus on the nuclear medicine community as a result of close collaboration with nuclear medicine specialists during the past five years. As soon as feasible, other biomedical disciplines will be incorporated.

The immediate impact of the Center will be through the communication that it will afford, including (1) newsletters which highlight the important developments in computer-based nuclear medicine systems; (2) proceedings of topical meetings which provide concentrated references on technological advances, both in the computing resources and in the protocols used; (3) a Notebook of Clinical Resources designed to provide a profile of computing resources and protocols utilized by clinics and laboratories throughout the world and to facilitate communication between present and prospective users of similar computing resources and clinical procedures; (4) a clearinghouse for the exchange of computing software, data, and interface designs.

BCTIC services are available to its sponsors and their contractors and to any individual or group willing to participate in mutual exchange.

1. Summary of paper presented at 1975 Winter Meeting of the American Nuclear Society, San Francisco, California, November 16–21, 1975, and at the 16th Annual Meeting of the Society of Nuclear Medicine, Southeastern Chapter, Atlanta, Georgia, October 22–25, 1975.

2. Research sponsored by ERDA's Division of Biomedical and Environmental Research, the Society of Nuclear Medicine, the Bureau of Radiological Health (HEW PHS FDA), and the Society for Computer Medicine.

3. Computer Sciences Division, UCC Nuclear Division.

4. *Proceedings of Symposium on Sharing of Computer Programs and Technology in Nuclear Medicine*, compiled by F. H. Clark, Betty F. Maskewitz, Jane Gurney, and Mildred W. Landay, CONF-710425 (1971).

5. *Proceedings of Second Symposium on Sharing of Computer Programs and Technology in Nuclear Medicine*, compiled by Jane Gurney and Betty F. Maskewitz, CONF-720430 (1972).

6. *Proceedings of Third Symposium on Sharing of Computer Programs and Technology in Nuclear Medicine*, compiled by W. J. McClain and Betty F. Maskewitz, CONF-730627 (1973).

7. *Proceedings of Fourth Symposium on Sharing of Computer Programs and Technology in Nuclear Medicine*, compiled by W. J. McClain and Betty F. Maskewitz, CONF-740531 (1974)

8. *Proceedings of Fifth Symposium on Sharing of Computer Programs and Technology in Nuclear Medicine*, compiled by W. J. McClain and Betty F. Maskewitz, CONF-750124 (1975).

71. AVAILABLE COMPUTER CODES AND DATA FOR RADIATION TRANSPORT ANALYSIS[1,2]

D. K. Trubey Betty F. Maskewitz R. W. Roussin

The Radiation Shielding Information Center, sponsored and supported by the Energy Research and Development Administration and the Defense Nuclear Agency, is a technical institute serving the national shielding community. It acquires, selects, stores, retrieves, evaluates, analyzes, synthesizes, and disseminates information on shielding and ionizing radiation transport. The major activities include: (1) operating a computer-based information system and answering inquiries on radiation analysis, and (2) collecting, checking out, packaging, and distributing large computer codes and evaluated and processed data libraries. The data packages include multigroup-coupled neutron-gamma-ray cross sections and kerma coefficients, other nuclear data, and radiation transport benchmark problem results.

1. Abstract of paper presented at Symposium on Vulnerability and Survivability of Surface and Aerial Targets, San Diego, California, October 21–23, 1975.

2. Research jointly sponsored by Defense Nuclear Agency and ERDA's Division of Reactor Research and Development.

72. SOLUTION OF THE DIFFUSION AND P_1 FINITE-ELEMENT EQUATIONS BY ITERATION[1,2]

E. T. Tomlinson[3] J. C. Robinson[4] D. R. Vondy

To date, various direct techniques have been established as viable methods for obtaining solutions to the finite-element form of the multigroup diffusion equations.[4] For larger problems in which the nodal coupling becomes more complex and the system matrix less dense, iterative procedures such as those used to solve the finite-difference equations become more attractive. However, to efficiently employ these procedures it is necessary to have a sound mathematical understanding of the problem at hand. If no theory is available for the particular problem being studied, the properties of the system can be assumed and then a number of numerical experiments can be performed to test the validity of the assumptions.

The matrix equation obtained from the finite-element form of the multigroup diffusion equations in triangular geometry has the following form:

$$M\phi = S \; . \tag{1}$$

It is assumed that, like the finite-difference equations, M is p-cyclic[5] of order 2. Based on the properties of p-cyclic matrices, the iterative fluxes can be driven by the following equation:

$$\phi_i^n = \phi_i^{n-1} + \beta_n (\phi_i^{n*} - \phi_i^{n-1}) \; , \tag{2}$$

where ϕ_i^{n*} is the newly calculated flux value at node i, and β_n is the overrelaxation coefficient, which can be optimized as follows:

$$\beta_{opt} = \frac{2.0}{1.0 + \sqrt{1.0 - \mu^2}} \; , \tag{3}$$

where

$$\mu^2 = \frac{1}{\lambda}\left(\frac{\lambda + \beta - 1}{\beta}\right)^2 \; . \tag{4}$$

λ = dominate eigenvalue of the iterative process,
β = current overrelaxation coefficient,
μ = spectral radius.

To test the applicability of this theory, an iterative procedure consisting of block inversions coupled with successive overrelaxation of the fluxes was developed. An estimate of the dominant eigenvalue is obtained using the ratio of successive values of the L_1 norm. The equations are iterated with a fixed value of β until the process stabilizes and the dominant eigenvalue is identified. This is done for various values of β.

The results for a one-group homogeneous problem are shown in Fig. 72.1. The theory, using the calculated value of μ, predicts that the optimum value of β is 1.385. The numerical results indicate that the optimum value is 1.382.

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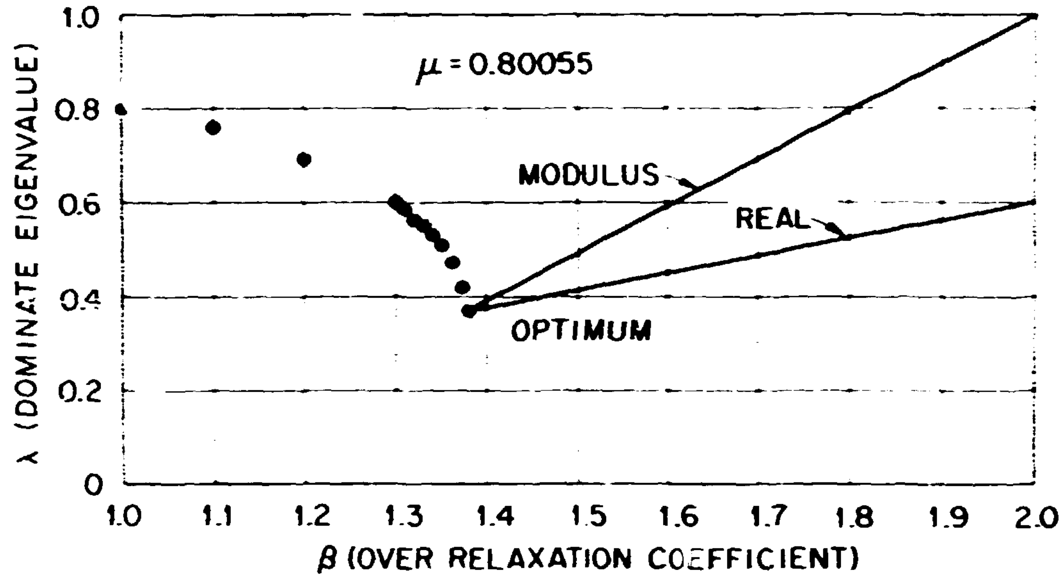

Fig. 72.1. Dependence of the dominating eigen value on the overrelaxation coefficient.

These results indicate that the finite-element equations can be iterated using the procedures developed for the finite-difference method. The ability to incorporate these iterative procedures with the geometric flexibility of the finite-element method can make this a powerful tool.

1. Summary of paper presented at 1975 Winter Meeting of the American Nuclear Society, San Francisco, California, November 16–21, 1975.
2. Research sponsored by ERDA's Division of Reactor Research and Development.
3. The University of Tennessee.
4. H. G. Kaper, G. K. Leaf, and A. J. Lindeman, *HOD, a Higher Order Diffusion Program*, ANL-TM-243 (September 1973).
5. R. S. Varga, *Matrix Iterative Analysis*, chap. 4, p. 106, Prentice-Hall, Englewood Cliffs, New Jersey, 1962.

73. VENTURE: A CODE BLOCK FOR SOLVING MULTIGROUP NEUTRONICS PROBLEMS APPLYING THE FINITE-DIFFERENCE DIFFUSION-THEORY APPROXIMATION TO NEUTRON TRANSPORT[1,2]

D. R. Vondy T. B. Fowler G. W. Cunningham[3]

This report documents the computer code block VENTURE designed to solve multigroup neutronics problems with application of the finite-difference diffusion-theory approximation to neutron transport (or alternatively simple P_1) in up to three-dimensional geometry. It uses and generates interface data files adopted in the cooperative effort sponsored by the Reactor Physics Branch of the Division of Reactor Research and Development of the U.S. Energy Research and Development Administration. Several different data handling procedures have been incorporated to provide considerable flexibility; it should be possible to solve a wide variety of problems on a variety of computer configurations relatively efficiently. Also, it should be straightforward to improve the efficiency for a particular computer and small range of problem type by changing one of the programmed data handling procedures. The programming in FORTRAN is straightforward, although data are transferred in blocks between auxiliary storage devices and main core, and direct access schemes are used. The size of problems which can be handled is essentially limited only by cost of calculation, since the arrays are variably dimensioned.

The more common orthogonal coordinate systems arising in reactor analysis applications have been treated in from one through three dimensions. These include the slab, the cylinder, θ-R, θ-R-Z, and hexagonal and triagonal coordinate systems in two and three dimensions. Only the mesh-centered finite-difference formulation has been programmed. There is provision for the more common boundary conditions, including the repeating boundary, 180° rotational symmetry, and the 90° slab and the 60 and 120° triangle rotational symmetry conditions.

A variety of types of problems may be solved: the usual eigenvalue problem; a direct criticality search on the buckling, on a reciprocal velocity absorber (prompt mode), or on nuclide concentrations; or an indirect criticality search on nuclide concentrations or on dimensions. First-order perturbation analysis capability is available at the macroscopic cross-section level.

1. Abstract of ORNL-5062 (1975); see, also, "Computer Code Abstract VENTURE," *Nucl. Sci. Eng.* **58**, 339 (1975).
2. Research sponsored by ERDA's Division of Reactor Research and Development.
3. Computer Sciences Division, UCC Nuclear Division.

74. A USER'S MANUAL FOR TDT (TIME-DEPENDENT TASK)[1,2]

E. T. Tomlinson[3] H. L. Dodds[4]
R. A. Lillie J. C. Robinson[3]

The TDT solves the one-dimensional multigroup form of the time-dependent reactor kinetics equations, allowing an arbitrary number of delayed neutron groups. The program can also solve standard static problems, such as eigenvalue problems (transport only), distribution source problems and boundary source problems, using either diffusion theory or transport theory. Convergence problems associated with highly multiplicative media are circumvented, and such problems are readily calculable.

TDT is written in FORTRAN-IV, using flexible dimensioning, and is designed for computers in the IBM-360 series. It employs a combination scattering and transfer matrix method to solve the transport equation in the spatial domain and a predictor-corrector method to obtain solutions in the temporal domain.

The principal restrictions are available storage and computational time. Since the code is flexibly dimensioned, there are no internal restrictions on group structure, quadrature, and number of coordinates. The code contains an option whereby it will calculate the most efficient time mesh. The cylindrical geometry option is not operational in this version.

A 10-sec rod-drop transient in a 7-region, 5-group, S_4 problem requires 5 min on the IBM-360/91. A point kinetics approximation is available which significantly reduces the computational time.

This version of TDT requires 178K of storage plus data block storage to execute on the IBM-360/91. One tape is required to utilize the flux tape option.

1. Abstract of ORNL-TM-4869 (May 1975).
2. Research sponsored by ERDA's Division of Reactor Research and Development.
3. The University of Tennessee.
4. E. I. du Pont de Nemours and Co., Aiken, South Carolina.

75. THE MORSE MONTE CARLO RADIATION TRANSPORT CODE SYSTEM[1,2]

M. B. Emmett[3]

The MORSE Monte Carlo code system has been in continuous use for several years now and has subsequently undergone much revision. In addition to this, its development has continued, and its applications have been expanded into new areas requiring development of additional computer programs. Because the documentation of the various modules of MORSE is scattered throughout several reports and because numerous revisions have been required to update the documentation for the current versions of routines, it was decided to write one complete report describing all aspects of the MORSE system and related computer codes. As might be expected, this report draws heavily from the previous reports. All information necessary to use these codes is consolidated into one report. There are sections containing descriptions of the MORSE and PICTURE codes, input descriptions, sample problems, derivations of the physical equations, and explanations of the various error messages. This report attempts to document the IBM-360, UNIVAC-1108, and CDC-6600 versions of MORSE and its auxiliary codes. Most options are available on all three machines, but some are not because one or more of the machines do not have some particular capability. It is expected that some of the parts being published now will be expanded. In particular, the sample problem notebook in Part 3 will have additional examples to illustrate more of the options available. Additional parts such as a manual for DOMINO and a description of the collision density fluence estimator, which is included in the basic MORSE on the UNIVAC machine, will also be added. Because time was at a premium, no attempt was made to update flow charts and include them. Users may refer to the previous reports but should keep in mind that some of the charts are out of date.

This report describes only the MORSE with combinatorial geometry (CG) and not the 05R-type geometries that are no longer maintained in MORSE. This decision was made because CG will handle all types of geometries expeditiously and because it is quite cumbersome to maintain multiple versions of a code as large and complicated as MORSE.

It is worth noting that the MORSE with combinatorial geometry *does not always produce the same random number sequence* when a job is restarted at some intermediate point. This results from the fact that this geometry package has a "memory," that is, it uses a table lookup rather than a recalculation when the same path is encountered. All 05R-type geometries calculated the path each time and, therefore, repeated random number sequences.

The format of this documentation is such that updates, deletions, and additions should be easily done. Each "part" is like a separate report, because it contains its own table of contents and references and its pages are numbered beginning with 1.

1. Abstract of ORNL-4972 (February 1975).
2. Research sponsored by the Defense Nuclear Agency.
3. Computer Sciences Division, UCC Nuclear Division.

76. TECHNIQUES FOR EFFICIENT MONTE CARLO SIMULATION. VOLUMES I–III[1,2]

E. J. McGrath, D. C. Irving et al.[3]

This report consists of three volumes that present techniques and methods for developing efficient Monte Carlo simulation. Originally published as a series of reports prepared by Science Applications, Inc., for the Office of Naval Research, they have been edited and reprinted by the Radiation Shielding Information Center in three volumes as follows:

Vol. I. Selecting Probability Distributions

This volume provides a straightforward approach and associated techniques for selecting the most appropriate probability distributions for use in Monte Carlo simulations. Part I, "Basic Considerations," presents the underlying concepts and principles for selecting probability distributions. Part II, "Selection of Distributions," gives the mathematical models representing stochastic processes and presents step-by-step procedures for identification and selection of the appropriate probability distributions based upon the degree of knowledge and available data for the random variable under study.

Vol. II. Random Number Generation for Selected Probability Distributions

Algorithms for efficient generation of random numbers from various probability distributions are presented in both a flowchart form and as a sample FORTRAN subroutine. Twenty-two different distributions, including all commonly encountered discrete and continuous functions, the Weibull, Johnson, and Pearson families of empirical distributions, and histogram distributions, are covered. The general techniques to apply in deriving a random-number selection scheme for an arbitrary distribution are discussed. A machine-independent subroutine for generating uniform random numbers is also described.

Vol. III. Variance Reduction

This document describes the basic concepts of variance reduction (Part I), and a methodology for application of variance reduction techniques is presented in Part II. Appendixes include the basic analytical expressions for application of variance reduction schemes as well as an abstracted bibliography.

The techniques considered here include importance sampling, Russian roulette and splitting, systematic sampling, stratified sampling, expected values, statistical estimation, correlated sampling, history reanalysis, control variates, antithetic variates, regression, sequential sampling, adjoint formulation, transformations, and orthonormal and conditional Monte Carlo. Emphasis has been placed on presentation of the material for application by the general user. This has been accomplished by presenting a step-by-step procedure for selection and application of the appropriate technique(s) for a given problem.

1. Abstracts of ORNL-RSIC-38, Vols. I–III (April 1975).
2. RSIC publication partially sponsored by Defense Nuclear Agency.
3. Science Applications, Inc.

77. DEVELOPMENT OF A CODE SYSTEM FOR DETERMINING RADIATION PROTECTION OF ARMORED VEHICLES (THE VCS CODE)[1,2]

W. A. Rhoades

A system of coupled computer codes is developed for the calculation of the protection from nuclear radiation provided by an armored vehicle. The code system uses a discrete-ordinates air-transport calculation coupled to an adjoint Monte Carlo calculation. The vehicle description is supplied in combinatorial geometry form. Elements of the method were tested against experimental data. A realistic tactical nuclear weapon problem is demonstrated.

1. Abstract of ORNL-TM-4664 (October 1974).
2. Research sponsored by the Ballistics Research Laboratories, Aberdeen, Maryland.

78. OPERATING INSTRUCTIONS FOR THE HEAVY-ION CODE HIC-1[1,2]

R. T. Santoro   H. W. Bertini
T. A. Gabriel   N. M. Larson[3]
O. W. Hermann[3]

The heavy-ion code HIC-1 is a Monte Carlo code that calculates continuum-state transitions between projectile and target in heavy-ion reactions at energies $\gtrsim$50 MeV/nucleon. The underlying assumption in the model is that the reaction can be represented as the interaction of two Fermi gases that pass through each other. During the passage of the projectile through the target, those individual nucleons of the projectile that are in the region of overlap undergo quasi-free reactions with the nucleons of the target. During this time, a cascade takes place simultaneously in the projectile and in the target, and the nucleons that have been jarred loose from their bound states and that have managed to survive capture during the development of the cascade escape from the projectile and the target. They are emitted as free nucleons in various directions and with a variety of energies. Depending on the projectile energy, pions may also be produced and emitted from the projectile and from the target. The remaining fragments of the projectile and target move off in highly excited states emitting evaporation particles, and ultimately gamma rays, until this excited energy is lost, at which time the reaction is completed. The calculation is carried out up to the point of gamma-ray emission, but it does not include this process.

The physical processes and calculational sequences used in the code have been described elsewhere. It should be pointed out that this version of the code is a preliminary version only. Work on improved versions will depend on available funding for this program. Only the salient features of the code necessary to facilitate the calculations are discussed in this report.

There are three phases in the calculation of heavy-ion reactions: the development of the nuclear cascade, the treatment of excited nuclei by evaporation, and the analysis of data to obtain the desired information (i.e., spectra of the heavy fragments and emitted nucleons and pions, their angular distributions, calculation of the cross sections for the production of residual nuclei emitted at all angles and energies, etc.).

A brief description of the organization of the code and of the preparation of data is discussed in Sect. II. The input requirements and programming options of the code are given in Sect. III, and, finally, a sample problem illustrating the input requirements and the resulting output is given in Sect. IV.

1. Abstract of ORNL-TM-4791 (March 1975).
2. Research sponsored by the National Aeronautics and Space Administration.
3. Computer Sciences Division, UCC Nuclear Division.

79. SHIELDING CALCULATIONS FOR A 200-MeV PROTON ACCELERATOR AND COMPARISONS WITH EXPERIMENTAL DATA[1,2]

R. G. Alsmiller, Jr. R. T. Santoro J. Barish[3]

In a series of previous papers,[4-7] the results of neutron transport calculations carried out with the method of discrete ordinates for various neutron spectra incident on a variety of thick shields have been presented. In this paper, results similar to those given previously will be presented and compared with the experimental data of Distenfeld[8] for 200-MeV protons incident on a thick water shield. In the experiment of Distenfeld, and using activation detectors, data were taken as a function of distance measured transversely to the proton beam axis. The activation reactions considered were $^{12}C(n,2n)^{11}C$, $^{27}Al(n,\alpha)^{24}Na$, $^{23}Na(n,\gamma)^{24}Na$, and the production of ^{24}Na from neutron-induced reactions in chlorine.

The geometry used to obtain the calculated results for comparison with the experimental data is only approximately that used experimentally. In the calculations, the energy distribution of neutrons produced in a specific angular interval by proton-nucleus nonelastic collisions when 200-MeV prot are stopped in a thick water shield was obtained using the Monte Carlo transport code HETC,[9] and this distribution was transported as if it were emitted isotropically at the center of a water sphere. Neutron energy distributions have been calculated for the angular intervals of 60 to 120° and 80 to 100° with respect to the direction of the incident proton beam, and the results of transport calculations obtained using both of these distributions will be presented and compared with the experimental data. The neutron transport calculations were carried out with the one-dimensional discrete ordinates code ANISN[10] in the same manner as in refs. 4–7. The sources of the cross-section data are the same as that given in ref. 7.

The calculated and experimental activation results are compared in Fig. 79.1. On the ordinate of the figure, ϕ_i is the neutron flux per unit energy averaged over the activation cross section of type i, and r is the radial distance into the water sphere. The experimental results shown in the figure have an absolute normalization. The calculated results, except for those given for the reaction $^{23}Na(n,\gamma)^{24}Na$, also have an absolute normalization. At the larger radii, the two calculated curves bracket the experimental results. Distenfeld[8] has also measured the dose equivalent at a depth of 30 ft in the water, and comparisons between calculated dose-equivalent results and this measured value have also been obtained and will be presented.

Results have also been obtained and will be presented for the neutron spectra, averaged over various angular intervals, produced when 200-MeV protons are incident on thin and thick targets of aluminum and for the dose equivalent as a function of radius in a concrete shield when these neutron spectra are isotropically incident at the center of a spherical concrete shield.

1. Summary of paper presented at the American Nuclear Society Meeting, New Orleans, Louisiana, June 8-13, 1975; *Trans. Am. Nucl. Soc.* **546** (1975); also summary of ORNL-TM-4754 (January 1975).
2. Research sponsored by ERDA's Division of Physical Research.

