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NEUTRON PHYSICS DIVISION

Progress Report

Period Ending February 28, 1977

OAK RIDGE NATIONAL LABORATORY

OPERATED BY UNION CARBIDE CORPORATION FOR THE ENERGY RESEARCH AND DEVELOPMENT ADMINISTRATION

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FOR PERIOD ENDING FEBRUARY 28, 1977

F. C. Maienschein, Director

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OAK RIDGE NATIONAL LABORATORY
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INTRODUCTION AND SUMMARY

This report is prepared as a record of the progress of the Neutron Physics Division scientific programs for the time period extending from October 31, 1975, to February 28, 1977, no progress report having been issued from the Division during the calendar year 1976. As has been the custom for the past several years, the report consists of a compilation of abstracts and summaries of papers and reports published or presented at scientific meetings during the reporting period. Under this policy any work that has only recently been initiated or has not yet reached the reporting stage is not represented among the compilation. Thus to present an overview of the Division's program as of this writing (March 31, 1977) and to show the interrelationship of the various areas of research, we offer the following discussion of the Division's current activities.

Division Programs

The program of the Neutron Physics Division continues to be performed primarily under the aegis of several of the research divisions of the Energy Research and Development Administration (ERDA), with additional research sponsored by the Nuclear Regulatory Commission (NRC) and the Defense Nuclear Agency (DNA). We also continue to perform special studies for other agencies, most recently for the Electric Power Research Institute (EPRI), the U. S. Army Ballistics Research Laboratory (BRL), and Princeton University. In addition, we have during this reporting period initiated other studies of which one or more could materialize into significant programs.

Research performed for ERDA's Division of Reactor Development and Demonstration (DRDD) still constitutes the largest program within the Division; however, during the past few months the program has experienced a shift of emphasis. The shift results from the suspension of experiments at the Tower Shielding Facility, which has recently been placed in standby, together with an increase in analysis-type activities. In particular, studies of the neutronics of fast-reactor cores have expanded, and the large calculational system (FORSS) for studying the sensitivity of transport/diffusion calculations to nuclear data is undergoing considerable additional development and testing, primarily under DRDD sponsorship. On the other hand, shield analysis efforts have been slowed as we have essentially completed the work for the Fast Flux Test Facility (FFTF) and have entered the final phases for the Clinch River Breeder Reactor (CRBR). Similarly, the development of radiation transport techniques is receiving less emphasis as the large code systems we developed in earlier years are applied routinely, not only by us but by many throughout the world. This widespread use of the codes is facilitated by the efforts of the Radiation Shielding Information Center, which is also largely under the sponsorship of DRDD. RSIC not only distributes the codes but works directly with the users. The Center also continues to serve the shielding community in numerous other areas.

The more basic studies which have received long-term support from DRRD, that is, cross-section measurements, evaluations, and processing, together with cross-section uncertainty studies, remain an important part of the Division's program. We also now have the recently established nuclear safety data base in which we are collecting and analyzing data to be used in fast reactor safety analysis codes, and we expect to soon have a second data base, called the "reliability" data base, which will be used in risk analyses of fast reactors.

The Division's shielding studies for the Gas Cooled Fast Reactor, formerly supported under the DRDD program but now funded under the Division of Nuclear Research and Applications, have continued as a combination of method application to specific GCFR shield proposals and method adaptation and verification, by comparison with TSF experiments. Future plans will include the performance and analysis of a prototypic grid plate shielding experiment at the TSF.

In other work, partially sponsored by DNRA and partially sponsored by the Division of Nuclear Fuel Cycle and Production, we have begun reactor physics studies of alternate fuel cycles that may mitigate the weapons proliferation threats through the theft of plutonium. In a related DNRA study, the feasibility of employing "denatured" reactor fuel that would not be appropriate for weapons use has been explored. Two other "safeguard-type" activities, both supported by DNFCP, have included calculations to aid in the design

NPD RESEARCH PROGRAM, OCTOBER 31, 1975 - FEBRUARY 28, 1977

Studies for ERDA Division of Reactor Development and Demonstration

Cross-section measurements for the fissile and fertile materials
Cross-section measurements for low- and medium-A nuclides
Cross-section evaluations and uncertainty studies
Cross-section processing
Core analysis method development
Shield analysis method development
Sensitivity analysis method development
Core neutronics studies for fast reactors
Shielding analyses for FFTF and CRBR
Shielding experiments and analyses
Waste management studies
Nuclear safety data base
RSIC-LMFBR studies

Studies for ERDA Division of Nuclear Research and Applications

Gas-Cooled Fast Reactor shielding experiments and analyses
Analyses of GCFR shield designs
Thorium assessment studies
Thorium and uranium fuel cycles in fast breeder reactors

Studies for ERDA Division of Nuclear Fuel Cycle and Production

Gas-Cooled Reactor studies
Thorium utilization studies
Safeguards calculations
Actinide transmutation calculations

Studies for ERDA Division of Magnetic Fusion Energy

Plasma modeling
Neutronics studies for EPR, TNS, DPR, and EBT
MFE integral shielding experiments and analyses
RSIC-MFE activities

Studies for ERDA Division of Physical Research

Transport calculations for high-energy particles
Ionization calorimeter design
Accelerator breeding studies
Cross-section measurements and evaluations for fusion and fission reactors
Studies of neutron capture in actinide wastes
ORELA operation and improvements

Work Supported by ERDA Division of Biomedical and Environmental Research (with the HEW/FDA Bureau of Radiological Health and the Society of Nuclear Medicine)

Biomedical Computing Technology Information Center (BCTIC)

Studies for Nuclear Regulatory Commission

Fission-product decay heat studies
Fission-product transport calculations - UO_2 fragmentation calculations
RSIC-NRC activities

Work Performed for Defense Nuclear Agency

Cross-section measurements and evaluations
Cross-section processing and library development
Radiation transport code maintenance
Integral experiment analysis
Transport calculations
RSIC-DNA activities

Other Studies

RSIC survey of shielding needs of nuclear power industry for EPRI
Cross-section sensitivity analyses of thermal reactors for EPRI
TFTR neutronics calculations for Princeton University
Preliminary calculations of atmospheric diffusion of particles released at low altitudes (ORNL Exploratory Studies Program)
Advanced fuels sensitivity studies (ORNL Exploratory Studies Program)
Improvement of tank modeling code for Ballistics Research Laboratory

of techniques for assaying the fission content of spent fuel assemblies and calculations to aid in the development of techniques for transmuting actinide wastes into less offensive nuclides.

Our program under ERDA's Division of Magnetic Fusion Energy has recently been expanded to include responsibility for carrying out the integral measurements required to validate neutronics calculations for the blankets and shields of the nation's proposed magnetic fusion energy devices. The other studies in this program, plasma modeling and neutronics calculations for specific devices, have continued, as have several MFE activities in RSIC.

One of the longest standing programs within the Neutron Physics Division is sponsored by ERDA's Division of Physical Research. Within it falls the continuing calculations of the transport of high-energy particles, usually performed to aid specific accelerator designs, and the design of ionization calorimeters that measure the energy of various high-energy particles. DPR also sponsors measurements and evaluations of cross sections of interest to fission and fusion reactor programs, and to it recently has been added measurements carried out in a cooperative program with the ORNL Physics Division to study neutron capture in actinide wastes. In addition, the operation and improvements of the Oak Ridge Electron Linear Accelerator (ORELA) come under this program. Also under DPR funding, we have recently performed a study on using an accelerator for the breeding of fissile material, which could develop into a large program.

The ERDA Division of Biomedical and Environmental Research (DBER) also supports one of our programs. DBER, with assistance from the HEW/FDA Bureau of Radiological Health and the Society of Nuclear Medicine, sponsors the Biomedical Computing Technology Information Center (BCTIC) which was established within our division in 1975 and operates in conjunction with but separate from RSIC. BCTIC and RSIC maintain their own technical staffs, but in order to most efficiently and economically provide the administration of the two centers, we have transferred the administrative staff organized under RSIC into a Technology Resource Group (TRG) which serves as an administrative umbrella for both centers. Under this umbrella will be added in 1978 a third center under the auspices of the Nuclear Regulatory Commission (NRC). Called the NRC Measured Data Repository (NRC-MDR), this center will assume responsibility for handling the results of NRC's LOCA Heat Transfer Testing Programs. We also expect to add another NRC center which will be concerned with computing contract support. In the meantime, feasibility studies for this center are being carried out by RSIC under funding from NRC.

NRC also continues to support a program within the Division to measure the heat associated with the decay of fission products. In addition, we have during the past year developed for NRC a calculational model for analyzing a series of reactor safety experiments.

During the past decade the Division has performed a number of studies for the Defense Nuclear Agency which by its support has greatly accelerated the development of radiation transport calculational techniques and data used so widely in reactor shield analyses. DNA continues to support the maintenance of a cross-section library in RSIC and the maintenance of the Monte Carlo code MORSE. In addition, we have an on-going study for DNA in which the transport of radiation from weapons bursts is calculated.

During this reporting period we have also performed two studies for the Electric Power Research Institute (EPRI); RSIC made a survey of the radiation protection and shielding needs of the nuclear power industry, and our sensitivity analysis group performed a number of cross-section sensitivity calculations for thermal reactors. At the request of and with the support of Princeton University, an ERDA contractor, we also carried out a series of neutronics calculations for their Tokamak Fusion Test Reactor (TFTR). And for the U. S. Army Ballistics Research Laboratory, we made some improvements in a tank modeling code that we had developed earlier.

This reporting period has also seen the initiation of two studies under ORNL's Exploratory Studies Program. One is a cross-section sensitivity study which has been initiated with a calculation for the LMFBR advanced fuels critical experiment at ZPR-9. The other is an attempt to develop models calculating the diffusion of particles released at low altitudes through the atmosphere to distances as great as 150 km. Both of these studies have the possibility of developing into important programs.

Finally, there is a possibility that we will soon be engaged in a program to develop models for calculating one of the environmental effects of an increased use of coal as an energy source. An anticipated problem associated with such an increase is the buildup within the atmosphere of CO_2 which would allow energy to enter the atmosphere but would prevent it from escaping. Thus the average temperature of the atmosphere would increase with possibly catastrophic results.

From the above it is obvious that the Neutron Physics Division is engaged in a number of programs sponsored by several agencies and that many of the programs have overlapping areas. This, of course, is mutually beneficial to the various sponsors, and in some cases has allowed an expansion of several of the research areas summarized below.

Research Areas

Measurements of Neutron Cross Sections and Related Quantities; Cross-Section Evaluations

The Division's program for measuring neutron cross sections and related quantities at ORELA (the Oak Ridge Electron Linear Accelerator) has continued, both for the fertile and fissile materials important to fission reactor programs and also for nuclides important to fission and fusion reactor shielding. Some of the work is described in the papers reported in Section 1 of this report, but other similar studies have not yet reached the reporting stage.

Included in this research area have been experiments to measure the ratio of the fission cross section of ^{238}U to the fission cross section of ^{235}U. These measurements are now complete and together with other experiments performed in the recent past have resulted in a remarkably coherent body of data which it is believed both will satisfy fast reactor needs for ^{238}U fission cross sections and will give a detailed high-resolution exhibition of the subthreshold fission resonance behavior. We have found a substantial amount of fine structure in the region between 6.7 eV and the fission plateau in the MeV range. In the region of most practical importance, greater than a few hundred keV, two or three experiments, including ours, agree conclusively and even at low-energy regions, where there is less overlap of the measurements, no disagreements are apparent.