ORNL-DWG 74-12284

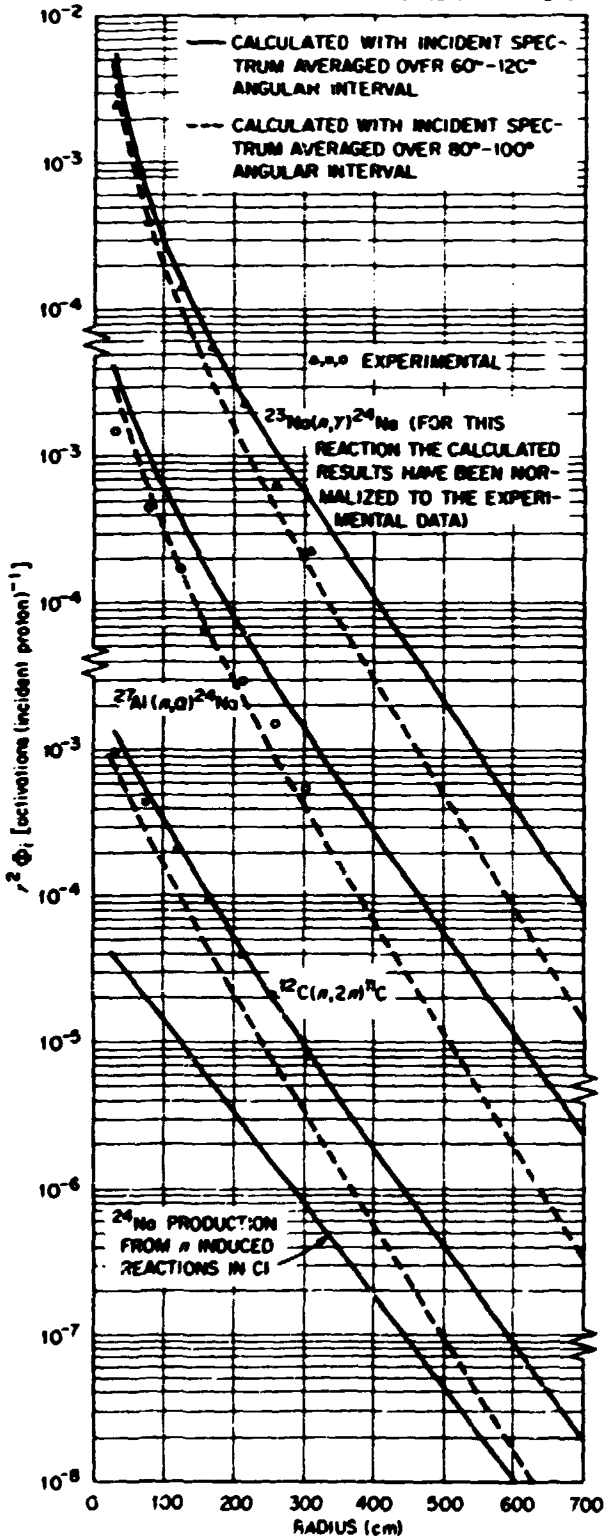

Fig. 79.1. Experimental and calculated activation data multiplied by r^2 vs radius in the water shield.

3. Computer Sciences Division, UCC Nuclear Division.
4. R. G. Alsmiller, Jr., F. R. Mynatt, J. Barish, and W. W. Engle, Jr., "High-Energy (<400-MeV) Neutron Transport Using the Method of Discrete Ordinates," *Nucl. Sci. Eng.* **36**, 251 (1969).
5. R. G. Alsmiller, Jr., F. R. Mynatt, J. Barish, and W. W. Engle, Jr., "Shielding Against Neutrons in the Energy Range 50 to 400 MeV," *Nucl. Instr. Meth.* **72**, 213 (1969).
6. R. W. Roussin, R. G. Alsmiller, Jr., and J. Barish, "Calculations of the Transport of Neutrons and Secondary Gamma Rays through Concrete for Incident Neutrons in the Energy Range 15 to 75 MeV," *Nucl. Eng. Design* **24**(2) 250 (1973).
7. R. G. Alsmiller, Jr., and J. Barish, *Particle Accelerators* **5**, 155 (1973).
8. C. H. Distenfeld, *Shielding Measurements—200-MeV Linac*, BNL-18025 (1973).
9. K. C. Chandler and T. W. Armstrong, *Operating Instructions for the High-Energy Nucleon-Meson Transport Code HETC*, ORNL-4744 (1972).
10. W. W. Engle, Jr., *A Users Manual for ANISN, a One-Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering*, K-1693, Computing Technology Center, Union Carbide Corporation (1967).

80. HIGH-ENERGY (40 MeV $\leq E_\gamma \leq$ 400 MeV) PHOTONUCLEAR INTERACTIONS[1,2]

T. A. Gabriel

Comparison between theoretical and recent experimental photoproton spectra are presented for 400-MeV bremsstrahlung on carbon, aluminum, copper, and gold. Experiments are suggested which would facilitate further theoretical considerations.

1. Abstract of ORNL-TM-4926 (May 1975) and of a paper accepted for publication in *The Physical Review*.
2. Research sponsored by ERDA's Division of Physical Research.

81. RELATIVISTIC ANGULAR MOMENTUM RELATIONSHIPS FOR HIGH-ENERGY HEAVY-ION REACTIONS[1,2]

H. W. Bertini N. M. Larson[3]

Established properties of the relativistic intrinsic and orbital angular-momentum tensors are reviewed, and relationships between the components of these tensors are derived. These relationships reduce the number of unknowns that must be determined for the calculation of the relativistic spin of the heavy fragments in high-energy heavy-ion interactions, in the limit where quantum mechanics may be ignored. An example using the intranuclear-cascade model is presented.

1. Abstract of ORNL-TM-4667 (January 1975) and paper published in *Phys. Rev.* **C12**(3), 871 (1975).
2. Research sponsored jointly by the National Aeronautics and Space Administration and ERDA's Division of Physical Research.
3. Computer Sciences Division, UCC Nuclear Division.

82. SPALLATION REACTIONS: CALCULATIONS[1,2]

H. W. Bertini

Current methods for calculating spallation reactions over various energy ranges are described and evaluated. Recent semiempirical fits to existing data will probably yield the most accurate predictions for these reactions in general. However, if the products in question have binding energies appreciably different from their isotopic neighbors and if the cross section is ~30 mb or larger, then the intranuclear-cascade-evaporation approach is probably better suited.

1. Abstract of paper presented at the International Symposium on Spallation Nuclear Reactions and Their Applications in Astrophysics and Radiotherapy, University of Pennsylvania, Philadelphia, May 2, 1975.
2. Research sponsored by ERDA's Division of Physical Research.

83. NUCLEON-MESON TRANSPORT CALCULATIONS[1,2]

R. G. Alsmiller, Jr.

A review of medium- and high-energy nucleon-meson transport calculations carried out at the Oak Ridge National Laboratory in the past several years is presented.

1. Abstract of paper presented at the International Symposium on Spallation Nuclear Reactions and Their Applications in Astrophysics and Radiotherapy, University of Pennsylvania, Philadelphia, May 2, 1975.
2. Research sponsored by ERDA's Division of Physical Research

84. FREQUENCY OF OCCURRENCE OF VARIOUS NUCLEAR REACTIONS WHEN FAST NEUTRONS ($\lesssim$50 MeV) PASS THROUGH TISSUE-EQUIVALENT MATERIAL[1,2]

R. G. Alsmiller, Jr. J. Barish[3]

Calculated results are presented for the frequency with which various partial nuclear-reaction cross sections are utilized when fast neutrons ($\lesssim$50 MeV) are transported through a tissue-equivalent phantom to obtain an indication of which cross sections are of most importance for radiotherapy applications and are therefore in need of experimental verification.

1. Abstract of ORNL-TM-4970 (July 1975).
2. Research sponsored by ERDA's Division of Physical Research.
3. Computer Sciences Division, UCC Nuclear Division.

85. BIOMEDICAL RADIATION TRANSPORT CALCULATIONS AS AN APPLICATION OF NUCLEAR DATA[1,2]

R. G. Alsmiller, Jr.

The potential advantages of using negatively charged pions, neutrons, protons, alpha particles, and heavier ions in cancer radiotherapy have been recognized for some time, and radiobiological experiments and therapeutic exposures with many of these radiation modalities are in progress or will be undertaken in the near future.[3] For use in radiotherapy, the energy range of interest of the various particles is such that the basic nuclear-physics data needed to study their transport and the nuclear-reaction products produced in tissue during the transport process are very extensive and, in general, are not available experimentally. For biomedical applications, it is particularly important that the charged particles produced by nuclear reactions in tissue be considered, since these charged particles are very significant in calculating quantities of biological interest, such as cell-survival probabilities and oxygen enhancement ratios. Much of the data needed to carry out the transport calculations of biomedical interest can be obtained from the intranuclear-cascade-evaporation model of nuclear reactions, and a variety of calculations that utilize particle-production data obtained from this model have been carried out at ORNL for incident pions, neutrons, protons, and alpha particles. Many of the calculated results have been compared with experimental data, and good agreement has been obtained. In the case of incident heavy ions ($A > 4$), the available experimental data on particle production from their nonelastic interactions with tissue nuclei are very limited, and no satisfactory theoretical model for producing the required data is available. Nevertheless, calculated results based on experimental fragmentation data have been obtained at the Princeton Particle Accelerator and at the Lawrence Berkeley Laboratory, and the results have been found to be in good agreement with the available experimental data.

1. Abstract of invited paper presented at the Conference on Nuclear Cross Sections and Technology, Washington, D.C., March 3–7, 1975, *Bull. Am. Phys. Soc., Ser. II* **20**, 157 (1975); full paper published in *Nuclear Cross Sections and Technology*, vol. II, ed. by R. A. Schrack and C. D. Bowman, NBS Special Publication 425, p. 533 (1975).

2. Research sponsored by ERDA's Division of Physical Research.

3. *Proceedings of the Conference on Particle Accelerators in Radiation Therapy, October 2–5, 1972*, Los Alamos Scientific Laboratory, Los Alamos, N.M., LA-5180-C (1973), and the references given therein.

86. THEORETICAL DOSIMETRY[1,2]

R. G. Alsmiller, Jr.

The potential advantages of using negatively charged pions, neutrons, protons, alpha particles, and heavier ions in cancer radiotherapy have been recognized for some time, and radiobiological experiments and therapeutic exposures with many of these radiation modalities are in progress or will be undertaken in the near future.[3] The basic problem of theoretical dosimetry is to calculate, as a function of position, the type, energy, and angular distributions of the charged particles that are produced when a phantom is irradiated by any of these incident particles, since this information may be combined with other data to obtain quantities of interest such as absorbed dose, LET spectra, and cell-survival probabilities. The validity of calculated results must be determined by comparison with experimental data, but the value of the calculations lies in the fact that they can provide much detailed information that cannot readily be obtained experimentally. Uncertainties in the calculations arise primarily from uncertainties in the basic nuclear cross-section data that are available rather than from the transport methods that are used, since, for use in radiotherapy, the energy range of interest of the various particles is such that the available experimental data on particle production from nonelastic collisions are very meager. Thus transport calculations must be based almost entirely on theoretical cross-section data such as those provided by the intranuclear-cascade-evaporation model of nuclear reactions. The majority of the dosimetry calculations performed to date have been done using Monte Carlo methods. For many of the particle types being considered for use in radiotherapy, comparisons between calculated and experimental absorbed-dose data are available, and, for a few of the particle types, comparisons between calculated and experimental cell-survival probabilities are available. In general, the comparisons of cell-survival probabilities provide a more stringent test of the calculations than do comparisons of absorbed doses, because the high LET particles that contribute very appreciably to cell-survival probabilities contribute only a small amount to the absorbed dose.

1. Abstract of invited paper presented at the American Physical Society Meeting, Washington, D.C., April 28–May 1, 1975, *Bull Am Phys. Soc.* **20**(4), 595 (1975).

2. Research sponsored by ERDA's Division of Physical Research.
3. *Proceedings of the Conference on Particle Accelerators in Radiation Therapy*, October 2–5, 1972, Los Alamos Scientific Laboratory, Los Alamos, New Mexico, LA-5180-C (1973), and the references given therein.

87. A CALCULATIONAL APPROACH TO IONIZATION SPECTROMETER DESIGN[1,2]

T. A. Gabriel

Many factors contribute to the design and overall performance of an ionization spectrometer. These factors include the conditions under which the spectrometer is to be used, the required performance, the development of the hadronic and electromagnetic cascades, leakage and binding energies, saturation effects of densely ionizing particles, nonuniform light collection, sampling fluctuations, etc. In this paper, the calculational procedures developed at ORNL which have been applied to many spectrometer designs and which include many of the influencing factors in spectrometer design are discussed. The incident-particle types that can be considered with some generality are protons, neutrons, pions, muons, electrons, positrons, and gamma rays. Charged kaons can also be considered but with less generality. The incident-particle energy range can extend into the hundreds of GeV. The calculations have been verified by comparison with experimental data but only up to approximately 30 GeV. Some comparisons with experimental data will also be discussed and presented so that the flexibility of the calculational methods can be demonstrated.

1. Abstract of paper presented at the International Workshop on Spectrometer Design, Fermi National Accelerator Laboratory, Batavia, Illinois, May 9–10, 1975; full text published in *Proceedings of the Calorimeter Workshop, May, 1975*, ed. by M. Atac, p. 13.
2. Research sponsored by ERDA's Division of Physical Research.

88. PERFORMANCE OF A LARGE IRON-PLASTIC IONIZATION SPECTROMETER[1,2]

J. D. Amburgey[3]   T. A. Gabriel

The calculated performance of a large iron-plastic ionization spectrometer to incident charged particles in the low-GeV/c momentum range is presented. The spectrometer consists of six modules with successive modules having increasing heights and widths. The first four modules have the same depth, and each module consists of 16 sheets of iron (1 cm thick) sandwiched between 16 sheets of solid scintillator (0.476 cm thick). The last two modules each consist of eight sheets of iron (1 cm thick) and eight sheets of plastic (0.476 cm thick). The average height and width are 282 and 155 cm respectively.

The particle types considered are electrons, protons, and pions, with momenta ranging from 0.5 to 8 GeV/c. The calculational procedure is the same as that described elsewhere.[4] Many of the factors that contribute to the overall performance of a spectrometer, such as details of the hadronic and electromagnetic cascades, leakage and binding energies, saturation effects of densely ionizing particles, etc., and that are often omitted or extremely simplified in other calculations are included in these calculations.

The expected pulse-height distributions for 1- and 2-GeV/c π^- and 2-GeV/c π^+ are given in Fig. 88.1. Even though the resolution is quite large, the pulse-height distributions do not exhibit a slowly decreasing tail. At 2 GeV/c there is little difference between the π^- and π^+ pulse-height distributions.

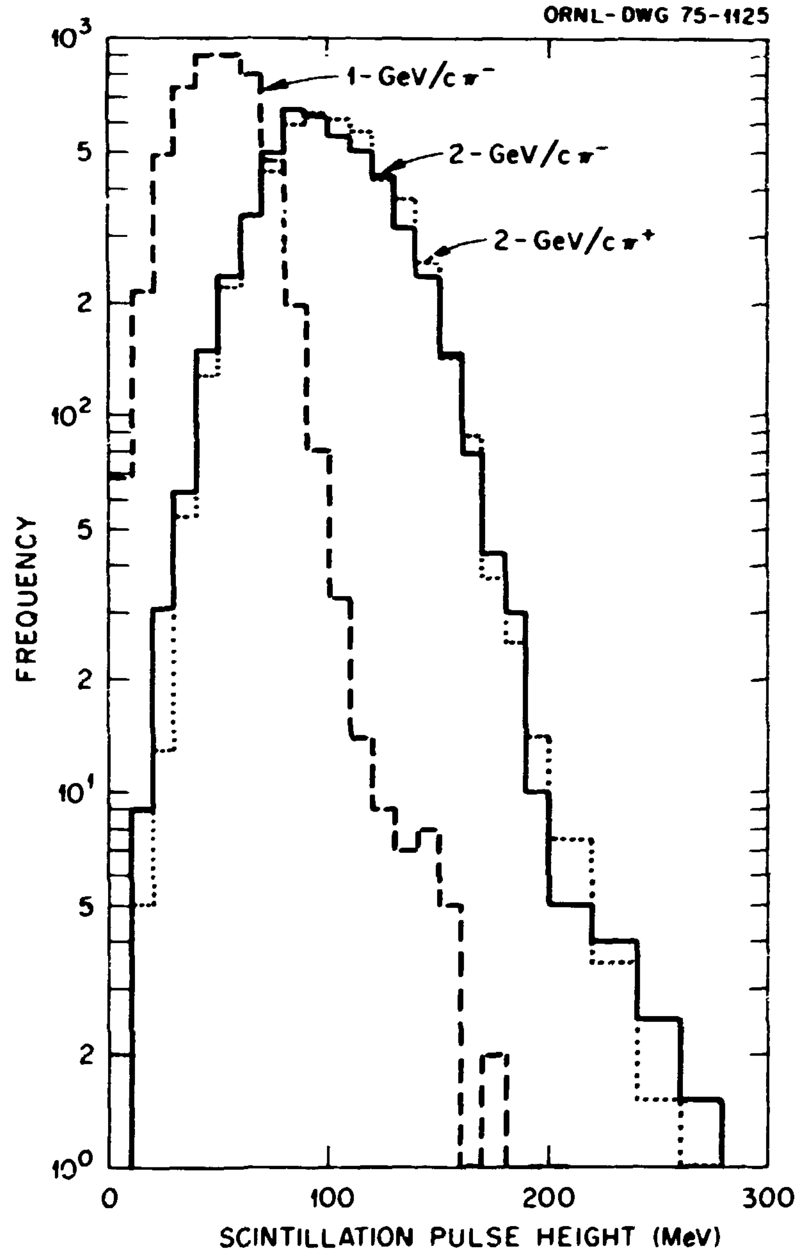

Fig. 88.1. Pulse-height distribution for 1- and 2-GeV/c incident π^- and 2-GeV/c incident π^+.

1. Summary of paper presented at the American Nuclear Society Meeting, New Orleans, Louisiana, June 8–13, 1975; *Trans. Am. Nucl. Soc.* **21**, 541 (1975).
2. Research sponsored by ERDA's Division of Physical Research.
3. Computer Sciences Division, UCC Nuclear Division.
4. T. A. Gabriel and J. D. Amburgey, *Nucl. Instr. Meth.* **116**, 333 (1974).

89. CALCULATED PERFORMANCE OF A SEGMENTED PYRAMID-SHAPED CALORIMETER OF IRON AND PLASTIC[1,2]

J. D. Amburgey[3] T. A. Gabriel

The calculated performance of a segmented pyramid-shaped calorimeter of iron and plastic is presented. The types of incident particles considered are protons, charged pions, and electrons. The momentum range is 0.2 to 8 GeV/c. The calculated resolution ($\Delta E/E$) obtained for protons and charged pions decreases from ~90% at 1 GeV to ~50% at 8 GeV. For incident electrons, the resolution decreases from ~25% at 1 GeV to ~8% at 8 GeV.

1. Abstract of ORNL-TM-5118 (in press).
2. Research sponsored by ERDA's Division of Physical Research.
3. Computer Sciences Division, UCC Nuclear Division.

90. PRELIMINARY DESIGN CALCULATIONS FOR AN IONIZATION SPECTROMETER FOR USE IN COLLIDING BEAM EXPERIMENTS[1,2]

T. A. Gabriel J. D. Amburgey[3] R. T. Santoro

The calculated response of an ionization spectrometer containing NaI, Fe, and Ar to charged pions and kaons in the 0.1- to 1.0-GeV/c momentum range is presented.

1. Abstract of ORNL-TM-5040 (September 1975) and of a paper accepted for publication in *Nuclear Instruments and Methods*.
2. Research sponsored by ERDA's Division of Physical Research.
3. Computer Sciences Division, UCC Nuclear Division.

91. CALCULATED PERFORMANCE OF A MINERAL-OIL-IRON IONIZATION SPECTROMETER[1,2]

T. A. Gabriel J. D. Amburgey[3] R. T. Santoro

The calculated performances of two slightly different mineral-oil-iron ionization spectrometer designs are presented and compared with experimental data. The particle types considered are electrons, protons, and pions with momenta ranging from 3 to 17.3 GeV/c. Many of the factors that contribute to the overall performance of such a spectrometer and that are often omitted in other calculations are included in these calculations. In general, the agreement between the calculated and the experimental results is good.

1. Abstract of ORNL-TM-4803 (June 1975) and of a paper accepted for publication in *Nuclear Instruments and Methods*.
2. Research sponsored by ERDA's Division of Physical Research.
3. Computer Sciences Division, UCC Nuclear Division.

92. COMPARISON OF EXPERIMENT AND CALCULATION OF THE ORNL OXYGEN INTEGRAL EXPERIMENT[1,2]

S. N. Cramer E. M. Oblow

In ORNL's continuing cross-section testing and evaluation program for the Defense Nuclear Agency, an oxygen integral experiment and associated calculations have been performed. Comparison of neutron and gamma-ray count rates and energy spectra between the experiment and calculations have been made in an effort to improve and validate the current oxygen data file, DNA 4134 MOD 2.

The experimental setup is similar to that used for the nitrogen experiment.[3,4] A linear accelerator (ORELA) produced a 13.25-cm-diam white neutron beam (up to 20 MeV) incident on a 20-cm-diam liquid-oxygen sample contained in a thin evacuated Pyrex Dewar. An NE-213 detector, used for both neutrons and gamma rays, was placed 18 cm from the beam center line and at four different positions corresponding to angles of 30, 55, 90, and 125° with respect to the incident beam at the center of the sample. The experiment and the calculation (using the MORSE multigroup Monte Carlo code) were both performed in the time-dependent mode. The results are converted to incident neutron energy dependence from the neutron flight times in the beam tube.