Measurements of the transmission of neutrons through ^{238}U samples having a large range of thicknesses, performed at the 150-m station of ORELA, have also been completed and a resonance analysis has been performed through 4 keV. It is believed that this analysis using the several sample thicknesses is one of the most comprehensive ever performed. A concurrent effort in Belgium has also provided a rich new base of experimental data on neutron transmission through ^{238}U. With these data, a new evaluation of resonance parameters we are performing for the Version V release of the Evaluated Nuclear Data File (ENDF/F-V) will be a very soundly based and definitive job.

Other measurements completed are those begun in 1975 to measure $\bar{\nu}$ of ^{239}Pu and ^{235}U in the region below a few electron volts. Additional measurements of $\bar{\nu}(E)$ will focus on the region up to several MeV. For these new measurements we have built a second scintillation tank with some improvements over the old one, and later we will attempt to measure its absolute efficiency to permit a redetermination of the spontaneous fission of $\bar{\nu}$ for ^{252}Cf.

An experimental effort to determine the capture and fission cross sections of some of the actinides with $A > 239$ has progressed, with papers on ^{241}Am and ^{240}Pu having already been published and a paper on ^{241}Pu undergoing review. We are performing the ENDF/B-V evaluations for ^{240}Pu and ^{241}Pu. For these evaluations we are collecting data from around the world, and in an attempt to rectify some of the long-standing uncertainties in the fission cross section for ^{240}Pu, which displays marked subthreshold resonance behavior, we ourselves have recently conducted additional measurements for ^{240}Pu fission with a new fission chamber, but the data have not yet been analyzed. Work is continuing on the capture cross section of ^{242}Pu and will soon begin on fission of ^{241}Am. It appears likely that the existing accuracy goals for ^{241}Pu and ^{241}Am capture have now been satisfied, while the full adequacy of the remainder of the cross sections coming from this portion of the program remains in doubt. Work that had been planned on ^{242}Pu has been delayed because the sample obtained for the measurements (a borrowed one) was not good enough.

In order to continue helping solve the cross-section needs for assessing production and burnup of these so-called actinide waste nuclides, we formed the cooperative program with the ORNL Physics Division mentioned above to obtain fission and capture cross sections for selected isotopes of the various elements. Our main effort will be centering on capture measurements, initially on ^{243}Am and ^{237}Np.

In other studies, the first series of high-resolution measurements of neutron elastic and inelastic scattering from Fe, Na, and Si over the energy range from 30 keV to 2.5 MeV have been completed. Analyses of the elastic scattering data using R-matrix theory has resulted in the assignment of spins and parities of the resonances up to several hundred kilovolts. This information will help reduce large uncertainties in self-shielded capture cross sections and promises to allow us to reach the required accuracy in capture data for transport applications.

Inelastic cross-section measurements for ^{238}U are currently under way, the technique used being to observe the de-excitation gamma rays with high-resolution detectors. Recent studies have shown very high sensitivity of calculations of critical experiment parameters to inelastic scattering from ^{238}U in the region of 600 kV to 2 MeV.

We are also continuing in our role as ENDF/B evaluators for elements other than those mentioned above. We have just completed re-evaluations of lead, calcium, and carbon, and we are now in the process of re-evaluating iron. In some cases we are making measurements to obtain a firmer experimental base for the evaluations, all of which are being prepared for the Version V release of ENDF/B. Work on copper will begin in July.

We are also preparing updates of the evaluations of sodium, silicon and fluorine. Total cross-section measurements have been made at ORELA for all three of these elements, and in addition gamma-ray-production cross-section measurements have been made for sodium. The sodium measurements, especially the gamma-ray production measurements, were prompted when in the course of an analysis of a Tower Shielding Facility experiment (see below) it became obvious that there were deficiencies in the cross-section data available for this element.

In addition to the work on the general-purpose ENDF/B files, we are also involved in the preparation of some of the ENDF/B special-purpose files, in particular, the standards files, the gas production files, and the dosimetry files. The work on standards has been for carbon, which will be a neutron scattering standard. We have especially concentrated on evaluating differential elastic scattering for this element in the energy region between 0 and 2 MeV. The dosimetry work has been on the ^{23}Na(n,γ) and ^{56}Fe(n,p) reactions, and the gas-production work is aimed at all reactions that release protons (hydrogen gas) or alpha particles (helium gas). Currently we are investigating the isotopes ^{56}Fe, ^{58}Fe, ^{12}C, ^{63}Cu, ^{19}F, and all the silicon isotopes.

We also continue to maintain the Defense Nuclear Agency's Working Cross Section Library, currently in ENDF/B Version IV format. This library now includes evaluations for 27 materials, and new materials are being added in the ENDF/B Version V format. Similar activity is under way for an evaluated library to meet the needs of ERDA's Division of Magetic Fusion Energy, and some data in ENDF/B-V format are now available. Evaluations from both libraries will be submitted to the National Neutron Cross Section Center for inclusion in its Version V release of ENDF/B, which is expected in 1978.

In connection with the cross-section evaluations, we are also continuing our studies on cross-section uncertainties for nuclides important in the reactor development programs. We have reviewed and improved the covariance files for ^{239}Pu capture and fission and have generated new comprehensive files for fission in the uranium isotopes. An extended method of "external uncertainty analysis" has been developed which, conceptually, reflects the covariance information for the evaluated cross sections themselves rather than for the dispersion of the measurements, and additional extensions have been made in developing uncertainty files for the low-resonance region in ^{238}U. For those cases for which the ENDF/B-V evaluation is being performed at ORNL, our technique will surely be used to generate uncertainty files for the evaluation. In other cases we stand ready to serve a cooperative role and will offer the files we have developed for the consideration of the corresponding cross-section evaluators.

The long-standing goal to develop a sufficiently complete library of responsible uncertainty quantities has not yet been attained. Before the work already done can be combined with existing sensitivity analysis tools to yield sensitivity analyses of reactor systems, it will be necessary to obtain uncertainty files for several nuclides, at least for inelastic scattering and fission spectral shape. In some cases it will be necessary to repeat the initial uncertainty evaluations.

The responsibility for generating the complete format and procedures for the ENDF/B-V data covariance files also lies within the Division, the chairman of the Data Covariance Committee of the Cross-Section Evaluation Working Group (CSEWG) being a Division staff member. The status of this work is that the formats and procedures proposed by the Committee are now being reviewed by various members of CSWEG, and once they are approved they will be issued as a formal document. In the meantime all of the uncertainty information being developed within the Division for ENDF/B-V is being cast in the proposed formats and procedures.

From the above discussion, it is obvious that the Division performs a number of experiments at the ORELA Facility. As has been pointed out in earlier progress reports, this facility is shared with the ORNL Physics Division, and in the previous reporting period the two divisions submitted a proposal to ERDA's DPR Accelerator Improvement Program to install a series of prebunching devices between the electron gun and the accelerator proper to "bunch" electrons in a shorter burst width. The proposal was accepted and the initial design of the prebuncher system is complete. We are now entering the final design and procurement phases. In the meantime, calculational efforts have continued to model the longitudinal behavior of the beam, and new calculations have been begun to determine the radial behavior of the beam. The results will be used in the final design to ensure that the electron beam confinement is optimized.

Another series of measurements which is not performed at ORELA is the program initiated at the Oak Ridge Research Reactor (ORR) in 1975 to determine the amount of heat associated with the decay of fission products. This program has progressed to the point that final measurements have been made of the beta and gamma-ray decay heat following the thermal fission of ^{235}U to a precision of $\sim$3-1/2%, and work has started on similar measurements for ^{239}Pu. The data obtained are spectra of gamma rays, and also beta rays, from which the total decay heat is deduced, and the results are used for safety and accident analyses for light-water reactors. The spectra also provide an experimental base against which calculations performed with the ORIGEN isotope generation and depletion code can be checked. The comparisons are being used to aid the CSWEG fission-product subcommittee in identifying those isotopes that are important in decay-heat measurements.

In addition to our own work on decay from ^{235}U fission heat, there have been a number of similar measurements in the United States and other countries, and the results are all in sufficiently good agreement that the ANS Standard Subcommittee 5.1 is in the process of making a decay heat power standard for thermal fission of ^{235}U in uranium-fueled light-water reactors. Experimental groups whose recent work has contributed to this body of information are located at Los Alamos and Intelcom Rad Tech and in France.

Cross-Section Processing, Validation, and Sensitivity Analysis

During 1976 reports describing the AMPX neutron-gamma multigroup cross-section processing code and the MINX neutron cross-section processing code were published at ORNL and Los Alamos Scientific Laboratory, respectively, with one of the NPD staff members being a primary author of the MINX report. Both these codes had been under development and in use for several years, and, in fact, were used by us in combination (MINX for neutrons and AMPX for gamma rays) to produce a coupled master cross-section library consisting of 171 neutron groups and 36 gamma-ray groups. This library covers a large number of materials of interest to both the MFE and the RDD Divisions of ERDA. Based on

ENDF/B-IV data and variously referred to as the CTR library, the PMCSL, and more recently as VITAMIN-C, the library is being subjected to a systematic validation procedure in which at least eight different organizations are participating. The library is now available from RSIC's Data Library Collection (DLC-41) and is being presented as a possible standard for DMFE blanket design and LMFBR core and shield problems.

Two subsets of the 171-36 library have also been developed and are in use, both of which will be released through RSIC in the near future. One is a 126-36 set tailored primarily for LMFBR calculations, and the other is a broad-group 45-16 set which can be used on relatively small computers. Recently, EPRI has requested that the 45-16 set be tested as a possible standard for light-water reactor shielding calculations.

In conjunction with the 171-36 cross-section set and its subsets, we have developed a number of utility codes which permit the users to select options for self-shielding, Doppler broadening, group collapse, etc. This activity is continuing, work already having begun on updating the libraries with ENDF/B-V data. The updated version will include covariance files and new data types such as activation cross sections, delayed radiation production, gas production data, etc. Recently we have used the ORIGEN code to generate multigroup time-dependent delayed gamma-ray spectra from fission and activation of uranium and plutonium. Multigroup cross-section files for the ORIGEN actinide libraries have also been updated based on processed data from recent re-evaluations in ENDF/B format. We have also published kerma factors and recoil spectra.

Work on other cross-section sets has also progressed. The EPR coupled library consisting of 100 neutron groups and 21 gamma-ray groups, developed for ERDA-DMFE work at ORNL, has undergone its second modification, and the few-group DNA library (37 neutron groups and 21 gamma-ray groups) that has been in use for some time was officially documented.

Testing of the adequacy of the ENDF/B-IV data for shielding applications through their use in calculations that analyze TSF experiments has also been carried out. In addition, we continue to be active participants in the cross-section testing program of the CSEWG Shielding Subcommittee.