The integral count rates are shown in Fig. 92.1 for the 90° detector position. These comparisons are in generally good agreement for both neutron and gamma rays, and they are also representative

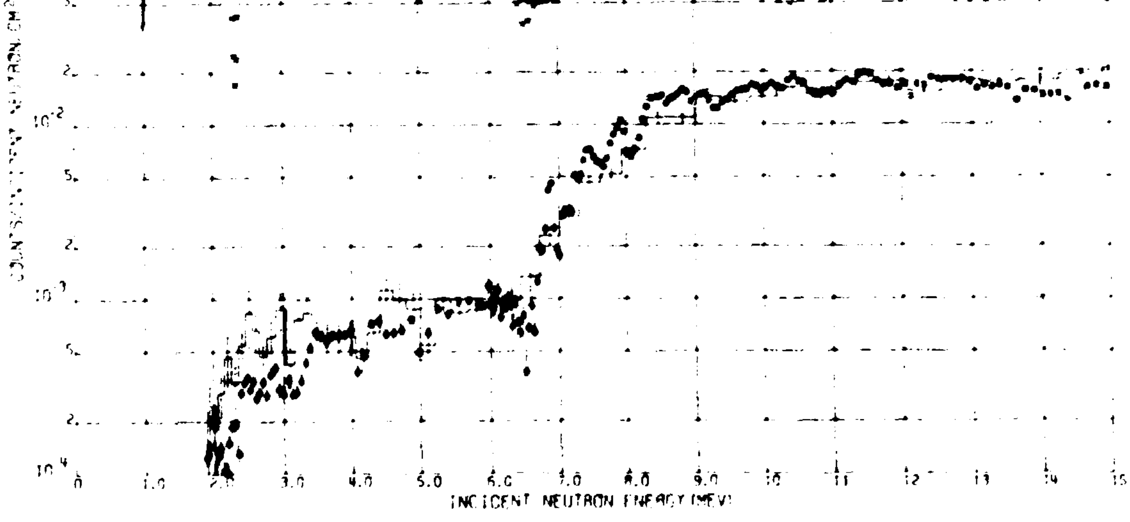

Fig. 92.1. Comparison of measured and calculated integral count rates due to neutrons and gamma rays emitted from 90° from a liquid oxygen sample positioned in a white neutron beam.

of the other three detector results. The calculated 30° neutron counts are consistently a few percent higher than the experimental results for all energies above the 2.37-MeV minimum.

Figure 92.2 shows the 90° neutron energy spectra for four incident energy ranges. A disagreement is seen on the upper side of the elastic-scattering (high-energy) peaks, and this discrepancy is reflected in the integral curve. The calculated results between the peaks are due primarily to an empirical energy resolution function. The spectra comparisons for the other 90° energy intervals and also the other detector positions are in the same general agreement as in Fig. 92.2, with the 30° spectra having the largest difference in the elastic peak. In the gamma-ray spectra comparisons, the calculations show less well-defined peaks and valleys than the experiments, due to the multigroup structure, but the agreement is good.

From the comparison of the experiments and calculated results of the oxygen integral experiment, it can be seen that the neutron and gamma-ray cross-section data and gamma-ray production data for the DNA 4134 MOD 2 data file appear to be in good order. The major deficiency seems to be in the neutron elastic-scattering angular distributions at forward angles.

ORNL-DWG 75-1437

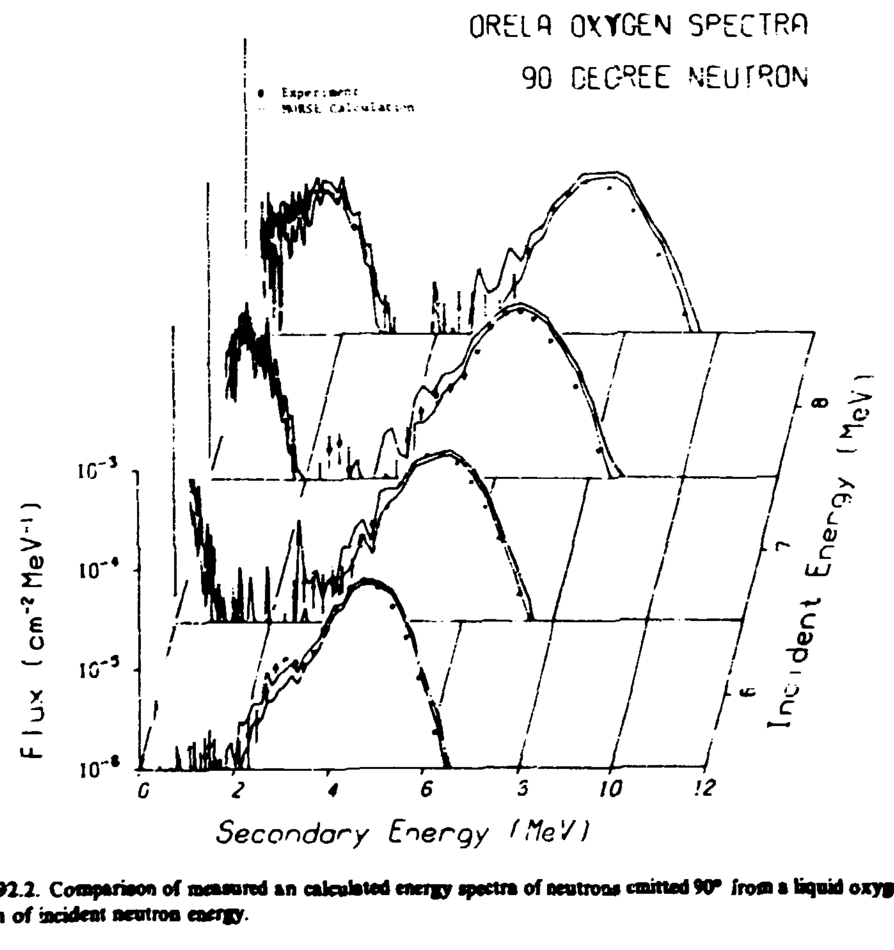

Fig. 92.2. Comparison of measured an calculated energy spectra of neutrons emitted 90° from a liquid oxygen sample as a function of incident neutron energy.

1. Summary of paper presented at the American Nuclear Society Meeting, New Orleans, Louisiana, June 8–13, 1975; *Trans. Am. Nucl. Soc.* **21**, 532 (1975).
2. Research sponsored by Defense Nuclear Agency.
3. S. N. Cramer and E. M. Oblow, "Calculational Analysis of Neutron and Gamma-Ray Experiments in Nitrogen," *Trans. Am. Nucl. Soc.* **17**, 552 (1973).
4. S. N. Cramer and E. M. Oblow, "Neutron and Photon Energy Spectrum Calculations for the Nitrogen Integral Experiment," *Trans. Am. Nucl. Soc.* **19**, 471 (1974).

93. NEUTRON AND SECONDARY-GAMMA-RAY TRANSPORT CALCULATIONS FOR 14-MeV AND FISSION NEUTRON SOURCES IN AIR-OVER-GROUND AND AIR-OVER-SEAWATER GEOMETRIES[1,2]

J. V. Pace III[3] D. E. Bartine F. R. Mynatt

Radiation transport calculations have been performed with the DOT-III code for a 14-MeV neutron source and a tactical weapon fission neutron source in air-over-ground and air-over-seawater geometries. The source heights were 1, 50, 100, 200, and 300 m for the air-over-ground calculations and 50 m for the air-over-seawater calculations. The results were obtained as neutron and secondary-gamma-ray fluxes throughout a cylindrical system having a height of approximately 1300 m and a radius of about 1500 m, the lower 50 cm of the cylinder being either ground or seawater. Several ionization and tissue-dose response functions were applied to the fluxes obtained for positions on the interfaces. The air-over-ground results indicate that an optimal source height can be specified for the maximum neutron dose at a given ground range. They also show the source height and ground-range combinations for which the air-over-ground neutron and gamma-ray tissue doses are greater than those at equivalent ranges in infinite air.

1. Abstract of ORNL-TM-4841 (August 1975).
2. Research sponsored by the Defense Nuclear Agency.
3. Computer Sciences Division, UCC Nuclear Division.

94. PRELIMINARY CROSS-SECTION SENSITIVITY ANALYSIS FOR AN AIR-OVER-GROUND ENVIRONMENT[1,2]

J. V. Pace III[3] D. E. Bartine

Two-dimensional sensitivity calculations[4] have been made for an air-over-ground geometry to determine the effect of air and ground cross-section perturbations on the total neutron and gamma-ray dose near the air-ground interface and 415 m above the ground. The cross sections used in all computations were ENDF/B-IV cross sections in 22 neutron groups and 18 gamma-ray groups.

Two-dimensional air-over-ground transport calculations were first performed with the DOT code,[5] using cylindrical geometry. The cylinder had an upper region of air 1300 m high and a lower region of ground 50 cm deep. The maximum horizontal range was 1500 m, and the source height was 50 m. A weapon fission source spectrum was input as a point source in the forward DOT run. For the two adjoint runs, the adjoint source, consisting of the Auxier-Snyder neutron dose factors[6] and the Claiborne-Trubey gamma dose factors,[7] was input at heights of 0.5 and 415 m and at ground ranges of 605 and 485 m, respectively, thus keeping slant range approximately constant. The VIP code[8] provided the integrals of the adjoint and forward fluxes over angle to the SWANLAKE code.[9]

which calculated the sensitivity of the total neutron and gamma-ray dose to changes in the cross sections as functions of position, energy, and element.

For the two detector positions, the calculations showed that a P_1 expansion of the cross section was adequate, agreeing to within 1% of that calculated with a P_3 expansion. The neutron doses were 1.8×10^{-20} and 4.0×10^{-20} rads tissue per source neutron for the detector heights 0.5 and 415 m respectively. The corresponding gamma-ray doses were 1.0×10^{-21} and 2.5×10^{-21} rads tissue per source neutron. Therefore, for this set of problems, the ground acted as an overall absorber. Of the total sensitivity (percent change in the response per percent change in the cross section) of −2.62 and −2.24 for the detector heights 0.5 and 415 m, respectively, the nitrogen sensitivities are −2.12 and −1.86. Nitrogen cross sections in the energy range 2.35 to 0.00335 MeV accounted for 60% of the total sensitivity for the two cases.

Table 94.1 shows the percent of the gamma-ray dose produced by thermal (n,γ) capture. For both detector positions, the nitrogen and silicon thermal (n,γ) captures combined to contribute more than 50% of the total gamma-ray dose. Altogether, thermal (n,γ) capture contributes between 73 and 80% of the gamma-ray dose.

Table 94.1. Percent of gamma-ray dose produced by thermal (n,γ) capture as a function of element

Element	Percent of dose	
	Detector height = 0.5 m, ground range = 605 m	Detector height = 415 m, ground range = 485 m
N (air)	25.8	29.1
O (air)	~0	~0
H (ground)	14.5	8.6
O (ground)	~0	~0
Si (ground)	26.6	22.5
Al (ground)	13.9	12.6
Total	80.8	72.8

The constituent elements for the ground in this calculation used a water content of 8.6% by weight. Linear perturbation theory predicted that a 1% change in the percent by weight of water would decrease the total dose on the ground by 2.3%, and 415 m above the ground by 0.8%. The dominant element is hydrogen, which accounts for 99% of the reduction.

These calculations show that the total dose is very sensitive to nitrogen neutron cross sections for detectors both near and far above the ground. For the gamma-ray dose, the nitrogen and silicon thermal (n,γ) reactions contribute over 50% to the total gamma-ray dose. Hydrogen accounts for 99% of the reduction in the dose when water content is increased in the ground. Finally, a P_1 expansion of the cross sections is sufficient to calculate the dose to within 1% of that calculated with a P_3 expansion.

1. Summary of paper presented at the 1975 Winter Meeting of the American Nuclear Society, San Francisco, California, November 16–21, 1975.

2. Research sponsored by Defense Nuclear Agency.

3. Computer Sciences Division, UCC Nuclear Division.

4. D. E. Bartine et al., "ORNL Cross Section Sensitivity Analysis Applications for Radiation Shielding," p. 551 in *Advanced Reactors: Physics, Design, and Economics*, Proceedings of the International Conference, Atlanta, Georgia, September 8–11, 1974, Pergamon, New York, 1975.

5. W. A. Rhoades and F. R. Mynatt, *The DOT III Two-Dimensional Discrete Ordinates Transport Code*, ORNL-TM-4280 (1973).

6. National Council on Radiation Protection and Measurements. *Protection Against Neutron Radiation*, NCRP-38. Washington, D.C., 1971.

7. H. C. Claiborne and D. K. Trubey, "Dose Rates in a Slab Phantom from Monoenergetic Gamma Rays," *Nucl. Appl. Tech.* 8, 450 (1970).

8. R. L. Childs, *VIP, A Computer Program Using Two-Dimensional Discrete Ordinates Transport Calculations for Cross-Section Sensitivity Analysis*, UCCND-CSD-1 (to be published).

9. D. E. Bartine, F. R. Mynatt, and E. M. Oblow, *SWANLAKE, a Computer Code Utilizing ANISN Transport Calculations for Sensitivity Analysis*, ORNL-TM-3809 (1973).

95. SUBCRITICALITY CALCULATIONS FOR THE FFTF REVERSE APPROACH TO CRITICAL EXPERIMENTS[1,2]

D. L. Selby[3] G. F. Flanagan

Reverse approach to critical (RAC) experiments were performed in the ZPR-IX critical facility at Argonne National Laboratory. One of the major objectives of this project is to determine the adequacy of the low-level flux monitor (LLFM) detectors for initial loading of the Fast Flux Test Facility (FFTF). This objective can be achieved if a systematic approach can be found to relate the reactivity of the system to the available detector count rates. The method used most often to relate the reactivity and the count rate at some specific detector location is the modified source multiplication method. The basic modified source multiplication equation is[4]

$$\$_i = \$_0 \frac{S_i W_i \nu_0 CR_0 \beta_0}{S_0 W_0 \nu_i CR_i \beta_i} ,$$

where

$\$$ = reactivity in dollars.
S = effective source.
W = detector efficiency.
ν = number of neutrons produced per fission.
CR = detector count rate.
β = delayed neutron fraction.

and the subscripts 0 and i refer to the reference configuration and ith configuration respectively. The difficulty lies in calculating the detector efficiency.

As in earlier work,[4] the detector efficiencies were calculated with the two-dimensional discrete-ordinates transport theory code DOT III,[5] using an S_4P_3 expansion. The ENDF/B-III cross-section file was processed with a modified version of the AMPX[6] system available at ORNL to produce the 50-group cross-section set used in these calculations. The Nordheim method was used to weight the unresolved resonance regions, and a narrow resonance treatment with an energy-dependent potential cross section was used to weight the resolved resonance regions. Heterogeneity effects were accounted for in the core by cell weighting the core cross sections. Third-dimensional leakages were accounted for by the use of zone-dependent, energy-dependent bucklings calculated using the three-dimensional Monte Carlo code KENO.[7]

Comparison of initial calculations with experiments indicated that the results of the three LLFM detectors were in random disagreement. Similar results had been observed in the earlier analysis of the Experimental Mockup Critical project,[4] where it was found that a convergence problem in the shield contributed to errors in the detector efficiency at the LLFM detector locations.

However, upon attempting a tighter convergence by increasing the number of iterations, it was found that no change in detector efficiencies occurred for the RAC calculations. Thus it was concluded that a convergence problem in the shield did not exist.

After intense scrutiny, it was determined that problems existed in the buckling corrections. When the initial calculations for the RAC project were done, changes in the buckling correction factors due to asymmetric removal of fuel assemblies were not considered. However, after the poor results of the initial calculations, it was determined that streaming in the zones where fuel had been replaced by sodium might cause substantial changes in the buckling correction factors. Thus, using KENO, buckling correction factors were found for each step of the experiment. Large changes in the buckling correction factors did occur, and the resulting reactivities inferred were in much better agreement. Table 95.1 gives the results for six steps of the experiment at four detector locations before and after changes in the buckling correction factors were made as well as the percent change due to these buckling changes.

Although the agreement between the LLFM's was greatly improved, there were still two important instances where the agreement of the LLFM detectors was not within the required margins. However, it was determined that the in-core detector results were identical to results found from an independent noise analysis. Thus it was recommended that an in-core detector be used for the startup of the FTR.[1]

Table 95.1. Initial and final reactivities inferred by detectors for six selected cases

Detector	Case	Reactivity ($) Initial	Reactivity ($) Final	Percent change
In-core	B.2	30.52	30.52	0.0
	C.3	1.93	1.94	0.41
	D.1	46.12	40.49	12.21
	E.9	15.09	13.01	13.78
	F.1	99.17	78.11	21.24
	G.3	219.42	177.30	19.20
LLFM No. 1	B.2	28.79	28.79	0.0
	C.3	1.72	1.93	12.28
	D.1	38.44	41.14	7.02
	E.9	15.86	13.10	17.38
	F.1	87.35	74.44	14.89
	G.3	184.79	153.69	16.83
LLFM No. 2	B.2	28.35	28.35	0.0
	C.3	1.81	2.06	13.85
	D.1	42.81	41.15	3.88
	E.9	14.12	13.07	7.44
	F.1	83.11	78.39	5.68
	G.3	189.61	163.07	14.00
LLFM No. 3	B.2	28.03	28.03	0.0
	C.3	1.77	2.21	24.51
	D.1	41.11	39.65	3.55
	E.9	13.01	11.58	10.98
	F.1	78.96	77.54	1.80
	G.3	176.88	142.80	19.27

1 Summary of paper presented at the 1975 Winter Meeting of the American Nuclear Society, San Francisco, California, November 16-21, 1975.

2 Research sponsored by ERDA's Division of Reactor Research and Development.

3 Computer Sciences Division, UCC Nuclear Division.

4 D. L. Selby and G. F. Flanagan, *Subcriticality Calculations in Support of the Reactivity Surveillance Procedures Experiments for the FTR Engineering Mockup Facility*, ORNL-5004 (to be published).

5 W. A. Rhoades and F. R. Mynatt, *The DOT III Two-Dimensional Discrete Ordinates Transport Code*, ORNL-TM-4280 (September 1973).

6 N. M. Greene, J. L. Lucius, L. M. Petrie, W. E. Ford III, J. E. White, and R. Q. Wright, *AMPX: A Modular Code System for Generating Coupled Multigroup Neutron-Gamma Libraries from ENDF/B*, ORNL-TM-3706 (to be published).

7 L. M. Petrie and N. F. Cross, *KENO IV: An Improved Monte Carlo Criticality Program*, ORNL-4983 (to be published).

8 J. T. Mihalczo, G. C. [illegible], and D. L. Selby, "Evaluation of Initial Loading Count Rate Data for the FFTF Critical Experiments with a Mockup Core," *Trans. Am. Nucl. Soc.* 21, 436 (1975).

96. USE OF VARIATIONAL TECHNIQUES FOR THE ESTIMATION OF NEUTRON DETECTION EFFICIENCY[1,2]

S. C. Lin[3] J. C. Robinson[3] G. F. Flanagan

In this paper, approximate techniques for the computation of the neutron detection efficiency, W_n, are presented. This parameter, defined by[4]

$$W_n = \frac{\langle \Sigma_d \phi \rangle}{\langle \Sigma_f \phi \rangle} \tag{1}$$

where Σ_d and Σ_f are macroscopic detector and fission cross sections and $\langle\ \rangle$ represent integration over phase space, is required for the application of the modified source multiplication (MSM) technique for subcriticality determination.[4-8] The necessity of the approximate techniques lies in the excessive computation time required to obtain the neutron flux ϕ in phase space at the many different states at which W_n is required (i.e., at each state at which the MSM technique is applied).

The approximate methods employed are based on variational methods. The particular variational method used in this study is based on a functional $E(\tilde{\phi})$, which is the ratio of two other functionals:

$$F(\phi) = \frac{\langle \Sigma_d \phi \rangle + \langle \phi_1^*, S - H\phi \rangle}{\langle \Sigma_f \phi \rangle + \langle \phi_2^*, S - H\phi \rangle} \tag{2}$$

The stationary value of $E(\tilde{\phi})$ is the detection efficiency W_n of Eq. (1). Treating the numerator and denominator of Eq. (2) separately as functionals leads to three separate Euler equations: a forward equation with S as the source, an adjoint (importance) equation with Σ_d as the source, and an adjoint equation with Σ_f as the source. The desirable feature of this formulation over that presented by Pomraning and Stacey[9] is that the solution to the forward equation is not required for the construction of the source term to the adjoint equation. However, it will be shown that the functional used here is equivalent to that presented in the literature.

Table 96.1. Detection Efficiency (DE) calculated by variational methods for fission chamber[a]

Variational interpolation			Conventional variational			Multireference-state variational		
Perturbation state	DE[b]	Error (%)[c]	Perturbation state	DE	Error (%)	Perturbation state	DE[b]	Error (%)
$(2,5^d)\to 3^d$	5.33	0.75	$(5,5^d)\to 2$	5.54	−0.61	$A\to 6^e$	5.[illegible]	−0.[illegible]
$(2,11^d)\to 4$	5.95	4.[illegible]	$(5,5^d)\to 3$	5.45	−2.52	$A\to 7$	6.[illegible]	−1.6
$(2,11^d)\to 8$	5.28	1.75	$(5,5^d)\to 4$	5.9[illegible]	1.10	$A\to 9$	6.14	+0.24
$(8,5^d)\to 7$	6.23	2.03	$(5,5^d)\to 6$	5.[illegible]	−1.18	$B\to 6^f$	5.[illegible]	+1.55
$(8,11^d)\to 9$	6.11	0.2[illegible]	$(11,11^d)\to 9$	6.17	−0.76	$B\to 7$	6.39	−0.49
$(5,11^d)\to 6$	5.73	1.02	$(11,11^d)\to 10$	6.64	0.11	$B\to 9$	6.13	−0.16
$(11,5^d)\to 6$	5.65	2.34	$(11,11^d)\to 12$	6.33	−0.23	$B\to 4$	6.07	−0.78

[a] The subcriticalities for state 2 to state 12 are ranging from \$2.71 to \$29.96 in order.
[b] At 107 cm from the core center.
[c] Error (%) = [(exact value − estimated value)/exact value] × 100.
[d] δ at state 2, ϕ_1 and ϕ_2 at state 5, perturbation state at 3.
[e] $A = (2 + 8, 5^d + 11^d)$, forward fluxes from states 2 and 8, adjoint fluxes from states 5 and 11.
[f] $B = (5 + 11, 5^d + 11^d)$.

To demonstrate the flexibility of the functional proposed here, the efficiency (W_c) has been estimated [solution to Eq. (2)] using three different techniques (each based on reference state choices):

1. Variational interpolational method.[7] Two reference states are chosen such that the state of interest (perturbed state) is bracketed. The forward flux is computed for one of the reference states, and the two adjoint equations are solved for the second reference state.
2. Conventional variational method.[10] The forward and adjoint equations are all solved at the same reference state.
3. Multireference-state variational method.[9,11] The reference forward and adjoint fluxes (trial functions) are linear combinations of several reference states.

Results obtained for a one-dimensional calculational model of a fast test reactor for each of the three methods are presented in Table 96.1. The results from all methods are certainly acceptable, but the multireference state leads to the more accurate results.