By now the FORSS system, introduced in our last progress report as a modular code system for studying the sensitivity of calculated parameters of full reactor systems to the cross sections used in the calculations, has undergone considerable further development and extension. As it was described previously, FORSS provided several options in the selection of codes that could be used successively to perform a sensitivity and uncertainty analysis of a given system. They included codes for fine-group cross-section processing, codes for collapsing the cross sections into broader groups tailored for the particular problem, codes for performing the transport (or diffusion) calculation to obtain the desired responses, and finally, a code for determining the sensitivity of the responses to specified changes in the cross section. The current version of FORSS has several additional modules which were developed to extend the system capability in the projection of uncertainties and the definition of cross-section needs. The INTEXP module, an extension of the Italian AMARA code, allows identification of the uncertainties on the data responsible for the uncertainties on the responses. Moreover, the calculated uncertainties and multigroup cross sections are adjusted on the basis of information gained from appropriate integral experiments. Another module added to the FORSS system, NUTCRACKER, permits the solution of the inverse problem: that is, a calculation to determine the accuracy with which nuclear data must be measured in order for it to be used successfully in calculating parameters of a given reactor/shield system to within the accuracy set by design criteria. Yet another module, SENTINEL, permits "on-line" projections of changes in response associated with cross-section re-evaluations. This tool is expected to be used in the evaluation and data testing of ENDF/B-V.

Now having a complete one-dimensional FORSS system, we are in the process of preparing a user's manual and writing a driver to link all the modules. If the automated system can then be shown to reproduce results already obtained with the successive use of the individual modules for a sample problem (the analysis of GODIVA), it will be released for distribution through RSIC. During the past year, the one-dimensional system has also been applied to a series of calculations for a thermal reactor of interest to the Electric Power Research Institute, and to several fast reactor benchmarks designated by the Cross Section Evaluation Working Group (CSEWG). The most recent application of the system has been in the assessment of the impact of uncertainties in standard cross sections as they propagate through other cross sections (for example, by ratio measurements) in the analysis of reactor systems.

At the same time FORSS is being extended further. It is being developed in both diffusion and transport modes for two-dimensional core physics problems, and, in fact, has already been applied to a realistic two-dimensional model of a large LMFBR system. In addition, efforts to develop a time-dependent system are under way.

The application of the recently developed "channel theory" in the FORSS system is also being worked on, not only with respect to the contributons (particles that contribute to the response) flowing through channels in space but also with respect to their flowing through channels in energy. The equations developed for the contributons look very much like fluid-flow equations, and we are currently attempting to develop a theory for the contributon field. Once channel theory is incorporated into FORSS, the system will yield three-dimensional plots showing the spatial and energy regions allowing the contributons to reach the location of interest.

It also appears that FORSS sensitivity techniques developed for plasma physics problems relating to the interactions of neutral atoms are very similar to sensitivity studies required for light-water reactors relating to upscattering. Finally, we hope to extend our sensitivity work into the area of non-linear equations which will allow econometric modeling (energy systems analyses). Here the equations are very similar to those being used in the Division for reactor safety analyses (see below).

The applicability of the FORSS sensitivity analysis system to shielding experiments has been demonstrated since our last reporting period by one-dimensional sensitivity analyses of 21 different configurations of steel, sodium, and iron. Measurements of the neutron transmission and gamma-ray production and transmission in each of these configurations had been analyzed in two-dimensional geometry with the DOT discrete ordinates transport code using the 50-group LMFBR neutron cross-section set developed several years ago. In order to update these results, the configurations were recalculated with the one-dimensional code ANISN using both the 50-group set and the later DMFE-DRDD 171-group neutron set, and the ratios between the two were used to scale the two-dimensional fluxes. Considering the resulting ratios of these fluxes to the measured fluxes, which were in the range between 0.75 to 1.5, we are performing sensitivity analyses for each configuration to try to determine what causes the disagreement. By the time the first five configurations were completed, it was apparent that deficiencies existed in the sodium cross sections in the region of the 300-keV window and that the high-energy elastic and inelastic scattering data were suspect for both iron and sodium. Subsequent measurements at ORELA for sodium (see above) resulted in significant changes in the sodium cross sections which will be incorporated in the ENDF/B-V evaluation. These changes did much toward reconciling the experiments and their analyses, and if upon completion of the sensitivity analyses for all 21 configurations we observe a consistency between the experiments and calculations, we will be able to state with confidence that they represent the best set of two-dimensional steel-sodium-iron benchmarks available for shielding, especially since with the reactor source and detectors available at the TSF the entire energy range of interest is covered and the configurations used were sensitive to essentially all the sodium and iron cross sections.

Integral Experiments and Their Analyses

During this reporting period the analyses of several TSF experiments performed earlier to support the design of the CRBR were completed. Among these was a "three-dimensional stored-fuel" experiment in which the region between the CRBR core and the reactor vessel was mocked up, including a region located outside the radial shield for the temporary storage of fuel assemblies. The primary purpose of this experiment was to determine the enhancement of the fast-neutron fluxes outside the vessel wall due to fissions in the stored fuel. Other analyses completed were for an experiment in which an early design of the lower axial shield for the demonstration plant was tested and for experiments performed to test the neutron total cross sections and the gamma-ray-production cross sections included in the ENDF/B files for several materials.

In addition, the initial analysis of earlier experiments performed to study radiation heating in iron and stainless steel configurations simulating the CRBR radial shield was concluded and published. However, in this analysis the agreement between the measurements and calculations was considered to be poor; moreover, the agreement between the thermoluminescent dosimeters (TLDs) and the ion chamber used in the experiment was not good enough, nor even the agreement between the two types of TLDs. (The CaF_2 TLD and the LiF TLD differed by as much as 20%.) Since our goal had been to measure the heat within an accuracy of 5% in order to meet the criterion of Westinghouse/ARD that they know the heating within 10% with a 95% probability, it was mandatory that the discrepancies be resolved. We spent several months examining our measuring techniques by comparing the two TLDs and the ion chamber in pure gamma-ray fields produced by four different sources. Through modifications indicated by these studies, we were able to improve our techniques to the point that for the pure gamma-ray field all three instruments agree within 5% with a standard deviation of less than 2%. We then performed additional measurements at the TSF using only stainless steel in the configurations. The result was that the heating measurements and calculations, the latter employing the latest cross-section values, as a function of the stainless steel thickness agree to within 5%. At relatively large distances beyond the configurations, they still disagree by as much as 25%, but this is attributed to inadequacies in the calculations method for such large thicknesses.

Another experiment which was initiated during the last reporting period but extended into the current one employed configurations that mocked up the upper axial shield of the CRBR. In the first series of measurements for this experiment, the mockup consisted of 45 cm of stainless steel followed by up to 457 cm of sodium and up to 61 cm of carbon steel. Later design changes in the CRBR prompted a second series of measurements in which the stainless steel thickness was reduced to 30.5 cm and the carbon steel thickness was increased to 89 cm. This experiment has undergone rigorous analysis which included sensitivity studies that in turn prompted remeasurements of sodium cross sections (see discussion above).

Two other CRBR experiments have been performed in recent months but have not yet been analyzed. One was an investigation of the neutron streaming that could be expected through and around a sodium coolant pipe from the reactor to regions near the heat exchanger. In the experimental configuration a steel pipe having two 90-deg bends and filled with sodium carbonate and surrounded by Kaylo insulation passed through four concrete "chokes" (concrete walls of various thicknesses) and neutron measurements were made at various locations along the passageway with and without the pipe in place.

The other unanalyzed CRBR experiment was performed to determine how much the thickness of the radial shield could be reduced if inconel were substituted for stainless steel, the reduction being necessary if a parfait core design is adopted. In the experiment measurements were made behind 13-, 25-, and 38-cm thicknesses of the two materials, and the results indicated that using inconel instead of steel would reduce the dose to one-half of that realized if stainless steel were used.

The analysis of a prototypic experiment performed at the TSF for the GCFR has also been completed. This experiment was designed to investigate neutron streaming through coolant channels in a region similar to that from the GCFR core to the grid support plate located above the core. The analysis indicated that neutron streaming in the lattice had been observed in the experiment and it could be bracketed with two-dimensional discrete ordinates calculations, the homogeneous models giving lower estimates and the heterogeneous models giving upper estimates. Monte Carlo calculations were applied to come considerably closer to the experimental results, although the Monte Carlo calculations are more difficult to set up and to perform.

We are now designing, through analysis, a prototypic GCFR grid plate shielding experiment to simulate the neutron transport from the core up through the axial blanket and through the grid plate shield to the grid plate. Its purpose will be (1) to verify our ability to calculate the streaming out of the core, (2) to verify the effectiveness of the grid plate shield in its current design, and (3) to verify our ability to calculate the effectiveness of the grid plate shield. It is expected that this will be the first experiment to remove the TSF from its standby status.

In the meantime, the Division is initiating another experimental shielding program. As part of the ERDA-DFME Program, we have been given the responsibility for providing the data, methods, and prototypic measurements required for verification of the neutronics design of the shield and blanket for the TNS (the next step), which is expected to be the first break-even fusion device built. (The conceptual design for the TNS is being developed in a joint program by General Atomic and ORNL, with the final design scheduled for 1982.) The experiments will be similar to integral experiments which were performed by the Division in earlier years for DNA (see papers 4.16-4.18), but will be on a larger scale. The neutron source will be provided by the d(T,^{4}He)n reaction, with the deuterons produced by a 300-keV generator housed in a relatively small concrete-walled room. The work to date has consisted in doing some preliminary shielding studies in the room and performing preanalyses to design the experimental configurations. The neutron attenuation is expected to be on the order of 10^9, of which 10^5 will be geometric attenuation and 10^4 will be material attenuation. We will use NE-213 spectrometers and detectors that respond over a broad energy range (Bonner balls). The experiment the first year will be a material attenuation experiment, using a generic toroidal shield configuration of laminated iron and hydrogenous material (borated polyethylene representing water). Subsequently we will perform neutron streaming experiments, since the many holes that will be present in fusion reactor shields will probably comprise the dominant shielding problems.

Development of Methods for Shield and Reactor Analyses

The division's current development of shield analysis methods continues to consist of long-term efforts on the maintenance and improvement of the existing discrete ordinates and Monte Carlo transport codes, specifically the DOT and MORSE codes. As has been pointed out previously, a major effort has resulted in Version IV of the DOT code, which 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. The latter feature will allow streaming problems to be calculated at much lower cost than has been possible previously. DOT-IV is now fully operational and is being used regularly within the division. It should be available for LMFBR contractors during 1977.

In the meantime, a new version of the DOT-III code, called DOT 3.5, has been released. It incorporates some of the features of DOT-IV, and, in fact, was developed primarily as a test vehicle for these features.

During the past year we have also published a users' guide for the version of the MORSE code that substantially reduces the computer core size that must be used for solving problems with large cross-section storage requirements. Called MORSE-SGC (for Super Grouped Cross Sections), this version of MORSE also includes an improved combinatorial geometry package and a computer code that transforms ANISN cross-section libraries to the AMPX working library format.

Other work on Monte Carlo transport has resulted in an improved technique for importance sampling in deep-penetration problems. This study has shown that for some cases expressing the importance of a particle in terms of its entering a collision event (event-value function) rather than in terms of its leaving an event (point-value function) results in substantial variance reduction.

Another very important factor in shield development methods has been the application of spatial channel theory, which is a technique that reveals the channels through which particles flow to contribute to the calculated response. Channel theory is based on the observation that only a portion of the particles emitted from a source actually contribute to the response and further that of the particles that contribute, called contributons, some are more important than others. In channel theory the contributons are not only identified, but their course through the shield is charted. Channel theory has already become a standard part of the ORNL shielding program.