Results obtained for a two-dimensional model will be presented. From this analysis (results of one- and two-dimensional problems) it is concluded that acceptable estimates of W_c can be obtained using the multireference-state variational method based on the functional presented in Eq. (2).

1. Summary of paper presented at the 1975 Winter Meeting of the American Nuclear Society, San Francisco, California, November 16–21, 1975.

2. Research sponsored by ERDA's Division of Reactor Research and Development.

3. University of Tennessee.

4. R. C. Kryter, N. J. Ackermann, Jr., and A. R. Buhl, *Subcriticality Measurements in FTR-3: The Homogeneous Mockup of the Fast Test Reactor*, ORNL-TM-3715 (February 1972).

5. G. F. Flanagan, D. B. Simpson, and A. R. Buhl, *One-Dimensional Calculations in Support of the Reactivity Surveillance Procedures for the Fast Test Reactor*, ORNL-TM-4014 (April 1973).

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97. RELATION BETWEEN BASIC EXPERIMENTS AND REACTOR PERFORMANCE PARAMETERS (A Study Related to CRBR)[1,2]

G. F. Flanagan M. Salvatores[3] Y. Hsueh[3]

When assessing the sensitivity of various performance parameters to uncertainties in cross-section data, a considerable reduction in effort could be attained if integral experiment sensitivity results could be directly applied to the design. This study is an attempt to exploit experimental information on basic data for reactors in order to gain insight on the assessment of design parameters of a fast reactor.

In particular, we considered the breeding ratio of the Clinch River Breeder Reactor (CRBR), since the ability to make reliable predictions on this parameter is definitely of high interest to the designer, but direct integral measurements of this parameter are not possible. However, there exists information from integral experiments which could be used to make the predictions of the CRBR breeding ratio more reliable.

The study was completed using the following three-step procedure:

1. Define a model for both the CRBR and ZPR-6-7, a large experimental critical assembly fueled with PuO_2 and UO_2. Using a detailed energy group structure for a multigroup calculation, obtain a sensitivity profile of the breeding ratio for the CRBR.
2. Verify that the breeding ratio (BR) and, in particular, the internal breeding ratio (IBR) are strictly related to the ratio of the ^{238}U capture rate to the ^{239}Pu fission rate (C^{28}/f^{49}) at the core center. This relation is to be bound in terms of the sensitivity profiles for BR, IBR, and C^{28}/f^{49}.
3. Finally, verify that the sensitivity profile of C^{28}/f^{49} in the CRBR is similar to the corresponding profile in ZPR-6-7, where careful measurements of C^{28}/f^{49} were made.

From step 3, one should be able to relate the uncertainty in the breeding ratio to the discrepancy between calculational and experimental values of an integral quantity that was found to be "sensitive" to cross-section changes very much in the same way as the CRBR breeding ratio itself.

In this study, a two-dimensional (R-Z) model of the CRBR was used to calculate the breeding ratio sensitivities with a 51-group cross-section set, using the Italian two-dimensional sensitivity codes.[4-7] These profiles are compared with the central C^{28}/f^{49} profiles for the CRBR in Fig. 97.1.

In Fig. 97.2, the C^{28}/f^{49} CRBR sensitivity profiles are then compared with those obtained from a one-dimensional spherical model for ZPR-6-7, using the 51-group cross-section set and the one-dimensional Italian sensitivity code.

As can be seen in Fig. 97.1, the profiles for BR, IBR, and C^{28}/f^{49} are remarkably similar to the main discrepancies at low energies, where the BR shows an increased sensitivity to the blanket contribution to all low-energy effects; this trend, however, is quite limited.

In Fig. 97.2, the agreement is again remarkable, with ZPR-6-7 values being lower at higher energies, due to the diluted nature of this assembly relative to the CRBR.

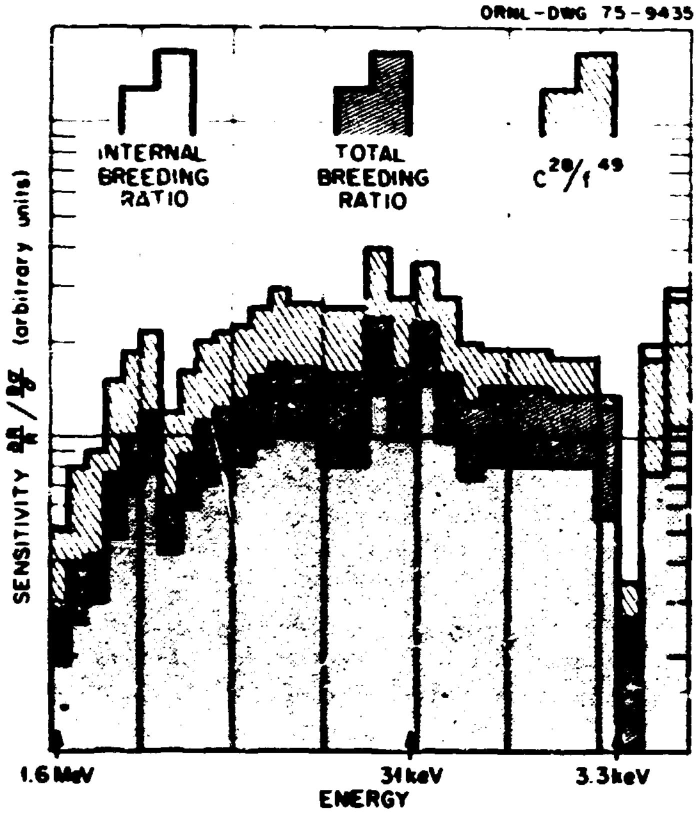

Fig. 97.1. Sensitivity of CRBR internal breeding ratio total breeding ratio, and ratio of ^{238}U capture rate to ^{239}Pu fission rate (C^{28}/f^{49}) to changes in the ^{238}U capture cross sections.

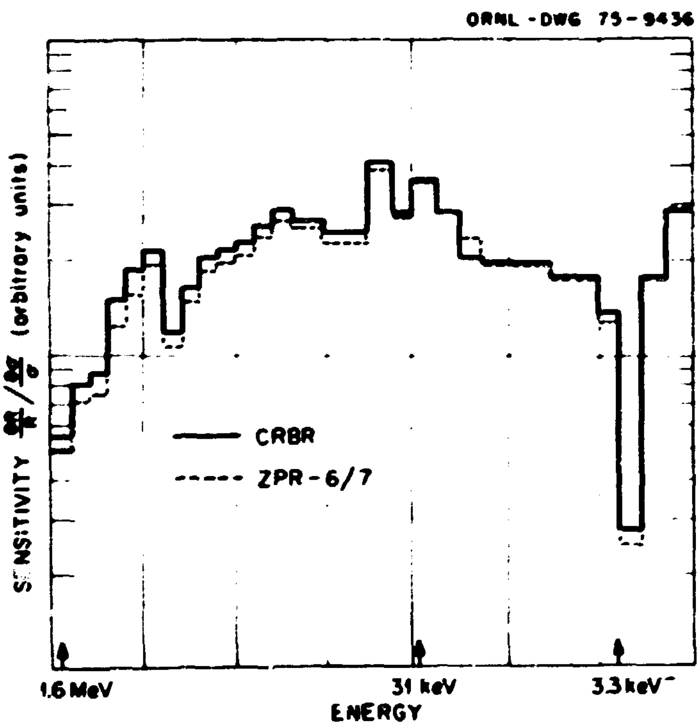

Fig. 97.2. Comparison of C^{28}/f^{49} sensitivity profiles for the CRBR and ZPR-6/7.

Using the results from this study, one can then use the profiles for the ZPR-6-7, C^{28} f^{49}, folded with the cross-section uncertainty data, along with the corresponding calculated to experimental values of C^{28} f^{49}, to obtain an improved uncertainty for the CRBR breeding ratio. The technique can be applied to the design of any large LMFBR core that is found to be similar to ZPR-6-7. Fur her studies are under way for various material worths.

1. Summary of paper presented at the 1975 Winter Meeting of the American Nuclear Society, San Francisco, California, November 16–21, 1975.

2. Research sponsored by ERDA's Division of Reactor Research and Development.

3. Georgia Institute of Technology, School of Nuclear Engineering, Atlanta.

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98. INVESTIGATION OF ^{238}U INTERMEDIATE STRUCTURE IMPORTANCE[1,2]

T. J. Burns C. R. Weisbin

The impact of recently measured cross-section structure in the unresolved resonance region of ^{238}U on two fast-reactor parameters has been evaluated. The measured structure was represented by a set of unresolved resonance parameters given at 171 energies[3] (as opposed to the 15 energy points utilized in ENDF B-IV). The new set of resonance parameters was normalized to give the ENDF B-IV average cross section in order to isolate the effect of the intermediate structure. Infinite-dilution capture cross-section data and Bondarenko factors were generated for both the new set of resonance parameters and the ENDF B-IV parameters, using the MINX[4,5] processing code. The resulting infinitely dilute cross-section data are illustrated in Fig. 98.1. Multigroup cross sections were then calculated in a 17-group subset of the ORNL standard DRRD neutron energy group structure,[6] which spanned the appropriate energy range 4–40 keV) for both data sets. Self-shielded multigroup cross sections were generated for three finite dilutions (σ_0 = 50, 10, and 1 b) in addition to the infinite-dilution case.

Each pair of multigroup cross sections was then combined with sensitivity profiles generated with the FORSS[7] code system for a one-dimensional model of the Clinch River Breeder Reactor[8] to evaluate the effects of the intermediate structure on the effective neutron multiplication factor and breeding ratio, using generalized linear perturbation theory. The results of these calculations are given in Table 98.1. The impact of the intermediate structure is seen to be an order of magnitude smaller than that attributable to the cross-section normalization. Calculations were also done utilizing multigroup cross sections generated using a macroscopic weighting function characterized by a large minimum due to the iron resonance at 28 keV. The macroscopic weighting function did produce small changes in the multigroup cross sections however, the effect of the intermediate structure was still very small.

The calculated effects of the intermediate structure are significantly less than similar effects resulting from processing code approximations such as the quadrature scheme used for evaluation of fluctuation integrals.[9] Further, the intermediate structure effect is at least an order of magnitude smaller than the effects of ^{238}U cross-section uncertainties that have been reported for related fast critical assemblies.[10,11] Although further refinements in the representation of the structure itself and the reactor model utilized (i.e., the sensitivity coefficients and macroscopic group constants) may modify the numerical values given in Table 98.1, it is anticipated that such refinements will not

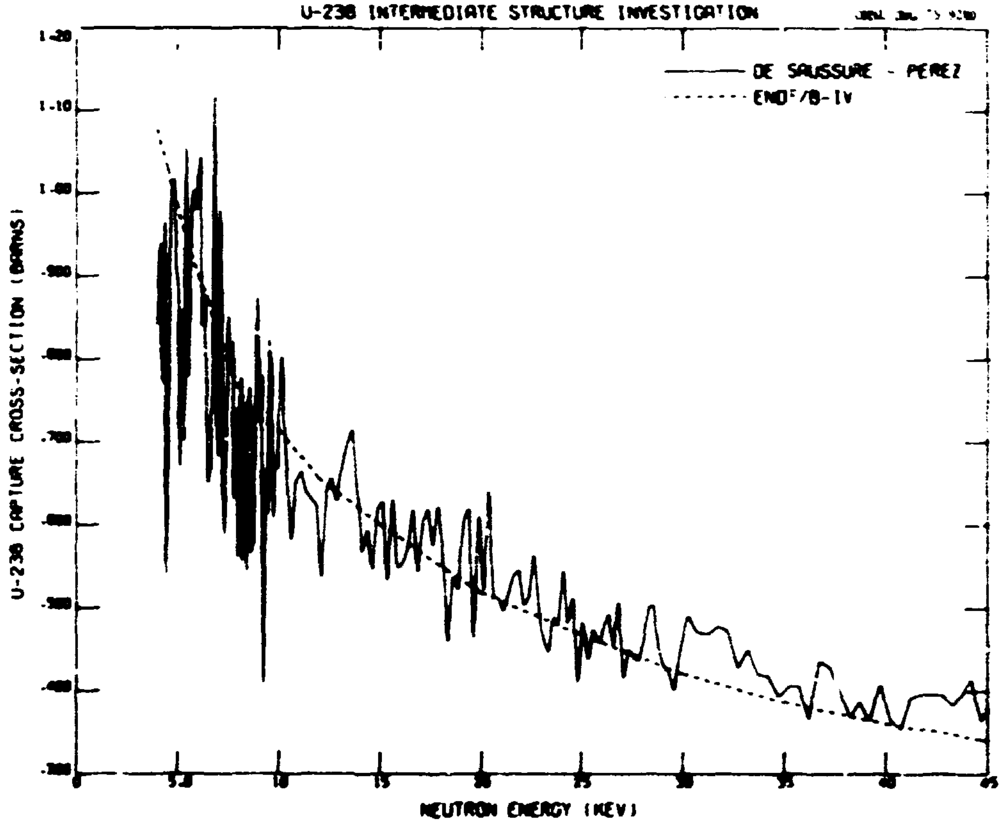

Fig. 98.1. ^{238}U infinitely dilute capture cross section.

Table 98.1. Fractional change in reactor parameters due to intermediate structure in ^{238}U capture cross section

Region	k_{eff}			Breeding ratio		
	Core	Blanket	Core + blanket	Core	Blanket	Core + blanket
$\sigma_0 = \infty$	+0.00017	+0.00000	+0.00018	0.00085	0.00027	0.00112
$\sigma_0 = 50$ b	+0.00001	0.00002	0.00001	0.00022	0.00008	0.00026
$\sigma_0 = 10$ b	0.00017	0.00004	0.00021	+0.00044	+0.00012	+0.00056
$\sigma_0 = 1$ b	0.00029	0.00005	0.00034	+0.00092	+0.00026	0.00118
Normalization (10%)	0.00395	0.00050	0.00445	+0.01478	+0.00435	+0.01914

materially affect the importance of the intermediate structure relative to uncertainties in differential data or numerical processing approximations.

1. Summary of paper presented at the 1975 Winter Meeting of the American Nuclear Society, San Francisco, California, November 16–21, 1975.

2. Research sponsored by ERDA's Division of Reactor Research and Development.

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99. REVIEW OF ORNL RADIATION SHIELDING ANALYSES OF THE FAST FLUX TEST FACILITY REACTOR[1,2]

Lorraine S. Abbott F. R. Mynatt

This report reviews the design support analyses performed at Oak Ridge National Laboratory to predict the neutron and secondary-gamma-ray fluxes and dose rates and the sodium-activation gamma-ray dose rates that can be expected at various locations in the Fast Flux Test Facility. The calculations were concentrated in areas of particular concern to the facility designers. The regions studied were: the stainless steel shield below the reactor core; the steel and sodium region above the core; the clearance gaps around massive plugs that rotate within the thermal shield and pressure vessel head above the sodium pool; the region between the side of the core and the pressure vessel wall, including a stored-fuel region; a large nitrogen-filled cavity surrounding the reactor pressure and guard vessels, in which a concrete shield was introduced, the vessel support system; the branch-arm pipe and center-island shielding in the head compartment; and the regions outside the reactor vessel through which the sodium coolant ducts lead to the primary heat exchangers.

1. Abstract of ORNL-5027 (July 1975).
2. Research sponsored by ERDA's Division of Reactor Research and Development.

100. ANALYSIS OF THE COMPLEX REACTOR CAVITY SHIELD IN THE FTR[1,2]

W. W. Engle, Jr. M. B. Emmett[3] M. L. Williams

Early in the ORNL design support shielding analysis effort for the Fast Flux Test Facility, the reactor cavity was identified as a significant streaming path that allowed neutrons to reach the vessel-support system and closure head area with a considerably harder spectrum than those neutrons that traverse the sodium pool. The importance of the reactor cavity was further enhanced with the discovery that the stored fuel arrays, which lie outside the radial shield, multiply the high-energy neutron flux by several orders of magnitude. In addition to design modifications to the vessel support system, a reactor cavity shield consisting of a B_4C collar and an annular concrete ring was included in the design by Westinghouse's Advanced Reactors Division (WARD).

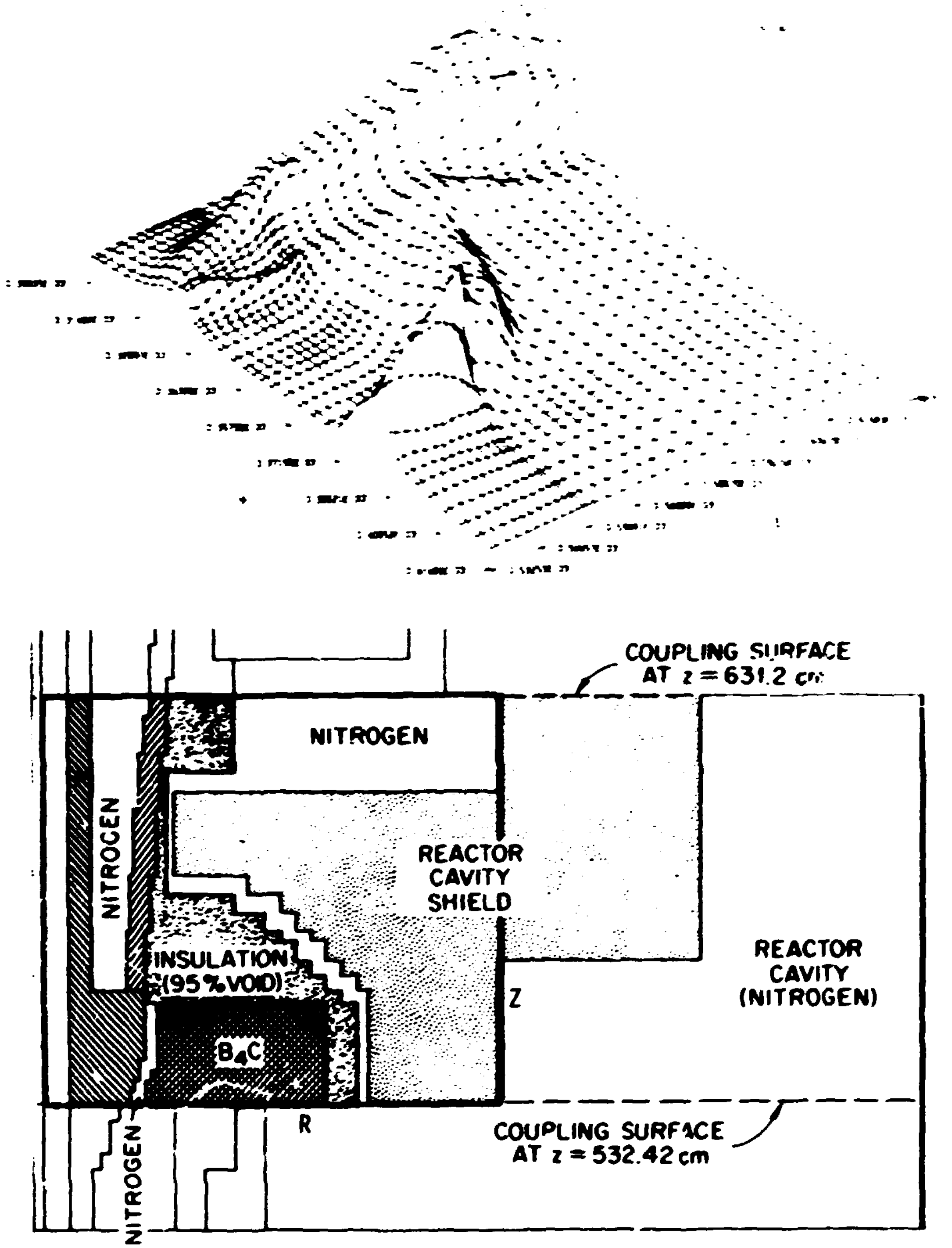

Fig. 100.1. Perspective plot showing streaming paths through reactor cavity shield region outlined by heavy box.

The analysis of this complex system required the modification and refinement of existing calculational techniques as well as the development of new methods for coupling two discrete-ordinates calculations with dissimilar mesh and quadrature, for coupling two-dimensional discrete-ordinates and three-dimensional Monte Carlo calculations, and for coupling adjoint discrete-ordinates calculations with forward calculations, using either discrete-ordinates or Monte Carlo techniques. Two-dimensional discrete-ordinates adjoint calculations were used extensively both to provide biasing parameters for Monte Carlo calculations and to efficiently analyze design changes when only a small segment of the geometry was affected.

Monte Carlo calculations, coupled to discrete-ordinates calculations as mentioned above, were required to analyze the three-dimensional geometries resulting from several cutouts and penetrations in the reactor cavity shield for piping and surveillance systems. For reasons of efficiency, the coupling between calculational techniques was two-fold: a forward discrete-ordinates calculation provided the source for the Monte Carlo calculation, and an adjoint discrete-ordinates calculation was used as a scoring function. The Monte Carlo calculation was thus confined only to that portion of the geometry which required a three-dimensional description.

Late in the analysis effort, the forward and adjoint flux moments from a complete two-dimensional calculation were folded and displayed using perspective plots. The result displayed in Fig. 100.1 shows the relative contribution to a desired response (defined by the source in the adjoint calculation) from every mesh cell in the problem. This technique, known as channel theory, appears to be a valuable tool to be used in understanding the important transport paths through a complex geometry.

As a result of the close cooperation between the shielding analysis effort at ORNL and the engineering design effort at Westinghouse ARD, the peak dose rates above the closure head as determined from the calculations have been reduced by a factor of approximately 10^4 over the original design and are nominally within a factor of 2 of the design constraints.

1. Summary of paper presented at the 1975 Winter Meeting of the American Nuclear Society, San Francisco, California, November 16-21, 1975.

2. Research sponsored by ERDA's Division of Reactor Research and Development.

3. Computer Sciences Division, UCC Nuclear Division.

101. ANALYSIS OF THE TSF FIRST-FISSION EXPERIMENT FOR THE FAST TEST REACTOR[1,2]

R. L. Childs[3]   F. R. Mynatt   Lorraine S. Abbott

The "first-fission" experiment performed at the ORNL Tower Shielding Facility in support of the design of the Fast Flux Test Facility was analyzed using discrete-ordinates method calculations. In the experiment the radial region from the core of the Fast Test Reactor to a stored fuel module was mocked up in slab geometry, and fission chamber responses at the stored-fuel location were interpreted as first fissions in the stored fuel. The purpose of the experiment was to verify the calculation of the first generation of fissions in the FTR stored-fuel array and thereby to investigate the most complex aspect of the stored-fuel source determination. The calculated fission chamber responses obtained in the analysis were consistently a factor of 2 lower than the measured responses. Calculated responses for BF_3 detectors and hydrogen counters at the same location were also lower than the measured responses by varying factors, indicating a spectral distortion. An investigation of cross-section grouping showed that the group structure did not greatly affect the calculated responses. Streaming between stainless steel rods mocking up the FTR radial shield was also discounted for an error of this magnitude. A complex combination of underprediction in the dominant channels for overall neutron transmission through the radial shield and an underprediction of the local moderation of near-thermal neutrons are now suspected.

1. Abstract of ORNL-TM-4960 (in press).
2. Research sponsored by ERDA's Division of Reactor Research and Development.
3. Computer Sciences Division, UCC Nuclear Division.

102. FAST REACTOR EXPERIMENTAL SHIELDING PROGRESS REPORT FOR 1974[1,2]

C. E. Clifford F. J. Muckenthaler P. N. Stevens[3]

This report describes work performed by the Neutron Physics Division in Fast Reactor Experimental Shielding Research during the month of March 1974. The Total Cross-Section Check Program was continued.[4] Background problems encountered in earlier measurements were resolved. Bonner ball and or NE-213 measurements to verify the total cross sections for iron, nickel, carbon, sodium, stainless steel, and chromium were completed, and the results are presented. Hydrogen counter measurements were also taken; however, unfolding problems still exist with the hydrogen counter data, and the results are not included in this report.

1. Abstract of ORNL-TM-4658 (August 1974).
2. Research sponsored by ERDA's Division of Reactor Research and Development.
3. The University of Tennessee.
4. See also paper 53.

103. FAST REACTOR EXPERIMENTAL SHIELDING PROGRESS REPORT FOR APRIL AND MAY 1974[1,2]

C. E. Clifford F. J. Muckenthaler P. N. Stevens[3]

This report describes the experimental results obtained for a simulated lower axial shield, designed and fabricated by Atomics International as a part of the LMFBR Demonstration Plant development program. The objective of the experiment was to provide results that could be used to evaluate the adequacy of calculational techniques for predicting neutron streaming through control rod penetrations in the lower axial shield and also for predicting neutron transport in the unperturbed lower axial shield. The neutron fluence and spectra emerging from five different shield configurations were measured. Also measured were the neutron fluence and spectra within special penetrations reaching to the center line of the configuration at a number of locations in a mockup of the complete lower axial shield and grid plates. The complete set of experimental results for the shield configurations provided by Atomics International are presented in this report.

1. Abstract of ORNL-TM-4794 (February 1975).
2. Research sponsored by ERDA's Division of Reactor Research and Development.
3. The University of Tennessee.

104. FISSION RATE DETERMINATION OF SIMULATED CRBR STORED FUEL FOR THE FAST-REACTOR EXPERIMENTAL SHIELDING PROGRAM[1,2]

C. E. Clifford   F. J. Muckenthaler   P. N. Stevens[3]

This report describes the experiments performed at the Tower Shielding Facility which simulated the neutron-induced fissions that occur in the CRBR stored fuel. This experiment was designed to provide experimentally determined fission rates that would be utilized to verify the theoretical models normally used to describe stored-fuel activity. Hydrogen counter, NE-213 spectrometer, and Bonner ball measurements were made of the neutron transmissions through the various experimental configurations. In this experiment, nearly all neutrons above 1 MeV were produced by fissions in the stored fuel. This was clearly demonstrated by including experimental configurations that contained "dummy fuel" in place of the ^{235}U fuel assemblies. This work was done during August and September 1974.

1. Abstract of ORNL-TM-4868 (April 1975). Note: The title of the published report incorrectly identifies this experiment with the FTR program.
2. Research sponsored by ERDA's Division of Reactor Research and Development.
3. The University of Tennessee.

105. ANALYSES OF THE TSF FIRST-FISSION AND EX-VESSEL EXPERIMENTS FOR THE CLINCH RIVER BREEDER REACTOR[1,2]

R. L. Childs[3]   F. R. Mynatt   Lorraine S. Abbott

One of the first experiments performed at the Tower Shielding Facility in support of the design of the Clinch River Breeder Reactor (CRBR) was a radial shield mockup in which the first-fission reaction at the stored-fuel position and the ex-vessel detector response were studied. In the first part, identified as the CRBR "first-fission" experiment, the radial region of the CRBR from the core to a stored-fuel module was mocked up in slab geometry, and fission chamber responses at the stored-fuel location were interpreted as first fissions in the stored fuel. In the second part, identified as the CRBR "ex-vessel" experiment, the configuration was extended to mock up the entire region from the core to a simulation of a low-level flux detector (fission chamber surrounded by graphite) located in the cavity outside the reactor vessel. This report describes the analyses of the experiments that were performed with use of the discrete-ordinates transport method.

For both experiments the ratios of the calculated-to-measured fission chamber responses (C/E) were less than unity, with the ratios for bare counters being smaller than those for cadmium-covered counters. In the first-fission experiment the C/E ratio for the bare ^{235}U fission chamber was about 0.47, whereas for the cadmium-covered chamber it was about 0.63. When the bare chamber was surrounded by graphite in the ex-vessel experiment, the ratio improved to 0.65. The ratios for points inside the configuration were better — 0.72 and 0.83 for the bare chamber in the Inconel region and in the outermost steel region respectively. Corresponding ratios for the cadmium-covered counters were 0.82 and 0.94. These results indicate that in the calculations the thermal and near-thermal neutron flux is underpredicted and that the trend worsens with deeper penetration in the shield and then improves slightly in the presence of a moderator.

1. Abstract of ORNL-TM-5057 (in press).
2. Research sponsored by ERDA's Division of Reactor Research and Development.
3. Computer Sciences Division, UCC Nuclear Division.

106. RADIATION HEATING STUDIES IN IRON AND STAINLESS STEEL CRBR SHIELDS[1,2]

C. E. Clifford R. E. Maerker F. J. Muckenthaler

An experiment was conducted at ORNL which was designed to evaluate the experimental techniques for measuring heating in the heavy-metal shields typical to the LMFBR. The experimental techniques being evaluated consisted in lithium fluoride and calcium fluoride thermoluminescent detector (TLD) measurements and measurements using a high-pressure argon-krypton-filled ion chamber designed to duplicate the energy absorption properties of iron. The TLD results were obtained by four separate groups. The CaF_2 TLD measurements were made by the ORNL group, by T. Yule of Argonne National Laboratory, Chicago, and by G. Simons of Argonne National Laboratory, Idaho Falls; LiF TLD measurements were made by Simons and by a group at the Massachusetts Institute of Technology, headed by M. Driscoll. Intercomparison of the results from these groups and the various types of detectors, after suitable corrections are made to the TLD measurements, should give an indication of the overall accuracy of the measurements of the integral heating rates in the steel systems.

The shield configurations consisted of 5-ft-square slabs. A spectral modifier was placed in the neutron beam emanating from the Tower Shielding Reactor which modified the spectrum to resemble that which leaks from a fast reactor. The modifier was followed by an 8.7-in.-thick simulated blanket that consisted of UO_2-filled aluminum-clad pins contained in a stainless steel box. Following the blanket, two different mockups of the radial shield were studied. The first consisted of an array of nine 2-in.-thick mild steel slabs, followed by a single 1-in.-thick steel slab (19 in. total). Between each slab, a $^3/_8$-in. gap was maintained to allow insertion of the TLDs. The second shielding arrangement was constructed by replacing 11 in. of mild steel in the first arrangement with stainless steel slabs, so that it contained 11 in. of stainless steel interspersed with 8 in. of mild steel. Again, a $^3/_8$-in. void was maintained between slabs. Direct comparisons between ion chamber and ORNL TLD measurements were made in a number of positions within both of the shield configurations. This was accomplished by opening a gap at one position to $1^1/_4$ in. for insertion of the 1-in.-diam ion chamber. The ORNL TLD measurements were made simultaneously with each of the other TLD and ion chamber measurements, eliminating reactor power uncertainties in determining the relative response of the two detectors.

Data were obtained for 11 positions. The beginning of the steel was position 1, and the end of the steel was position 11. Other points were at 2-in. intervals in between. The results are shown in Table 106.1. The calculated results were obtained with DOT, using 51 neutron and 25 gamma-ray groups, and these calculations were also used to correct all the TLD measurements for neutron and gamma-ray spectral effects.

Table 106.1. Energy deposition in two CRBR steel shield configurations

Detector position	CaF_2-TLD results[a]			LiF-TLD results[a,b]		Ion chamber results, ORNL	Calculated results, ORNL
	ORNL (MeV/g·min/kW)	ANL-Chi/ORNL	ANL-IF/ORNL	MIT/ORNL-CaF_2	ANL-IF/ORNL-CaF_2	ORNL-CaF_2	ORNL-CaF_2
				Iron shield configuration			
1	4.02 (04)[c]	1.04	1.04	1.17	1.13	1.10	0.99
2	2.16 (04)	1.02	1.08	1.38	1.25		1.05
3	1.57 (04)	1.01	1.07		1.23		0.97
4	1.21 (04)	1.03	1.04	1.34	1.18	1.15	0.92
5	8.39 (03)	1.07	1.04		1.19		0.95
6	6.08 (03)	1.08	1.02		1.18		0.95
7	4.22 (03)	1.06	1.09		1.21	1.15	0.98
8	2.90 (03)	1.04	1.05	1.37	1.22		0.97
9	1.74 (03)	1.01					0.99
10	7.72 (02)	1.06		1.54			1.03
11	3.64 (02)	1.03				1.16	0.86
				Stainless steel shield configuration			
1	4.42 (04)	1.09		1.13	1.17		1.02
2	2.81 (04)	1.02		1.21	1.24		1.04
3	2.03 (04)	1.06			1.22		0.95
4	1.44 (04)	1.08		1.13	1.19	1.14	0.90
5	8.55 (03)	1.06			1.14		0.83
6	5.57 (03)	1.03					0.82
7	4.18 (03)	1.04				1.12	0.78
8	2.60 (03)	1.06		1.19			0.76
9	1.59 (03)	1.12					0.86
10	7.78 (02)	1.12		1.39			0.91
11	3.50 (02)	1.10				1.15	0.78

[a] All TLD results were corrected for neutron and gamma-ray-spectral effects to give the energy deposition in steel; measurements were usually taken at a power of ~50 kW and for a 2-hr exposure.

[b] All MIT results are still preliminary.

[c] Read: 4.02×10^4.

1. Summary of paper presented at the American Nuclear Society Meeting, New Orleans, Louisiana, June 8–13, 1975; *Trans. Am. Nucl. Soc.* 21, 533 (1975).

2. Research sponsored by ERDA's Division of Reactor Research and Development.

107. ABSOLUTE NEUTRON SPECTRUM EMERGING THROUGH THE BIG BEAM COLLIMATOR FROM THE TSR-II REACTOR AT THE TOWER SHIELDING FACILITY[1,2]

R. E. Maerker F. J. Muckenthaler

Neutron spectral intensities in a 51-group structure, together with the angular distributions in an 8-angle quadrature set, are presented which describe the surface leakage from the hemispherical pressure vessel surrounding the Tower Shielding Reactor II. These data may be used as free-field source terms in the calculations of experiments performed using the "big beam" collimator. They were derived from many different measurements made at various positions in the beam as well as from ANISN calculations of the pressure vessel leakage.

1. Abstract of ORNL-TM-5183 (in press).
2. Research sponsored by ERDA's Division of Reactor Research and Development.

108. CRITICAL EXPERIMENTS AND THE 2200-m/sec NEUTRON PARAMETERS[1,2]

R. Gwin

The use of critical volumes of aqueous homogeneous solutions of uranium in defining the 2200-m sec neutron parameters for ENDF B has been examined. The parameters for ^{233}U and ^{235}U are constrained by relating $(\bar{\eta} - 1)\bar{\sigma}_{ar}$ to the constant K obtained from the analysis of the critical systems; K is directly proportional to the hydrogen capture cross section at 2200 m sec. This paper suggests that the capture cross section of hydrogen be removed from K and that a new constant K σ_{aH} be defined by the critical systems. This new constant is the hydrogen-to-uranium ratio for an infinite critical system populated with neutrons having a Maxwellian energy distribution.

1. Abstract of ORNL-TM-4550 (January 1975).
2. Research sponsored by ERDA's Division of Reactor Research and Development.

109. DECAY HEAT ANALYSIS FOR AN LMFBR FUEL ASSEMBLY USING ENDF/B-IV DATA[1,2]

G. W. Morrison[3] C. R. Weisbin C. W. Kee[4]

This paper presents a practical application of recently evaluated ENDF B-IV fission product data in the form of decay heat calculations for typical LMFBR fuel assemblies exposed to 100,000-MWd metric ton burnup. The calculated results are being used in the design of a fuel storage and test facility for spent LMFBR fuel. A modified verision of the ORIGEN program was employed. Separate calculations were performed for the upper axial blanket, the core, and the lower axial blanket. The results were then blended by the program to obtain overall heat rates. One of the power histories considered was full-power operation for 180 days and shutdown for 60 days alternately for four periods and subsequent full-power operation for 120 days for a total of 1080 days. The decay heat and spectra were calculated for various times after discharge to estimate heat loads and specify shielding requirements. As an example, after 60 days following irradiation, the thermal $(\beta + \gamma)$ power per assembly is estimated to be ≅4 kW; at 120 days, the heat load is decreased to ≅3.5 kW per assembly. Detailed comparisons of the new ENDF B-IV and older ORIGEN decay libraries have been performed. Important contributors to the decay heat have been identified (e.g., at 90 days following irradiation, important contributors include zirconium, niobium, and promethium).

1. Abstract of paper presented at the Conference on Nuclear Cross Sections and Technology, Washington, D.C., March 3–7, 1975, and published in *Nuclear Cross Sections and Technology*, vol. I, ed. by R. A. Schrack and C. D. Bowman, NBS Special Publication 425, p. 455. (1975); *Bull. Am. Phys. Soc., Ser. II* 20(2), 154 (1975).
2. Division participation sponsored jointly by ERDA's Division of Waste Management and Transportation and Division of Reactor Research and Development.
3. Computer Sciences Division, UCC Nuclear Division.
4. Chemical Technology Division.

110. PROJECTED CRBRP SPENT FUEL CHARACTERISTICS AND THEIR IMPACT ON NDA TECHNIQUES[1]

G. W. Morrison[2] C. R. Weisbin C. W. Kee[3]

One of the major goals in the nuclear energy field is the development of a Liquid Metal Fast Breeder Reactor industry capable of fulfilling a major portion of this country's energy requirements in a safe and economical fashion. In order to achieve this goal, fuel reprocessing plants must operate as efficiently as possible; optimally small cooling times are clearly desirable to reduce fissile holdup and actual doubling times.[4] However, the situation is considerably more complex, since reduced cooling times lead to more "hostile" radiation environments with obvious manifestations in problems of nuclear safety and safeguards. For nuclear safeguards, direct measurement of the isotopic content of fissile material present in irradiated fuel requires a precise knowledge of the

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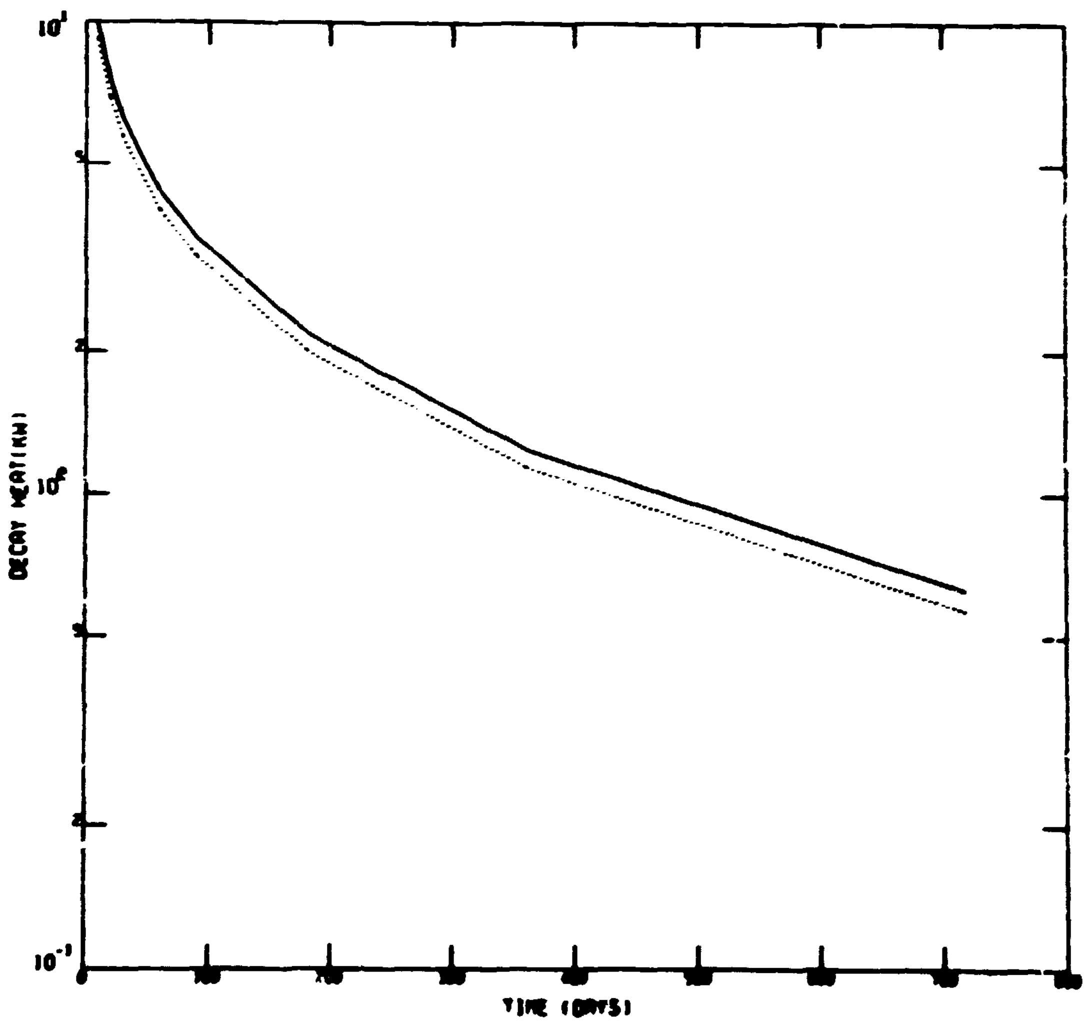

Fig. 110.1. CRBR fuel subassembly decay heat.

neutron and gamma-ray background. With the recent availability of extensive information describing fission product transmutation and decay,[5] a study has been initiated to project heat loadings from spent CRBRP fuel as well as the neutron and gamma-ray background anticipated at the head end of an LMFBR reprocessing plant.

A modified[6] version of the ORNL isotope generation and depletion code, ORIGEN,[7] has been employed for the calculation of nuclide composition, radioactivity, and thermal power for a prototypic CRBRP equilibrium core assembly exposed to high burnup (~150,000 MWd/metric ton). These results have already been transmitted for use in the design of a fuel storage and test facility for spent LMFBR fuel.[8]

The present calculations include decay data for 825 nuclides and yield information for more than 1000 nuclides from six different fissioning species. Energy-dependent capture cross sections are described for approximately 170 of the more important fission products. These cross sections have been processed by the MINX[9] program into 124 energy groups, assuming the fission products to be infinitely dilute. The ORIGEN program was then modified to derive group-collapsed reaction rates "on-line" with an appropriate input reference spectrum.[10] Such a capability also permits more credible analysis of foil and/or capsule measurements designed for dosimetry purposes.

Separate computations were performed for the upper axial blanket, the core, and the lower axial blanket.[11] The "on-line collapse" technique was used to generate spectral-weighted cross sections appropriate for each region. The results were then blended by the program to obtain the characteristics of the fuel subassembly. One of the power histories considered was full-power operation for 274 days following a shutdown of 91.25 days alternately for two cycles followed by a final cycle of full-power operation for 274 days.

An example of the output for decay heat per fuel subassembly obtained from the ORIGEN calculations is given in Fig. 110.1, where the decay heat for a typical CRBR fuel assembly irradiated to 100,000 MWd/metric ton is shown as a function of cooling time. Individual isotope edits as well as gamma-ray spectra are also available.

1. Summary of paper presented at the American Nuclear Society Meeting, New Orleans, Louisiana, June 8–13, 1975; *Trans. Am. Nucl. Soc.* 21, 487 (1975); division participation sponsored by ERDA's Division of Reactor Research and Development.

2. Computer Sciences Division, UCC Nuclear Division.

3. Chemical Technology Division.

4. D. B. Trauger, "On the Successful Commercialization of the LMFBR," Oak Ridge National Laboratory, June 1974.

5. ENDF/B-IV Fission Product Library Tapes 414–419 available from National Neutron Cross Section Center, Brookhaven National Laboratory, December 1974.

6. C. W. Kee, C. R. Weisbin, and R. E. Schenter, *Trans. Am. Nucl. Soc.* 19, 398 (1974).

7. M. J. Bell, *ORIGEN – The ORNL Isotope Generation and Depletion Code*, ORNL-4628 (1973).

8. D. W. Turner, Letter to M. Bender on FFTF and CRBR Heat Generation and Radiation, January 10, 1975.

9. C. R. Weisbin, P. D. Soran, R. E. MacFarlane, D. R. Harris, R. J. LaBauve, J. S. Hendricks, and J. F. White, "MINX, A Multigroup Interpretation of Nuclear Cross Sections from ENDF/B," Los Alamos Scientific Laboratory (to be published); see, also, *Trans. Am. Nucl. Soc.* 16, 127 (1973).