In the development of core analysis methods, Version III of the diffusion theory neutronics code VENTURE has been released and is available for use on the ORNL computers, both locally and via remote terminals by General Electric and Westinghouse/ARD. Application testing of another neutronics code has allowed selection of a difference formulation for application to complicated geometries that makes use of the linear finite element approach in part. Rather elaborate exposure capability is now available in the code module called BURNER which was developed to apply a variety of techniques to solve the chain equations and generate auxiliary results. These and related codes are used in a simple modular code system which allows a reactor history to be followed. The effort continues toward enhancing this analysis capability, to satisfy the needs which are identified in application, and to satisfy the requirements for documentation and user instructions.

In other work a finite element formulation utilizing a canonical form of the transport equation was developed to obtain approximate numerical solutions to neutron transport problems. The formulation is based on the use of linear Lagrange-type polynomials and a general treatment of anisotropic scattering was included by employing discrete-ordinates-like approximations. In a related but separate formulation, quadratic Lagrange-type polynomials were employed to span the spatial domain. Linear anisotropic scattering was treated by removing the angular variable using a second-order spherical harmonic expansion.

Finally, a technique has been developed which is less complicated than those offered previously for calculating the reactivity changes resulting from the collapse of a large number of small bubbles in an LMFBR core. In this technique the collision sites in the bubble system are reduced to a set of cross sections and their probabilities, and the problem, which is stochastic in nature, is solved with a deterministic calculation (see dicussion of application below).

Analyses for Specific Systems or Applications

The analyses performed by the Division for specific systems or applications have been many and varied, as would be expected by the diversified interests of the sponsoring agencies. In addition to the calculations of experiments discussed above, the analyses have included radiation damage, neutronics, and plasma modeling studies for specific fusion energy devices; design calculations for ionization calorimeters; several calculations of high-energy particle transport; calculations for fuel-cycle alternatives; and neutronics and shielding calcualtions for fission reactors; plus other special studies.

Among the calculations for fusion energy devices was a cross-section sensitivity study that gave the uncertainties in heating and radiation damage responses in a toroidal field coil of a tokamak experimental power reactor that were due to uncertainties in the iron and carbon cross sections used in the calculations. The results indicated a high sensitivity in the energy group containing the 14-MeV source. In addition, a calculation was made of the spatial variation of the damage energy and gas production in the experimental volume of a Li(D,n) neutron radiation damage facility, and to aid experimentalists in assessing radiation-damage effects due to heavy recoil charged particles produced by different neutron environments, specifically but not exclusively those in fusion energy devices, a recoil data base developed during the previous reporting period has now been made available.

The neutronics calculations performed for the fusion energy program have been concerned with four different devices: the EPR (Experimental Power Reactor); the TNS (The Next Step), which has replaced the EPR; the EBT (Elmo Bumpy Torus); and the TFTR (the Tokamak Fusion Test Reactor to be built at Princeton University). For the EPR, TNS, and EBTR one-dimensional neutronic and photonic calculations have yielded comparisons of the nuclear performance of several proposed blanket and shield designs. For the TFTR similar calculations were made to obtain preliminary estimates of some of the effects of radiation on the operation and maintenance of the neutral beam injector. The radiations considered were the 14-MeV neutrons produced by D-T reactions in the plasma and 2.6-MeV neutrons produced by D-D reactions in the calorimeter and the charged-deuteron beam dump. This study is being continued in more detail with both two-dimensional and three-dimensional calculations.

Another study in this program has been a Monte Carlo analysis to estimate the effects on the performance of a D-T Tokamak fusion reactor that penetrations through the blanket-shield assembly would have. A great many of these penetrations will exist in the assembly, some of them very large, and radiation streaming through them could result in excessive heating and radiation damage in vital components, particularly in cyrogenic TF coils. (See above discussion of fusion energy experimental shielding program recently begun.)

The plasma modeling studies during this reporting period have mainly consisted of a completion and publication of a calculation of the interaction of hydrogen atoms within a typical ORMAK plasma.

In another area, a series of calculations performed with the HETC code (High-Energy Transport Code) to aid in the design of ionization calorimeters that measure the energy of various high-energy particles has continued from earlier years. 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. Originally iron had been the preferred absorber, but experiments recently completed at CERN in Switzerland and calculations performed by us have shown that ^{238}U could be used as the absorber with liquid argon as the dielectric, the advantage being that fast fission 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 was anticipated that pulses from incident hadrons will then be approximately equal to those from incident electrons.

The Division's calculations of the transport of high-energy particles are essentially all done on request from other groups. Examples are a series of discrete ordinates calculations carried out to aid in the design of the shielding for the experimental areas if the 400-MeV electron accelerator at the Laboratory of Nuclear Science at MIT is upgraded to 900 MeV, and recently initiated calculations to determine the concrete shielding needed around beam dumps for the proposed 200-GeV proton storage rings at Brookhaven National Laboratory (ISABELLE) to rapidly reduce the radiation from induced activity in the beam dumps to an acceptable level ($\sim$15 min) after the 200-GeV proton beam has been dumped. In addition, kerma (kinetic energy release in matter) factors have been obtained for neutrons in the energy range 20 to 70 MeV in tissue to aid in the use of neutrons in cancer radiotherapy at the Fermi National Accelerator Laboratory, and differential cross-section data have been calculated with the intranuclear cascade code MECC-7 for comparison with experimental data obtained at Michigan State University. We have recently extended our capabilities for calculating intranuclear cascades by completing the development of the high-energy code HECC which allows us to go above the MECC-7 code's $\sim$3-GeV energy limit.