10. G. F. Flanagan, Oak Ridge National Laboratory, private communication, December 1974.

11. James A. Lake, Letter to M. W. Dyos on LMFBR Nuclear Analytical Model, LRA-73-251, August 17, 1974.

III. PRELIMINARY SHIELDING ANALYSIS OF A REFERENCE DESIGN 300-MW(e) GAS-COOLED FAST BREEDER REACTOR[1,2]

D. E. Bartine J. V. Pace III[3]
L. R. Williams F. R. Mynatt

The purpose of this preliminary investigation was to assess basic areas of concern in the development of a shielding program for a Gas-Cooled Fast Breeder Reactor conceptual design.[4] It is part of a comprehensive program of calculation and measurement being carried out cooperatively between the General Atomic Company (GAC) and ORNL to provide a sound technology for shielding for a 300-MW(e) GCFBR. GAC provided the two-dimensional cylindrical model used for shielding calculations, which is shown in Fig. III.1. The core is represented in zones 9 through 12, the axial and radial blankets are in zones 8 and 13, and the grid plate, from which the core is suspended, is in zone 19. Shielding is contained in the shaded zones: specifically 14 and 15, the inner and outer radial shields; 21 and 22, the upper axial shields; and 3, the lower axial shield. Special shielding to protect the grid plate is contained in zone 16, and zone 24 represents the concrete containment that extends radially beyond the iron liner (zone 1) in the form of the prestressed concrete reactor vessel (PCRV).

Transport calculations for the GAC two-dimensional shielding model were run with DOT[5] in r-z geometry, using the 50-group cross-section library employed at ORNL for FFTF studies. Initial calculations P_1 and S_6 were run, and an additional P_1 calculation of the lower outer radial shield "wrap-around" region was run with a biased quadrature set containing 306 angles designed to transport the flux below the outer shield and back up to the PCRV liner. Isoflux levels were plotted for energy ranges above 1 MeV, above 0.1 MeV, below 2.38 eV (thermal), and for the total flux. Isodose and isodamage plots were also obtained, and flux spectra were plotted at several positions of interest. A total flux isoplot on the right-hand side of Fig. III.1 indicates that the flux levels for this model range from 7×10^{14} to 7×10^{-6} neutrons cm^{-2}-sec.

The basic areas of concern identified in this preliminary study include axial rod streaming up to the stainless steel grid support plate, transport through the thorium oxide radial blanket and radial shield, streaming through the inlet and outlet coolant ducts, and upper shield plug streaming. Shielding optimization calculations are under way[6] to better meet flux constraints at the PCRV liner, and have been extended to include neutron and gamma-ray heating in the PCRV and in the lubricant for the tendons compressing the PCRV. The analytic techniques used in this preliminary study will be subject to experimental verification as part of the overall shielding design effort.

1. Summary of paper presented at the American Nuclear Society Meeting, New Orleans, Louisiana, June 8–13, 1975; *Trans. Am. Nucl. Soc.* 21, 525 (1975).

2. Research sponsored by ERDA's Division of Reactor Research and Development.

3. Computer Sciences Division, UCC Nuclear Division.

4. *300-MW(e) Gas-Cooled Fast Breeder Reactor Demonstration Plant*, GA-A13045, General Atomic Company (July 15, 1974).

5. F. R. Mynatt et al., *The DOT-III Two-Dimensional Discrete Ordinates Transport Code*, ORNL-TM-4280 (1973).

6. *Gas-Cooled Fast Breeder Reactor Quarterly Progress Report for the Period August 1, 1974 through October 31, 1974*, GA-A13228, General Atomic Company (December 12, 1974).

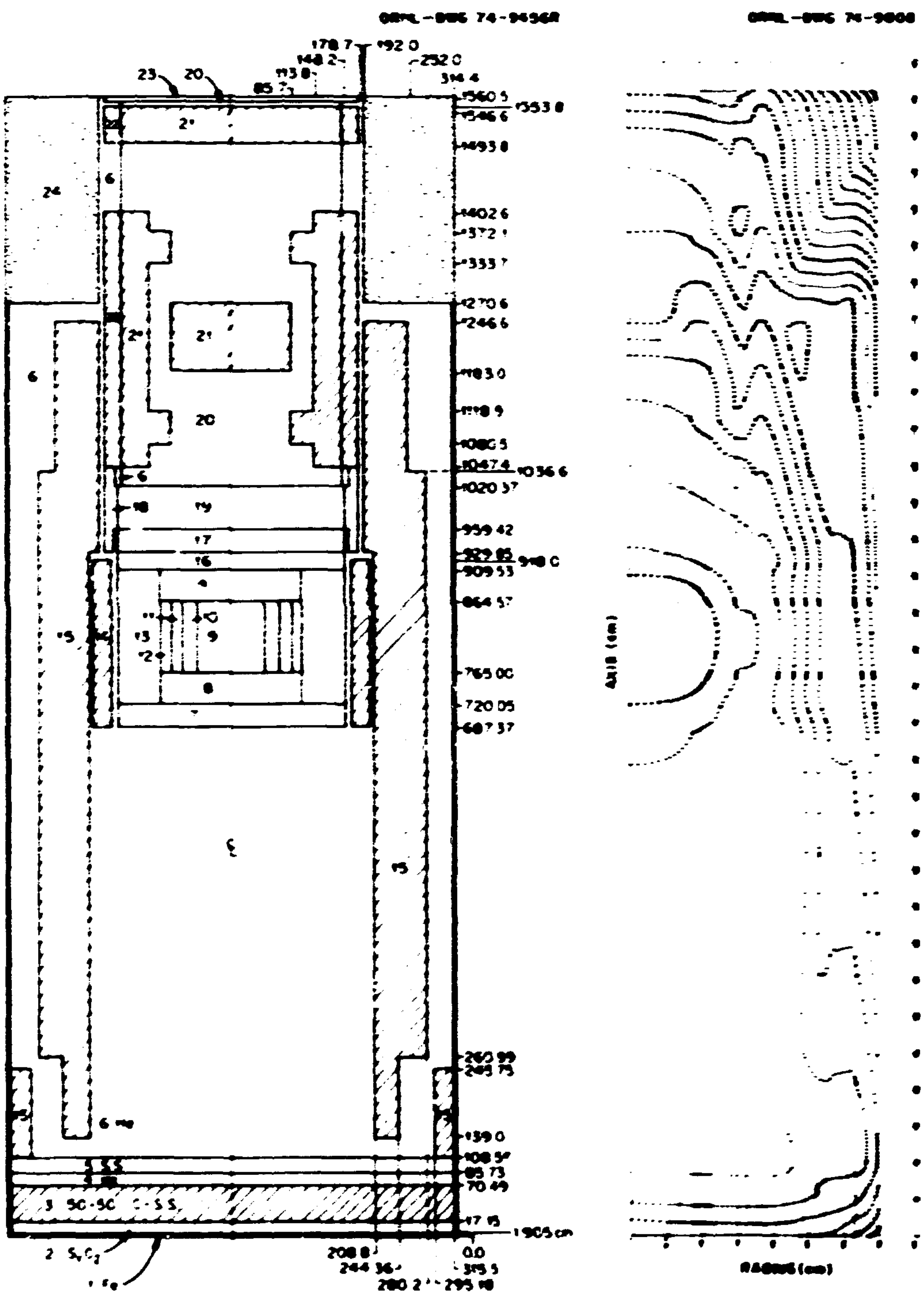

Fig. III.1. Shielding model (left) and total neutron flux isoplots (right) for 300-MW(e) GCFBR in C cylindrical geometry. Isoflux line at inner level is 7×10^{14} neutrons per square centimeter per second and other lines decrease by orders of magnitude out to 7×10^{4} neutrons per square centimeter per second.

112. SHIELDING CALCULATIONS FOR A 300-MW(e) GAS-COOLED FAST BREEDER REACTOR[1,2]

D. E. Bartine L. R. Williams
J. V. Pace III[3] J. R. Knight[3]

This paper is a presentation of calculations carried out to date at ORNL as part of a cooperative program with the General Atomic Company (GAC) to provide a sound technology for shielding for a 300-MW(e) Gas-Cooled Fast Breeder Reactor conceptual design.[4] The two-dimensional cylindrical model used for these calculations was the same as that shown in Fig. 111.1. The preliminary study reported in paper 111 identified basic areas of concern in the development of a shielding program for a GCFBR design and led to the calculations reported here dealing with the lower axial region.

All calculations of the system were performed in r-z geometry with DOT.[5] The preliminary study used a P_3 Legendre expansion of a 50-group cross-section set and had a homogeneous (50-50 graphite and stainless steel 304) representation of the inner and outer radial shields. The next step in the investigation was to perform a P_3 calculation with a 51-25 coupled neutron-gamma cross-section set, representing the inner and outer radial shields as laminated sections of graphite and stainless steel. The results given in Table 112.1 indicate that the initial radial shield configuration is conservative along the radial traverse (z = 0 plane) but has some problems in the lower axial region with thermal-neutron flux transmission to the PCRV liner and gamma-ray dose to the tendon lubricant. The constraints applied to GCFBR shielding design were originally adapted from HTGR

Table 112.1. A comparison of preliminary constraint radiation levels with two-dimensional calculated values for the heterogeneous reference model of the GCFR outer radial shield

Parameter and position	Preliminary constraint	Calculated results: Lower axial helium channel	Wraparound region	Peak above wraparound[a]	Radial traverse (z = 0 plane)
		Initial radial shield configuration			
Flux in PCRV liner (neutrons/cm²)					
(E > 1 MeV)	2 + 9[b]	2.4 + 7[c]	[illegible]1 + 7	5.3 + 7	5.6 + 5
(E < 2.38 eV)	1 + 10	3.7 + 10	3.3 + 10	7.8 + 10	3.0 + 8
Heating in PCRV (mW/cm³)	1 + 0	[illegible]	1.0 + 0	1.8 + 0	0.7 [illegible] 2
Heating in tendon lubricant (rads)	5 [illegible][d,e]	5.5 + [illegible][f]	7.7 + [illegible][f]	1.2 + [illegible][f]	1.0 [illegible][f]
		Revised radial shield configuration			
Flux in PCRV liner (neutrons/cm²)					
(E > 1 MeV)	2 + 9	9.9 + 7	[illegible].8 + 8	5.0 + 8	1.2 + 8
(E < 2.38 eV)	1 + 10	4.6 + 10	8.5 + 9	1.6 + 9	3.9 + 8
Heating in PCRV (mW/cm³)	1 + 0	0.9 + 0	0.3 + 0	0.3 + 0	0.2 + 0
Heating in tendon lubricant (rads)	5 [illegible][e]	7.1 + [illegible][f]	4.3 + [illegible][f]	2.4 + [illegible][f]	1.6 + [illegible][f]

[a] Occurs just above wraparound region for original design, just below inner shield for revised configuration.
[b] Read as 2 × 10^9. Constraint originally stated as 2 × 10^18 neutrons/cm² over plant lifetime (40 years at 80% operation).
[c] Flux levels normalized to a total flux of [illegible] × 10^13 neutrons.
[d] Constraint originally stated as 5 × 10^9 rads over plant lifetime.
[e] Rads in grease.
[f] Rads in concrete.

(High-Temperature Gas-Cooled Reactor) criteria and are still evolving to fit GCFBR requirements. Thus the constraint values presented in Table 112.1 should not be viewed as firm constraints but as radiation levels currently deemed desirable. It then appeared that with the addition of B_4C as a thermal poison and proper shield redesign, a reduction in the radial dimension of the reactor cavity might be possible which would significantly reduce capital costs. A study of potential revised radial shield designs was then carried out by GAC.[6] and a two-dimensional calculation was performed at ORNL for an initial revised configuration in which 1 ft was removed from the outer radial shield. The revised outer shield consisted of an 18-cm-thick graphite layer followed by an 18-cm-thick graphite-B_4C mixture (~19% B_4C by weight) and a 5-cm-thick stainless steel 304 region. The results of this calculation, also included in Table 112.1, show that the first redesign effort was reasonably successful in reducing the radial cavity, although some problem still exists with gamma-ray heating at the tendon lubricant and with thermal-neutron flux levels at the PCRV liner in the lower axial helium gap where the radial shield design is not the dominant factor.

In conclusion, two-dimensional calculations for an initial radial shield design and for a revised design in which the radius of the reactor cavity was reduced by a foot have been completed. Results indicate success in meeting current design constraints for high-energy flux and PCRV heating, but there are still some difficulties with the gamma-ray dose at the tendon lubricant and with the thermal-neutron flux level at the PCRV liner in the lower axial helium gap. Current design efforts are therefore focused on these areas.

1. Summary of paper presented at the 1975 Winter Meeting of the American Nuclear Society, San Francisco, California, November 16–21, 1975.

2. Research sponsored by ERDA's Division of Reactor Research and Development

3. Computer Sciences Division, UCC Nuclear Division.

4. *300-MW(e) Gas-Cooled Fast Breeder Reactor Demonstration Plant*, GA-A13045, General Atomic Company (July 15, 1974).

5. F. R. Mynatt et al., *The DOT-III Two-Dimensional Discrete Ordinates Transport Code*, ORNL-TM-4280 (1973).

6. M. Nagel, C. A. Rouse, A. Razani, and R. J. Cerbone, "Radial Shield Design Considerations for a 300-MW(e) Gas-Cooled Fast Breeder Reactor," *Trans. Am. Nucl. Soc.* 21, 526 (1975).

113. SURFACE PHYSICS AND TECHNOLOGY PROGRAM AT ORNL—FIRST-WALL IMPURITIES IN ORMAK[1,2]

R. J. Colchin[3] S. N. Cramer
R. E. Clausing[3] E. M. Oblow

Surface physics work in the Controlled Thermonuclear Reactor Program at ORNL is aimed both at understanding wall effects in present-day experiments and at designing low-contaminant walls for next generation devices. Experiments in progress are concerned with ORMAK wall impurities and with the honeycomb wall concept.

All present-day Tokamaks suffer from the effects of plasma impurities. These impurities get into the plasma via plasma interactions with surrounding walls and limiters. Although a fraction is attributable to sputtering and limiter evaporation, the largest source of impurities in the ORMAK arises from desorption of contaminants on vacuum walls. Carbon and oxygen are the principal impurities, as determined by uv spectroscopy. Each contributes 1 to 2% to the plasma density. Even such low-Z (effective nuclear charge) materials severely limit present experiments. This will remain true in the future, since experiments are always pushed to their limits.

The carbon and oxygen found in the ORMAK probably originate from layers of material adsorbed from the atmosphere. As in other Tokamaks, a period of vacuum cleanup is necessary after exposure to air before discharges will run reliably. Cleanup procedures typically include baking and discharge cleaning. The ORMAK usually experiences an exponential cleanup with time, with several days being necessary before high-current discharges can be run.

Auger spectroscopy of gold surfaces similar to ORMAKs has shown that carbon (probably hydrocarbons) and oxygen, most likely in the form of water, are adsorbed on exposure to the atmosphere. The extent of this coating depends on time, and surface coverages as high as 0.8 (atom fraction) have been found. This illustrates the general theorem that it is not possible to build a Tokamak with contaminant-free surfaces, and so cleanup procedures must be found. Laboratory tests have shown that discharge cleaning in hydrogen purges some impurities but that oxygen is more effective, particularly for carbon. The ORMAK plasma following oxygen discharge cleaning has demonstrated dramatic improvement both in terms of increased temperature and plasma current.

To extend our knowledge of plasma-wall interactions, a new Tokamak, called ISX (impurity studies experiment), is being designed with interchangeable walls and limiters. We hope such experiments will lead to the development of materials and procedures for solving present impurity problems before next-generation Tokamaks are constructed. If so, sputtering will loom as the dominant impurity source. Sputtering has not yet been fully faced, since ion temperatures in present Tokamaks are too low.

Sustained Tokamak plasmas must be continuously fed by neutral beams or pellet injection. This implies that a continuous flux of charge-exchange neutrals will bombard the walls of future Tokamaks, leading to two deleterious effects: sputtering of wall materials and the reflection of neutrals back into the plasma. Honeycomb walls are designed to alleviate both these effects. With honeycomb walls a large fraction of sputtered particles is trapped inside the honeycomb structure, reducing sputtering by an estimated 3 to 4 times; likewise, the number of reflected neutrals is cut by about six.

In summary, present Tokamaks are afflicted by wall-related impurities. The most important of these impurities have been identified, their source found, and cleanup methods tested. However, these problems are far from solved. Looking ahead to Tokamaks with ion temperatures in the multi-keV range, honeycomb walls are one way to decrease both sputtering and particle reflection.

1. Summary of paper presented at the American Nuclear Society Meeting, New Orleans, Louisiana, June 8–13, 1975; *Trans. Am. Nucl. Soc.* 21, 53 (1975).

2. Research supported by ERDA's Division of Controlled Thermonuclear Research.

3. Thermonuclear Division.

114. REDUCTION OF NEUTRAL PARTICLE EMISSION INTO A CTR PLASMA BY USE OF HONEYCOMB WALLS[1,2]

S. N. Cramer E. M. Oblow

A major problem facing the fusion reactor program is the quenching of the confined plasma by refluxing neutral particles and sputtering neutral impurities from the vacuum wall. A honeycomb wall design has been investigated as a simple means of reducing the number of such particles that penetrate into the plasma.[3-4] Escaping plasma particles entering the cells of the honeycomb and sputtered particles produced there are reduced in number, and also in energy, by colliding with the honeycomb cell walls before entering the plasma.

To determine quantitatively the effectiveness of a honeycomb wall, stochastic simulation calculations were performed to follow the interactions of plasma and sputtered particles in an infinite lattice of honeycomb cells. The principal objective was to determine how effective a honeycomb would be compared with a smooth wall in reducing the number of particles entering the plasma. Also calculated were the energy and angular distributions of the escaping particles. The length-to-diameter ratio, L/D, of a single cell and the energy and angular distribution of the plasma particles entering the cell were parameters in the calculations. The scattering probabilities and angular distributions of the particle interactions with the cell wall and the sputtering yields were taken as rough approximations to experimental data and theoretical predictions.

Results of calculations shown in Fig. 114.1 indicate the reduced level of particles escaping from the open end of the honeycomb cells as a function of L/D. The ratios indicated for the ordinate in

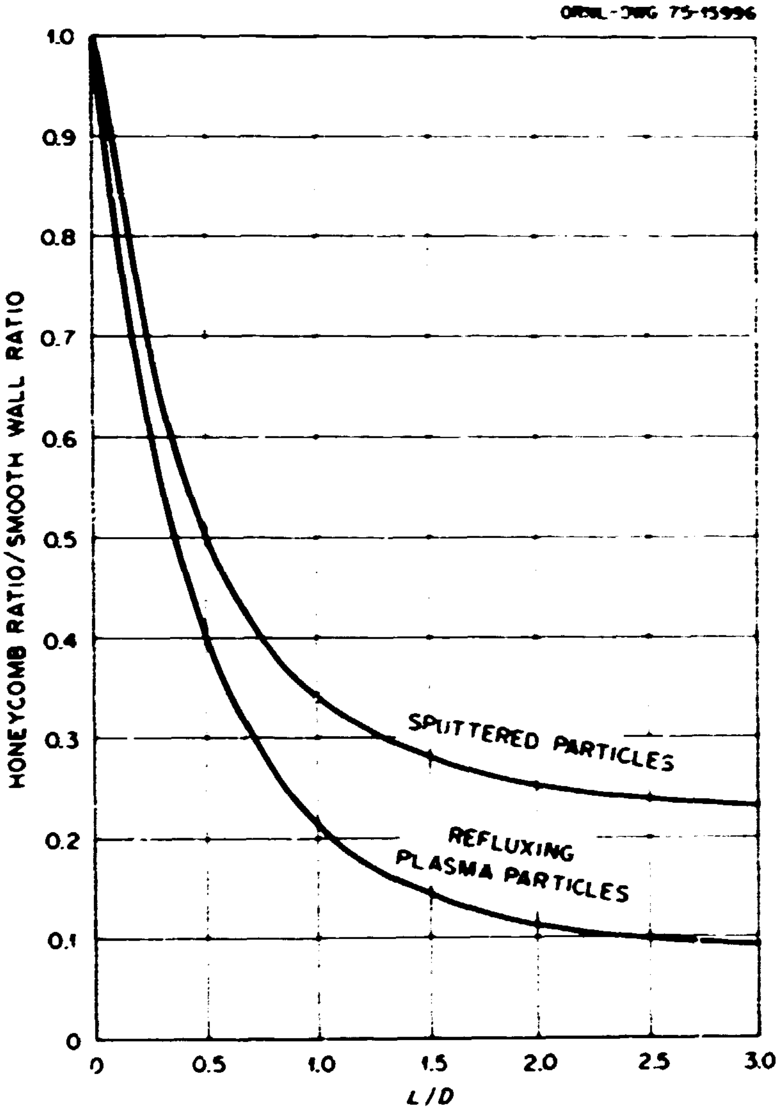

Fig. 114.1. Reduction of sputtered particles and refluxing plasma particles into a plasma by a honeycomb wall.

the figure refer to the number of particles escaping the cell divided by the number of particles entering it (or produced in the cell for the case of sputtering) for the honeycomb case divided by the same ratios for a smooth wall. It can be seen that the honeycomb wall reduces the sputtered particles by a factor of 4 and the refluxing neutral plasma particles by a factor of 10. In addition, these reductions were achieved for L/D values less than 2 (i.e., the cells are effectively infinite in length for L/D greater than 2).

Also calculated were the average energy and average angle cosines of the particles leaving the cells. It was found that the average energy of the refluxing plasma particles from the honeycomb was 60% of that from the smooth wall. This reduction coupled with the factor of 10 reduction in particle density makes the honeycomb effective in reducing the total energy returned to the plasma to 6% of that from a smooth wall.

Although the calculations leading to the results cited here were made with simple approximations to physical data, the calculation itself was a complete geometric simulation of particle transport in the honeycomb lattice. It is felt that as data become available, more detailed calculations will show that the honeycomb wall concept is a simple and inexpensive method for reducing the neutral particle emission into CTR plasmas.

1. Summary of paper presented at the 1975 Winter Meeting of the American Nuclear Society, San Francisco, California, November 16–21, 1975; this summary incorporates the abstract of ORNL-TM-4981, *Reduction of Neutral Particles Refluxing into a Plasma by Use of a Honeycomb Wall*, by S. N. Cramer and E. M. Oblow (October 1975).

2. Research sponsored by ERDA's Division of Controlled Thermonuclear Research.

3. S. N. Cramer and E. M. Oblow, *Feasibility Study of a Honeycomb Vacuum Wall for Fusion Reactors*, ORNL-TM-4708 (1974).

4. S. N. Cramer and E. M. Oblow, *Reduction of Refluxing Neutral Particles into a CTR Plasma by Use of a Honeycomb Wall*, ORNL-TM-4981 (1975).

115. COMPARISON OF THE CROSS-SECTION SENSITIVITY OF THE TRITIUM BREEDING RATIO IN VARIOUS FUSION REACTOR BLANKETS[1,2]

R. G. Alsmiller, Jr.　J. Barish[3]
R. T. Santoro　T. A. Gabriel

For several proposed fusion reactor blanket designs, the sensitivities of the tritium breeding ratios to uncertainties in nuclear cross-section data are presented and compared. The designs considered are those proposed at ORNL, the reference theta-pinch reactor design proposed at the Los Alamos Scientific Laboratory, and the reference fusion power plant design proposed at the Princeton Plasma Physics Laboratory. Results are presented for the changes in the breeding ratios due to estimated energy-dependent uncertainties in various partial cross sections of ^{6}Li, ^{7}Li, C, Be, and F. The ^{7}Li$(n,n'\alpha)t$ cross section, the Be$(n,2n')$ cross section, and the fluorine total cross section are found to introduce uncertainties of the order of a few percent in the breeding ratios for the various designs. Sensitivity profiles that show the changes in the breeding ratios due to changes in these cross sections in specific energy ranges are presented.

1. Abstract of ORNL-TM-4696 (October 1974) and *Nucl. Sci. Eng.* 57, 122 (1975).

2. Research sponsored by ERDA's Division of Controlled Thermonuclear Research.

3. Computer Sciences Division, UCC Nuclear Division.

116. CROSS-SECTION SENSITIVITY OF THE ENERGY DEPOSITION AND RADIATION DAMAGE IN THE TOROIDAL FIELD COIL OF A TOKAMAK EXPERIMENTAL POWER REACTOR[1,2]

R. G. Alsmiller, Jr.   J. Barish[3]

As part of the conceptual design studies of a Tokamak experimental power reactor at ORNL, a variety of blanket-shield-coil configurations have been considered from the point of view of neutronic performance as well as other engineering aspects. In this paper, calculated results are presented of the cross-section sensitivity of the energy deposition per unit volume and the radiation damage, that is, displacements per atom, in the toroidal field coil of one of the configurations considered.

The dimensions and compositions of the blanket-shield-coil configuration used in obtaining the results presented here are given in Table 116.1. The transport calculations were carried out with the discrete-ordinates code[4] ANISN, using P_3S_{12}. In all of the calculations, the coupled neutron-gamma-ray cross-section data library compiled by Plaster, Santoro, and Ford[5] was used. The neutron kerma factors used in calculating the energy deposition in copper were taken from the work of Abdou and Roussin,[6] and the photon kerma factors were obtained from the code SMUG, which is one of the modules in the AMPX code system.[7] The displacement cross sections in copper used in the damage calculations were taken from the work of Gabriel, Amburgey, and Greene.[8] All

Table 116.1. Blanket-shield-coil configuration (Cylindrical symmetry about plasma axis)

Material	Thickness (cm)	Outer radius (cm)
Plasma	200.0	200.0
Void	23.7	223.7
Graphite	0.3	224.0
Void	1.0	225.0
Iron	4.4	229.4
Blanket[a]	28.0	257.4
Iron	1.2	258.6
Void	15.0	273.6
Iron	3.0	276.6
Void	20.0	296.6
Iron	5.0	301.6
Shield[b]	40.0	341.6
Lead	5.0	346.6
Iron	5.0	351.6
Void	48.4	400.0
Iron	1.0	401.0
Void	3.0	404.0
Iron	5.0	409.0
Copper	33.0	442.0
Iron	40.0	482.0
Void	3.0	485.0
Iron	1.0	486.0

[a]Blanket composition: 37 v/o He, 31.5 v/o B_4C, and 31.5 v/o Fe.

[b]Shield composition: 65 v/o Fe, 34.65 v/o H_2O, and 0.35 v/o B.

of the sensitivity calculations were carried out with SWANLAKE.[9] All of the sensitivity results refer explicitly to a 3-cm section at the inner edge of the copper region (see Table 116.1), where both the energy deposition per unit volume and the displacements per atom in the copper region are a maximum.

The fractional change per unit lethargy in the energy deposition per unit volume due to a 1% increase in the iron total neutron cross section is shown in Fig. 116.1 as a function of neutron energy. (A discussion of the meaning of sensitivity profiles such as that shown in Fig. 116.1 is given in refs. 10 and 11.) The values shown in the figure are negative at nearly all energies, indicating that an increase in the iron total neutron cross section causes a decrease in the energy deposition per unit volume at the inner portion of the copper region. The energy-deposition sensitivities per unit lethargy are very large in Fig. 116.1 at the highest neutron energies, and, thus, the accuracy of calculated results of the energy deposition per unit volume in the copper is very dependent on the accuracy of the iron total neutron cross section at these higher energies.

Energy-deposition sensitivities per unit lethargy due to changes in the partial neutron cross sections in iron and to the total and partial neutron cross sections in carbon have also been obtained and will be presented. In addition, sensitivity results similar to those obtained for energy deposition per unit volume in the copper have been obtained for displacements per atom in the copper and will be presented.

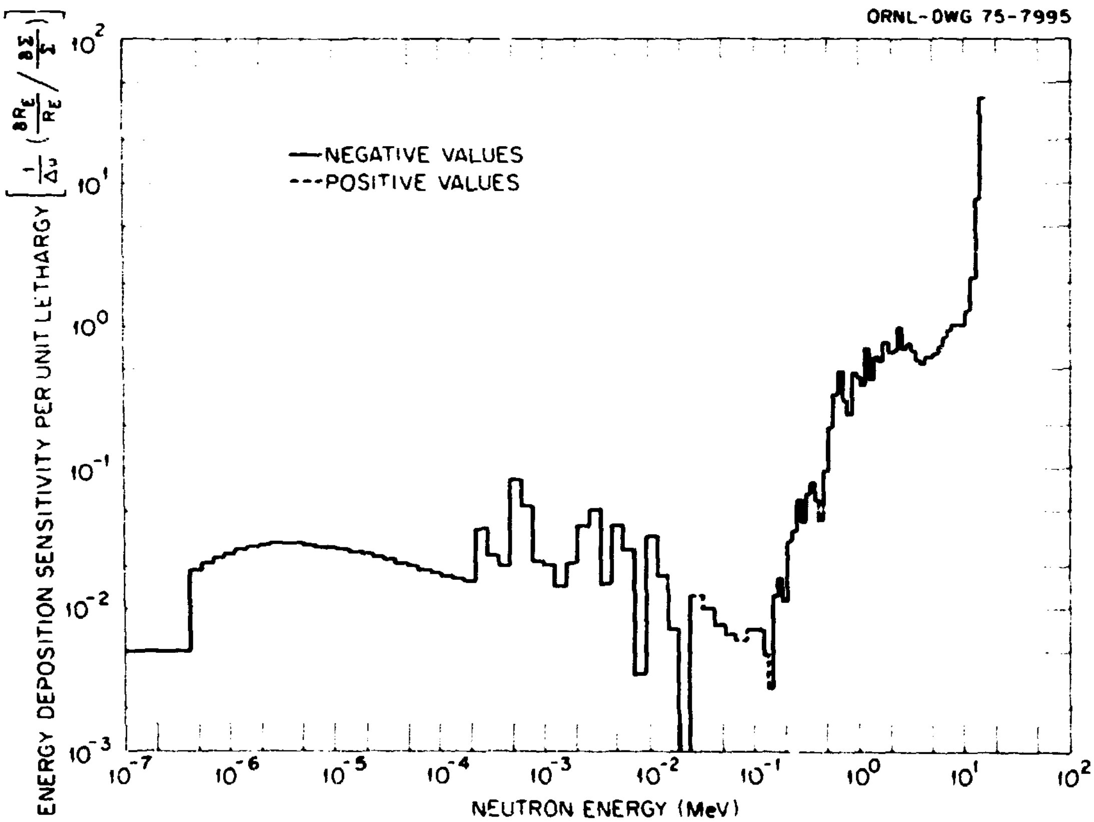

Fig. 116.1. Energy-deposition sensitivity per unit lethargy vs energy for increases in the iron total neutron cross section.

1. Summary of paper presented at the 1975 Winter Meeting of the American Nuclear Society, San Francisco, California, November 16–21, 1975.

2. Research sponsored by ERDA's Division of Controlled Thermonuclear Research.

3. Computer Sciences Division, UCC Nuclear Division.

4. W. W. Engle, Jr., *A Users Manual for ANISN, a One-Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering*, K-1693, Computing Technology Center, Union Carbide Corporation (1967).

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6. M. A. Abdou and R. W. Roussin, *MACKLIB – 100-Group Neutron Fluence-to-Kerma Factors and Reaction Cross Sections Generated by the MACK Computer Program from Data in ENDF Format*, ORNL-TM-3995 (1974).

7. N. M. Greene et al., *AMPX: A Modular Code System for Generating Coupled Multigroup Neutron-Gamma Libraries from ENDF/B*, ORNL-TM-3706 (to be published).

8. T. A. Gabriel, J. D. Amburgey, and N. M. Greene, *Trans. Am. Nucl. Soc.* 21, 67 (1975).

9. D. E. Bartine, F. R. Mynatt, and E. M. Oblow, *SWANLAKE, a Computer Code Utilizing ANISN Transport Calculations for Cross-Section Sensitivity Analysis*, ORNL-TM-3809 (1973).

10. D. E. Bartine, R. G. Alsmiller, Jr., E. M. Oblow, and F. R. Mynatt, *Nucl. Sci. Eng.* 53, 304 (1974).

11. D. E. Bartine, E. M. Oblow, and F. R. Mynatt, *Nucl. Sci. Eng.* 55, 147 (1974).

117. CROSS-SECTION SENSITIVITY OF THE D-T FUSION PROBABILITY AND THE D-T AND T-T REACTION RATES[1,2]

R. T. Santoro J. Barish[3]

The probability for producing positive thermonuclear power from the thermalization of fast deuterons in a tritium plasma has been studied by several researchers.[4,5] It has recently been suggested that the available D-T cross-section data may be in error and, therefore, that the calculated fusion probabilities may be quite uncertain.[5] To provide some insight into the influence of D-T cross-section errors in specific energy ranges on the fusion probability, the sensitivity of this probability has been calculated for various conditions of incident deuteron energy and plasma electron temperature.

The change in the fusion probability, f, due to a fractional change, $\delta\sigma/\sigma$, in the D-T cross section in the energy range $E_i - E_{i+1}$ may be calculated from the equation

$$\delta_f = \delta\sigma/\sigma \int_{E_i}^{E_{i+1}} I(E')\,dE' .$$

where $I(E')$, the fusion probability per unit energy, is, following Pistunovich,[5] directly proportional to the D-T cross section and inversely proportional to the energy loss per unit time of a deuteron in a tritium plasma. In the calculations, the analytic fit to the D-T cross-section data given by Duane[7] and the energy loss per unit time given by Sivukhin[3] were used.

The sensitivity per unit lethargy, $[(\delta f/f)/(\delta\sigma/\sigma)/\delta u]$, is shown in Fig. 117.1 as a function of energy when 200-keV deuterons are thermalized in a tritium plasma having an electron temperature of 5 keV. The results given in the figure are independent of ion density in the plasma, since this density enters into the calculations only through the Coulomb logarithm. When the histogram values in the figure are multiplied by the lethargy interval, the percentage change in the fusion probability for a 1% increase in the D-T cross section in a particular lethargy interval is obtained. The value in a given lethargy interval is directly proportional to $\delta\sigma/\sigma$, so results for any assumed change in the cross section can be obtained by multiplication. The results indicate that the fusion probability is most sensitive to the D-T cross section at the higher energies and that, based on the reported errors in the cross section in this energy range,[6] the errors in the calculated fusion probabilities should be $\lesssim 10\%$.

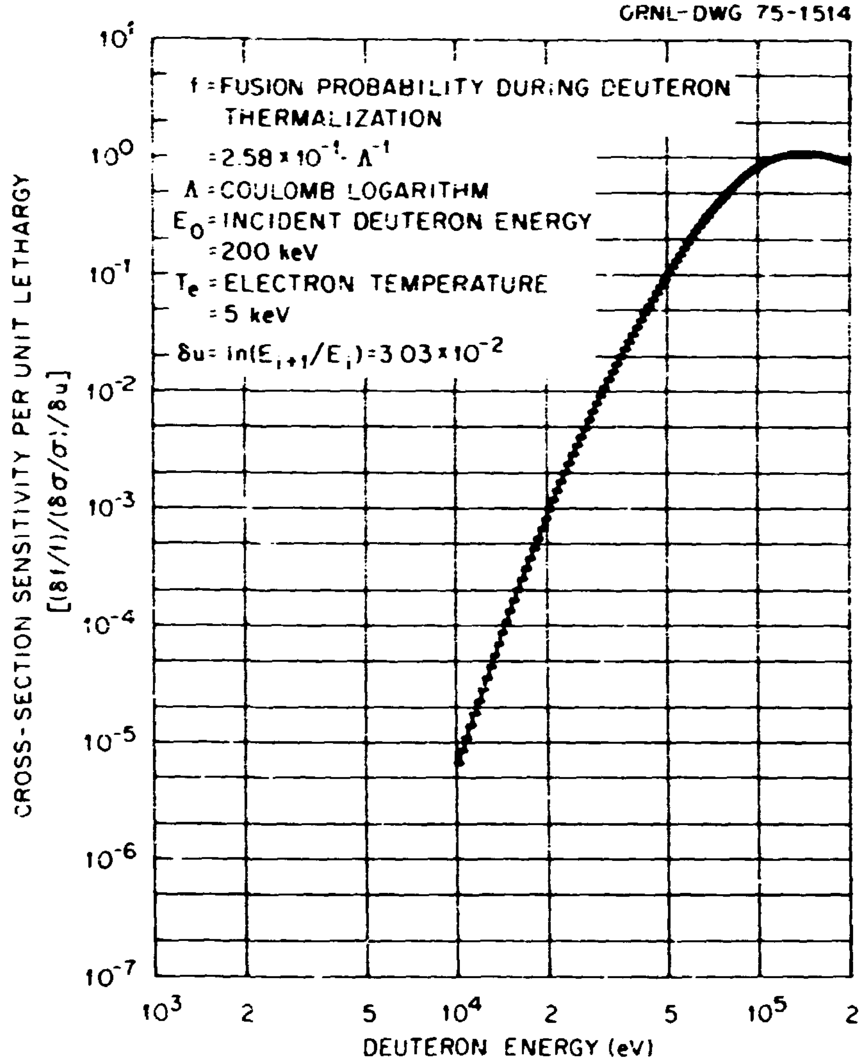

Fig. 117.1. Sensitivity of the fusion probability to the D-T cross section vs deuteron energy for deuteron injection in a tritium plasma.

The cross-section sensitivities of the D-T reaction rate in a D-T plasma and the T-T reaction rate in a tritium plasma have also been calculated for various assumed values of the plasma ion temperature. The T-T reaction rate, in particular, is of interest because of the suggestion that the tritium temperature in the TFTR, which will be built at Princeton, might be inferred from measurements on the neutrons produced by T-T reactions.[5] The results indicate that the T-T reaction rate is very uncertain, because essentially all the contributions to this reaction rate come from an energy region where no experimental data on the T-T reaction cross section have been reported.

1. Summary of paper presented at the American Nuclear Society Meeting, New Orleans, Louisiana, June 8-13, 1975; *Trans. Am. Nucl. Soc.* **21**, 47 (1975); also summary of ORNL-TM-4933 (July 1975).
2. Research sponsored by ERDA's Division of Controlled Thermonuclear Research.
3. Computer Sciences Division, UCC Nuclear Division.
4. J. M. Dawson, H. P. Furth, and F. H. Tenney, *Phys. Rev. Lett.* **26**, 1156 (1971).
5. V. I. Pistunovich, *At. Energ.* **35**, 11 (1973).

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8. D. V. Sivukhin, "Coulomb Collisions in a Fully Ionized Plasma," p. 93 in *Reviews of Plasma Physics*, Acad. Ed. M. A. Leontovich, vol. 4, Consultants Bureau, New York, 1966.

118. NEUTRONIC SCOPING STUDIES FOR A TOKAMAK EXPERIMENTAL POWER REACTOR[1,2]

R. T. Santoro D. J. McAlees[4]
E. S. Bettis[3] H. L. Watts[5]
M. W. Williams

Neutronics calculations have been used in scoping studies directed toward the selection of candidate blanket configurations and materials to meet Tokamak Experimental Power Reactor (EPR) objectives.[6] The intent of these calculations has been to systematically investigate and analyze selected blanket assemblies in a coordinated effort with plasma, design, thermodynamic, and materials engineers to survey these designs from the point of view of energy recovery, radiation attenuation, potential radiation damage, and tritium breeding.

The neutronics calculations were made for seven candidate blanket designs with the discrete ordinates code ANISN,[7] using P_3S_{12} and the coupled neutron-gamma-ray cross-section library of Plaster, Santoro, and Ford[8] (100 neutron groups, 21 gamma-ray groups). Energy deposition in the reactor was estimated using coupled neutron-photon kerma factors obtained from MACKLIB[9] and SMUG[10] respectively.

The reactor model used in these calculations is characterized by a 200-cm-radius plasma surrounded by a 0.3-cm-thick graphite curtain. The blanket assembly originates at a radius of 225 cm and has a radial dimension determined by the blanket configuration being studied. Surrounding the blanket at a radius of 296.6 cm is a 55-cm-thick shield containing a 40-cm-thick 65 v/o Fe, 35 v/o H_2O-B assembly sandwiched between stainless steel and lead liners. The shield serves to further protect the torroidal field coils that extend to a radius of 485 cm.

The seven blanket designs that were scoped are summarized in Table 118.1. Neutronics calculations that define the radial dependence of the energy deposited by neutrons and secondary

Table 118.1. EPR blanket characteristics (structure: stainless steel)

Model	Outer radius (cm)	Absorber	Coolant	Reflector	Heat deposition (W/neutron)	Breeding ratio
					$\times 10^{-12}$	
1	276.6	[illegible]·Fe	He[a]		2.12	
2	268.3	B_4C·Fe	He[b]		2.31	
3	273.0	Flibe[c]	Flibe	C	2.37	0.53
4	271.3	Be·Li	Flibe	C	2.39	0.66
5	258.1	Li[d]	He[b]		1.88	0.58
6	258.1	^{6}Li[e]	He[b]		1.86	0.79
7	276.5	Li[d]	He[b]	C	2.40	1.21

[a]Low-pressure helium.
[b]High-pressure helium.
[c]66 m/o LiF, 34 m/o BeF_2.
[d]Natural lithium.
[e]90% enrichment.

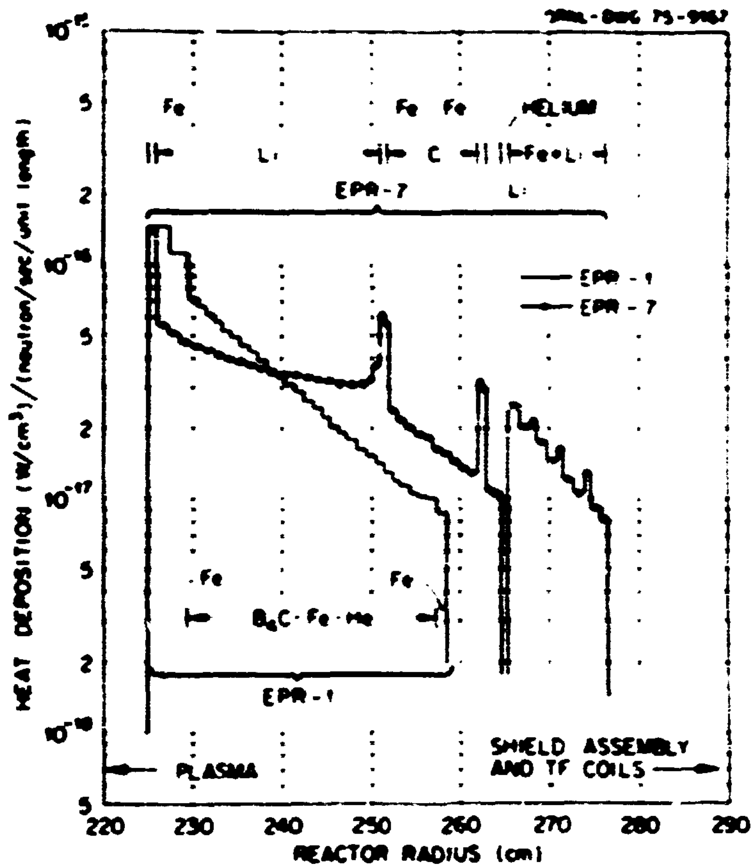

Fig. 118.1. Radial dependence of the energy deposition for the EPR-1 and EPR-7 blanket designs.

(The results are for the energy deposition per unit volume in a 1-cm section adjacent to the inner edge of the copper region in Table 118.1.)

gamma rays, the radial dependence of the neutron and photon attenuation, and, when applicable, the tritium breeding ratio were carried out for each design. The radial dependence of the energy deposition for the EPR-1 and EPR-7 designs is given in Fig. 118.1. In each design, approximately 90% of the incident radiation is converted to heat suitable for conversion to electrical power. Since the EPR must demonstrate tritium breeding in selected blanket modules, the EPR-7 design includes lithium. The remainder of the EPR-7 blanket modules required only for power production will have potassium as the absorber.

A discussion of the neutronic performance of each blanket type and the effectiveness of the shield in protecting the toroidal field coil will be presented.

1. Summary of paper presented at the 1975 Winter Meeting of the American Nuclear Society, San Francisco, California, November 16–21, 1975.

2. Research sponsored by ERDA's Division of Controlled Thermonuclear Research.

3. Consultant.

4. Exxon Nuclear Company, Inc.

5. Engineering Division, UCC Nuclear Division.

6. P. N. Haubenreich, ed., *Tokamak Experimental Power Reactor: Basic Considerations and Initiation of Studies at Oak Ridge*, ORNL-TM-4853 (1975).

7. W. W. Engle, Jr., *A Users Manual for ANISN, a One-Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering*, K-1693, Computing Technology Center, Union Carbide Corporation (1967).

8. D. M. Plaster, R. T. Santoro, and W. E. Ford III, *Coupled 100-Group Neutron and 21-Group Gamma-Ray Cross Sections for EPR Calculations*, ORNL-TM-4872 (1975).

9. M. A. Abdou and R. W. Roussin, *MACKLIB: 100-Group Neutron Fluence-to-Kerma Factors and Reaction Cross Sections Generated by the MACK Computer Program from Data in ENDF Format*, ORNL-TM-3995 (1974).

10. N. M. Greene et al., *AMPX: A Modular Code System for Generating Coupled Multigroup Neutron-Gamma Libraries from ENDF/B*, ORNL-TM-3706 (to be published).

119. NEUTRONICS CALCULATIONS FOR THE TOKAMAK EXPERIMENTAL POWER REACTOR REFERENCE DESIGN[1,2]

R. T. Santoro

The results of initial one-dimensional calculations evaluating the nuclear performance of the Tokamak Experimental Power Reactor are presented. Estimates of the tritium breeding, nuclear heating, and radiation damage are given for an assumed neutron wall loading of 0.168 MW/m^2 (100 MW).

1. Abstract of ORNL-5033 (September 1975).
2. Research sponsored by ERDA's Division of Controlled Thermonuclear Research.

120. INFLUENCE OF STRUCTURAL MATERIALS ON FUSION-REACTOR-BLANKET RESPONSE[1,2]

M. L. Williams R. T. Santoro T. A. Gabriel

The performance of fusion reactors, as well as the solution of problems concomitant with their operation and maintenance, will be influenced by the selection of structural material. In this paper, the effects on nuclear response functionals (e.g., tritium breeding ratio, energy deposition, absorbed dose during operation, material activation, activation dose after shutdown, and radiation damage) are calculated when niobium, SS-304, and nimonic-105 are considered as the structural materials in the ORNL conceptual reactor design.[3] The results differ from the earlier work of Steiner,[3] Price,[4] and Conn, Sung, and Abdou,[5] principally in the choice of materials and in the use of more recent nuclear data.

Transport calculations were performed with the discrete-ordinates code ANISN,[6] using a P_3 scattering expansion, an S_{12} quadrature, and the coupled neutron-gamma cross-section library of Plaster, Santoro, and Ford[7] (100 neutron groups and 21 gamma-ray groups). Neutron and gamma-ray kerma factors for computing the energy deposition were generated by MACKLIB[8] and SMUG[9] respectively. Activation cross sections were obtained from ENDF/B-IV and from the compilation of Muir,[10] and displacement cross sections were taken from the work of Gabriel, Amburgey, and Greene.[11] Blanket responses were calculated by changing structure cross sections and holding other parameters constant to obtain energy-dependent fluxes as functions of structural material. These values were then substituted into the appropriate functionals to evaluate the desired responses.

A summary of nuclear responses as functions of the selected materials for continuous operation with a wall loading of 1 MW/m^2 is given in Table 120.1. The lower breeding ratio with nimonic-105 structure is due to high parasitic absorption in nickel and cobalt. Absorbed dose at the outer wall during operation is highest with nimonic, due to secondary gamma rays from cobalt. The activation dose is high for all materials, but values will be reduced when conditions other than continuous operation are assumed.

Table 120.1. Comparison of the blanket responses as a function of structural material

Response	Structural Material		
	Nb	SS-304	Nimonic-105
Tritium breeding ratio	1.52	1.53	1.46
First-wall heating (W/cm of circumference)	3.05×10^{3}	4.42×10^{3}	5.19×10^{3}
Operating absorbed dose at the outer wall (rads/hour)	1.75×10^{7}	1.74×10^{7}	2.21×10^{7}
First-wall activation at shutdown (dis sec^{-1} cm^{-1})	7.99×10^{12}	2.29×10^{12}	6.61×10^{12}
Biological dose at one day after shutdown (rems/hour)	5.76×10^{6}	1.02×10^{6}	5.38×10^{6}
Afterheat at shutdown (W/cm of circumference)	3.86×10^{2}	7.93×10^{1}	5.22×10^{2}
First-wall damage (DPA/year)	12.84	10.80	16.89

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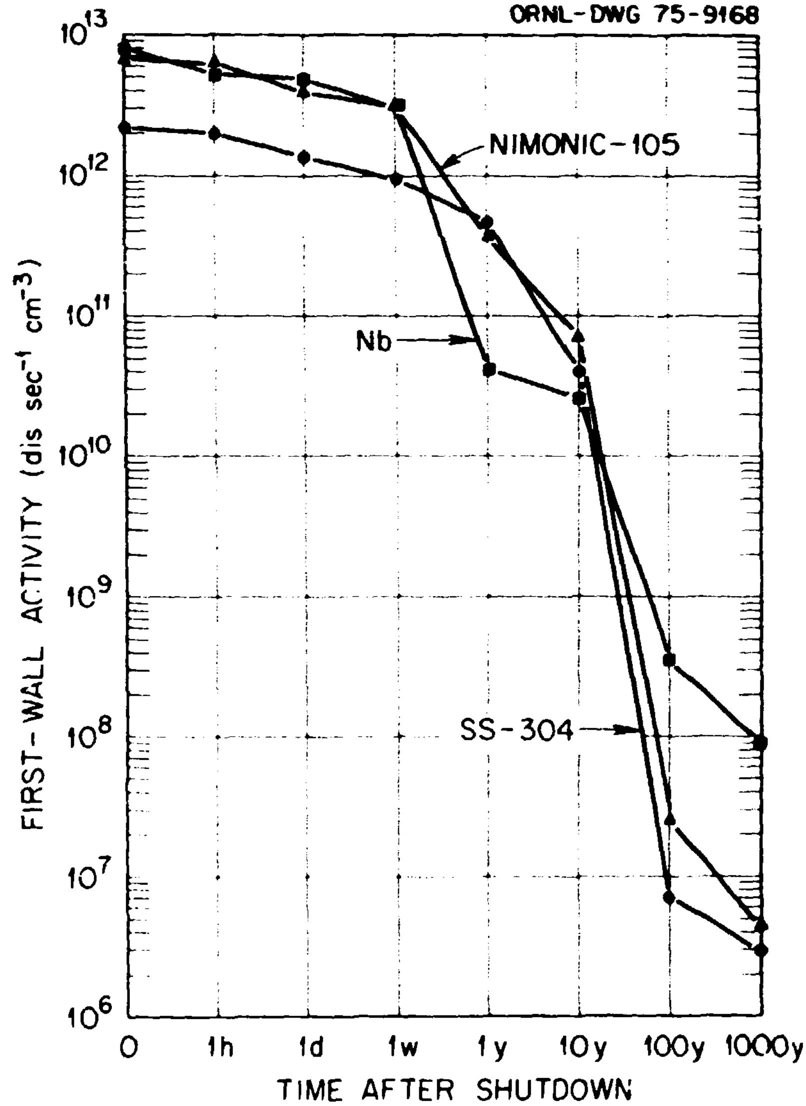

Fig. 120.1. First-wall activity as a function of time after shutdown after 1 year of continuous operation at 1 MW/m^2.

The first-wall activation after one year of continuous operation at 1 MW/m² is shown in Fig. 120.1 as a function of time after shutdown. The niobium activity is dominated at early times by ^{92m}Nb ($t_{1/2}$ = 10.1 days) and at later times by ^{91m}Nb ($t_{1/2}$ = 3.7 years). Type 304 stainless steel activity follows the decay of ^{54}Mn ($t_{1/2}$ = 313 days) and ^{55}Fe ($t_{1/2}$ = 2.4 years), while nimonic-105 activity depends mostly on ^{58}Co (71.4 days) and ^{60}Co (5.3 years).

1. Summary of paper presented at the 1975 Winter Meeting of the American Nuclear Society, San Francisco, California, November 16–21, 1975.

2. Research sponsored by ERDA's Division of Controlled Thermonuclear Research.

3. D. Steiner, *Nucl. Fusion* 14, 33 (1974).

4. W. G. Price, Jr., "Activation Hazards from Fusion Neutrons," p. 217 in *Proc. Fifth Symp. Eng. Problems of Fusion Res.*, Nov. 5–9, 1973, Princeton University, Princeton, N.J., IEEE Pub. No. 73 CHO 843-3-NPS.

5. R. W. Conn, T. Y. Sung, and M. A. Abdou, *Comparative Study of Radioactivity and Afterheat in Several Fusion Reactor Blanket Designs*, FDM-113, University of Wisconsin (1974).

6. W. W. Engle, Jr., *A Users Manual for ANISN, a One-Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering*, K-1693, Computing Technology Center, Union Carbide Corporation (1967).

7. D. M. Plaster, R. T. Santoro, and W. E. Ford III, *Coupled 100-Group Neutron and 21-Group Gamma-Ray Cross Sections for EPR Calculations*, ORNL-TM-4872 (1975).

8. M. A. Abdou and R. W. Roussin, *MACKLIB: 100-Group Neutron Fluence-to-Kerma Factors and Reaction Cross Sections Generated by the MACK Computer Program from Data in ENDF Format*, ORNL-TM-3995 (1974).

9. N. M. Greene et al., *AMPX: A Modular Code System for Generating Coupled Multigroup Neutron-Gamma Libraries from ENDF/B*, ORNL-TM-3706 (to be published).

10. D. W. Muir, DLC-33 Data Library MONTAGE, Radiation Shielding Information Center, Oak Ridge National Laboratory (1975).

11. T. A. Gabriel, J. D. Amburgey, and N. M. Greene, *Trans. Am. Nucl. Soc.* 21, 67 (1975).

121. DESIGN OF AN EPR BLANKET[1,2]

E. S. Bettis[3] R. T. Santoro
T. J. Huxford[4] H. L. Watts[4]
D. G. McAlees[5] M. L. Williams

Scoping studies have been made at Oak Ridge National Laboratory for an EPR-sized fusion reactor (100- to 400-MW thermal). An important part of the studies involved the preliminary design of an energy removal blanket.

The blanket must perform three functions. It must absorb a high fraction (~90%) of the plasma energy, 0.2 to 0.7 mW/m², and convert it to heat at a usefully high temperature. It must be capable of breeding tritium and, together with the shield, it must protect the toroidal field coils from radiation and neutrons. Several concepts of blankets were investigated. Solid absorbers, circulating absorber coolants, and gas-cooled solid and gas-cooled liquid absorbers were evaluated. Variations of seven different concepts were considered, and only one was found to satisfy all the criteria.

The reference design blanket segment is made of type 316 stainless steel and uses stagnant natural lithium as the absorbing fertile material. This liquid metal provides conductive cooling of the inner wall by effecting good heat transfer to the high-pressure helium cooling tubes. The blanket is 51.5 cm thick, and behind the 25-cm lithium compartment are a graphite reflector compartment (10 cm) and a stainless steel gamma-ray shield compartment (11.5 cm). Helium at 70 atm pressure is circulated through three sets of ½-in.-OD tubes to remove the heat. The coolant enters at about 316°C and exits at 528°C.

One-dimensional neutronic calculations were made that gave the breeding gain (1.21), the radial volumetric heat deposition, and the fraction (~90%) of the plasma power deposited in the blanket. Using these energy deposition calculations, one-dimensional thermal hydraulic analyses were made of the blanket. Temperatures were calculated for radial and azimuthal locations in the blanket and were found to be within acceptable limits. Also, the pumping power for the coolant was calculated to be about 3.5% of the total thermal power absorbed when circulating the coolant at the required mass flow of 147 kg sec.

By keeping the wall temperature at about 550°C, the corrosion of the structure by the lithium is well within tolerable rates. The inert helium does not become radioactive, and the tritium that diffuses through the tubes is removed by side stream processing.

The 60 segments of the blanket are mechanically clamped together to form a rigid torus structure. On the outside of the blanket on each side of each segment is a bellows. After the segments are mechanically clamped together for structural integrity, a seal weld is effected between these bellows. These welds provide the vacuum containment of the plasma. Both the clamping and welding are effected by totally remote means so that the blanket can be maintained after having become radioactive.

For most of the segments of the EPR, the lithium metal is replaced by potassium in order to minimize the tritium processing and handling problems. The blanket does not employ scarce materials, and it can be fabricated by well-established techniques.

More sophisticated multidimensional analyses for optimization remain to be done. Also needed are careful analyses of thermal stresses and the effects of thermal cycling.

The inner wall of the blanket is to be protected from the plasma environment by a separate wall if studies of the effects of the plasma on the stainless steel show that such a protection "curtain" is necessary.

1 Summary of paper presented at the Sixth Symposium on Engineering Problems of Fusion Research, San Diego, California, November 18-21, 1975

2 Research sponsored by ERDA's Division of Controlled Thermonuclear Research

3 Consultant

4 Engineering Division, UCC Nuclear Division

5 Exxon Nuclear Company, Inc

122. PHYSICS AND ENGINEERING ASPECTS OF THE OAK RIDGE EXPERIMENTAL POWER REACTOR[1,2]

D. G. McAlees[3] F. B. Marcus[4]
E. S. Bettis[4] R. T. Santoro
T. J. Huxford[5] H. L. Watts[5]

The first experimental fusion power reactor (EPR) is planned for operation in 1985. In this paper the technical aspects of the plasma, blanket shield (including the neutronics and heat transfer analyses), and the overall operating cycle of the Oak Ridge EPR reference design are summarized.

The EPR size is such that it should produce a reactor-grade, power-producing plasma. This is the objective of the system. The steps taken in selecting the device characteristics, that is, plasma radius a = 2.25 m, major radius R_0 = 6.75 m, toroidal field strength B_T = 4.8 T, and plasma current I = 7.2 MA, are outlined. Results from a time-dependent plasma model are used to describe the possible operating cycles for the system. They are found to include beam-driven and ignition modes due to the uncertainty in our present knowledge with respect to scaling in large Tokamaks. In both modes the system provides reactor-grade plasma operation (200 to 400 MW) and interesting neutron wall loadings, and it satisfies the EPR objectives.

The reference blanket is composed of 30 autonomous segments which, when clamped and welded together, make up the toroidal chambers. This structure also constitutes the vacuum enclosure for the plasma. There are three main regions in the blanket. The first is the main neutron absorber composed of potassium, except in the experimental breeding segments where lithium is used. The second region is a graphite reflector. The final region, a gamma-ray shield, is constructed of stainless steel. The blanket does not attenuate the neutron and gamma-ray fluxes sufficiently to protect the superconducting toroidal field coils. A water-cooled type 316 stainless steel shield provides the required attenuation. The key mechanical and nuclear characteristics of the blanket and shield are outlined in detail.

The one-dimensional discrete-ordinates code ANISN[7] was used to estimate the nuclear performance of the reference design, including the gamma-ray and neutron energy deposition and the radiation damage to be expected. Also, a breeding ratio of ~1.21 was calculated in the case of the experimental blanket module design. Based on the results of the energy production and deposition calculations, the characteristics for the blanket cooling system were determined. Additional criteria require the blanket structure to operate at a temperature of ~550°C and the inlet and outlet coolant temperatures to be ~315 and ~540°C respectively. The temperature distributions in the blanket were calculated using a one-dimensional conduction model. Results of these analyses are given.

The performance of the device is summarized in the context of one operating cycle. The overall duty cycle is expected to be >50%. The parameters attained, the energy required, and the energy produced during the cycle are discussed.

1. Summary of paper presented at the Sixth Symposium on Engineering Problems of Fusion Research, San Diego, California, November 18–21, 1975.
2. Research sponsored by ERDA's Division of Controlled Thermonuclear Research.
3. Exxon Nuclear Company, Inc.
4. Consultant.
5. Engineering Division, UCC Nuclear Division.
6. Thermonuclear Division.
7. W. W. Engle, Jr., *A Users Manual for ANISN, a One-Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering*, K-1693, Computing Technology Center, Union Carbide Corporation (1967).

123. MEASUREMENT OF NEUTRON FLUX FROM A TOKAMAK PLASMA DEVICE[1,2]

G. L. Morgan   A. C. England[3]

A system has been developed for neutron flux measurements at the ORMAK Tokamak device. This system allows measurement of the time dependence of the neutron flux during a single shot of the device as well as the energy spectrum of the neutrons. Pulse-shape discrimination techniques are used with a liquid scintillation counter (NE-213) to eliminate events in the detector due to gamma rays and x rays generated in the plasma. Results are presented for the mode in which neutral deuterium is injected into a deuterium plasma.

1. Abstract of published paper: *Nucl. Instrum. Methods* **128**, 1 (1975).
2. Research sponsored by ERDA's Division of Controlled Thermonuclear Research.
3. Thermonuclear Division.

124. NEUTRON EMISSION FROM ORMAK[1,2]

L. A. Berry[3] A. C. England[3] J. F. Lyon[3] G. L. Morgan

Neutron rates from reactions between a deuterium beam and a deuterium plasma have been studied in the ORMAK. A simple theory has been used to calculate the absolute rate from coinjection and counterinjection. This simple theory utilizes the ohmic heating toroidal electric field and the Spitzer slowing-down time, but neglects the loss region, the deuteron deficiency, the trapping efficiency, and the energy diffusion.

The theoretical rates are in close agreement with the measured rates. The measured rates have been obtained both with ^{3}He proportional counters and with an organic proton-recoil scintillator with pulse-shape discrimination against x and gamma rays.

Neutron energy measurements have also been made using pulse-height unfolding techniques on the scintillator data. A low-energy neutron group, probably from inelastic collisions of neutrons and aluminum, has been observed. Inelastic groups from copper and iron could not be resolved.

Comparison between measured neutron rates and rates expected from the tangential charge-exchange spectra are poor. At present there is no satisfactory explanation for this discrepancy.

1. Abstract of ORNL-TM-5054 (September 1975).
2. Research sponsored by ERDA's Division of Controlled Thermonuclear Research.
3. Thermonuclear Division

125. CRUNCHER: A CHANNEL COMPRESSION DEVICE FOR TIME-OF-FLIGHT EXPERIMENTS[1,2]

L. A. Remez[3]

The device described in this report performs a variable "channel compression" operation on the binary data provided by a time digitizer (EGG TDC-100) before this data is fed to an on-line data acquisition computer (SEL 810B). In this way, original data requiring 26 lines for transmission and over 67 million channels for storage can be transmitted over 15 lines to less than 33,000 storage channels (in the maximum compression mode) without loss in data quality (resolution) for normal experimental conditions. There are also significant savings of central processor time per data, which would be otherwise "crunched" by a software routine. This last feature is particularly interesting for time-shared data acquisition systems.

1. Abstract of ORNL-TM-4871 (March 1975).
2. Research sponsored by ERDA's Division of Reactor Research and Development.
3. IAEA Fellow. Present address: Comision Nacional de Energia Atomica, Centro Atomico Bariloche, Bariloche, RN, Argentina.

126. USE OF ORELA TO PRODUCE NEUTRONS FOR SCATTERING STUDIES ON CONDENSED MATTERS[1,2]

R. W. Peelle J. T. Mihalczo[3]
T. A. Lewis[3] H. A. Mook[4]
R. M. Moon[4]

The Oak Ridge Electron Linear Accelerator is evaluated as a source of neutrons for condensed-matter research. The neutron energies of interest lie in the region between 0.2 and 1 eV.

since the HFIR is available for energies less than 0.2 eV. For typical important experiments in this domain, the scattering sample should be placed 3 to 5 m from the neutron source. A pulse repetition rate ≥500 pps can be utilized. Clean hydrogenous moderators which give about 2-μsec FWHM pulses at 1 eV are indicated, because for a given resolution the effective counting rate is inversely proportional to the first or second power of the breadth of the neutron pulse at the moderator surface. Two options are assessed: (1) use of the present target arrangement with minor modifications and (2) the construction of a new target and experiment facility designed for condensed-matter research and equipped with a subcritical fission booster. The expected source strength and time behavior are discussed, including the fundamentals of moderator design. The effect on the programs presently using the linac is considered.

The ORELA with its present target meets very well some of the criteria for a good pulsed source for scattering studies. Its electron pulses are much shorter than the moderator pulse width, the neutrons are produced in a compact region with high source brightness, and few neutrons are produced between bursts. The ORELA now produces about 10^{14} fast neutrons per second at ≥600 pps. The capital cost of making the ORELA available for a scattering experiment will be about 0.1 M\$. The ORELA source strength may be compared with the 10^{13} neutrons per second linac pulsed source used for some years at Tohoku University for scattering studies, the 7×10^{14} neutrons per second expected at 120 pps from the Los Alamos WNR facility using protons from the LAMPF, the 5×10^{15} neutrons per second at about 60 pps from the proposed ZING facility at Argonne, and the 2×10^{16} neutrons per second expected in 3-μsec pulses from the IBR-II electron-driven booster at Dubna. The intensity advantage of the proton machines must be discounted by perhaps a factor of up to 2 because of their large source dimensions, while the ORELA output is reduced when an experiment cannot utilize a 600-pps repetition rate.

1. Abstract of ORNL-TM-4987 (August 1975).
2. Research sponsored by ERDA's Division of Reactor Research and Development.
3. Instrumentation and Controls Division.
4. Solid State Division.

127. COSMIC RADIATION EXPOSURE IN SUPERSONIC AND SUBSONIC FLIGHT[1]

Final Report of the Advisory Committee for Radiation Biology Aspects of the SST[2]

The main body of this document consists of four major sections: (*A*) an introduction describing the scope of Committee operations and providing a brief exposition of the concepts of radiation protection; (*B*) a survey of experimental and theoretical data on cosmic radiations that have been obtained in individual research projects, with emphasis on investigations performed under the sponsorship of the Committee (the studies evaluate galactic and solar radiation as a function of altitude and magnetic latitude); (*C*) best current estimates of cosmic radiation levels in the atmosphere; (*D*) radiation protection recommendations dealing with maximum permissible doses and operational aspects covering satellite warning systems, on-board instrumentation, and forecasting. Nine annexes submitted by individual authors cover various of these subjects in greater detail.[3]

1. Abstract of Advisory Committee for Radiation Biology Aspects of the SST, final report, *Aviat. Space Environ. Med.* 46(9), 1170 (1975).

2. F. C. Maienschein was a member of this committee, which was chaired by H. H. Rossi.

3. Annex IV, "Theoretical Assessment," was written by F. C. Maienschein, Neutron Physics Division, ORNL; C. F. McIlwain, University of California, San Diego; and Herbert Sauer, Space Environment Laboratory, NOAA.

PUBLICATIONS AND PAPERS PRESENTED AT SCIENTIFIC MEETINGS

PUBLICATIONS*

ABBOTT, LORRAINE S., AND F. R. MYNATT

Review of ORNL Radiation Shielding Analyses of the Fast Flux Test Facility Reactor. ORNL-5027 (July 1975). (95)

ALSMILLER, R. G., JR.

"Calculations of the Transport of Fast Neutrons ($\leq$50 MeV) Through Matter," p. 681 in *Proc. Second Symposium on Neutron Dosimetry in Biology and Medicine, Neuherberg Munchen, September 30–October 4, 1974*, EUR-5273 d-e-f, II, 1974. (1974)

ALSMILLER, R. G., JR., AND J. BARISH**

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