The high-energy code HETC played an important role in a preliminary study of the possibility of developing an accelerator breeding and converter reactor symbiotic system (ABACS) as an alternative to fast breeder reactors. This study, led by our Division staff and including the participation of members of other divisions, included target physics studies which required calculations of the interactions of protons going into the primary target to produce neutrons and of the neutrons going into a blanket of depleted uranium to produce plutonium (or into thorium to produce ^{233}U). Another part of this study performed within our Division was the consideration of several types of converter reactors, with the mass flows and economics of the complete symbiosis presented. The ABACS study was prompted by the current national concern over the proliferation of nuclear weapons and terrorist activity through the theft and use of plutonium, and particular attention was addressed to the implementation of the ^{233}U-^{238}U denatured fuel cycle as a deterrent.

In a closely related study, we have examined the feasibility of using the denatured ^{233}U-^{238}U fuel cycle in conjunction with an energy park. The envisioned scenario is that a restricted area will be established for the location of fuel reprocessing and fabricating facilities and also for converter reactors (recently referred to as "transmuter" reactors) which produce ^{233}U in thorium while consuming plutonium. Reactors outside the restricted area would operate on the denatured fuel which would be returned to the restricted area for reprocessing and removal of the plutonium, which then would be fabricated as fuel for a converter reactor. Several other studies are also under way to assess the possible uses of thorium in fuel cycles.

The Division's core analysis code VENTURE has also been applied to several problems during this reporting period. One is a large-scale three-dimensional light-water reactor benchmark problem originally defined by the 1971 IAEA panel on burnup physics and subsequently adopted and adapted by the Benchmark Problem Committee of the ANS Mathematics and Computation Division. VENTURE has produced a reference solution to this problem for testing other codes.

VENTURE is also one of three finite-difference codes that have been applied to an ANS benchmark problem defined for a high-temperature gas-cooled reactor. A simplified model of the Fort St. Vrain High Temperature Gas-Cooled Reactor (HTGR) was chosen, with solutions to be obtained both in two dimensions and in three dimensions. Thus far solutions have been obtained for the two-dimensional problem, with the results from the three codes in good agreement.

Finally, VENTURE is being applied to problems defined by the Large Core Code Evaluation Working Group of the Reactor Physics Branch of ERDA-DRRD. The initial problem is a "representative" 1200-MW(e) LMFBR in a two-dimensional radial geometry. Again the codes based on the same calculational method and using the same multigroup constants are in substantial agreement.

The Division has also performed several core neutronics calculations for the CRBR. Among these have been calculations on which to base the development of methods for monitoring the initial core loading. This is a particularly acute problem because the count rate for the beginning-of-life core will be so low. Several monitoring methods have been proposed to the CRBR Project both by our division and the Instrumentation and Controls Division, and next year we expect to be engaged with the I & C Division in a cooperative program to explore in depth several techniques for initial loading.

In other neutronics work, we performed an analysis of the CRBR ex-vessel fuel-assembly storage tank with the Monte Carlo code CHERIE/KENO to determine how near criticality a new design proposed by Atomics International would be. The results indicated the conditions were sufficiently close to criticality to merit redesign of the tank.

In addition, we performed calculations to verify the absolute count rate of the CRBR source level flux monitor (SLFM) during reactor startup, and we carried out an analysis of an SLFM experiment on the ZPPR-5 at Argonne National Laboratory (ANL). In addition, we did an analysis of the uncertainties associated with the proposed use of the SLFM by Westinghouse/ARD. [This work is similar to that performed earlier for the Fast Flux Test Facility (FFTF) and published during the current reporting period.] During this reporting period, we also initiated studies on the reactivity associated with bowing of the CRBR core, such as will occur during startup; however, owing to design changes in the core these studies were postponed after only preliminary results had been obtained.

In related LMFBR reactor safety work begun by the Division in 1975, efforts have continued on the development of a Monte Carlo three-dimensional quasi-static kinetics code for use in the analysis of severely disrupted cores, but progress has been slow owing to unexpected convergence problems in the code.

Studies of the reactivity associated with the collapse of bubbles produced by stainless steel vaporizing in a medium of molten fuel have also continued using both discrete ordinates and Monte Carlo techniques. In the discrete ordinates technique the bubbles were modeled discretely, while in the Monte Carlo technique they were represented statistically. The Monte Carlo method provided upper limits on the reactivity that were lower than those produced by the Behrens' method, indicating that the Behrens' method, which can be applied more easily, is indeed conservative. This led to the development of our own simpler technique (see discussion above) which retains a statistical representation of the bubbles. Preliminary results obtained with this method, which has the advantage of being applicable to multiregion, multigroup and partial bubble zone problems, agree with those obtained with the Behrens' method.

In addition, the reactor safety studies have included the comparison of various methods for neutronics analysis of severely disrupted LMFBR cores. The bases of comparison included k_{eff} values and material worths. Preliminary results indicate that diffusion theory methods may be inaccurate unless close attention is given to mesh spacing. Studies have also begun in this area to investigate methods applicable to the analysis of parfait cores.

In another safety-related study we completed an initial series of calculations to aid in the analysis of a series of NRC reactor safety experiments to be performed at ORNL. In the experiments UO_2 samples will be fragmented by a capacitor discharge, and our calculations were performed to predict such quantities as the amount of gas and liquid produced, the number of liquid fragments, etc.

The Division also participates in a safeguard study in which we are continuing to aid in the development of a system for assaying the fissile content of spent LMFBR fuel assemblies in fuel reprocessing plants. The system conceived is a so-called "active interrogation" system in which neutrons are introduced into the fuel and the resulting fissions are detected, with the design precluding the development of critical conditions. Our initial work consisted of feasibility calculations for the system, and our current participation consists of calculations of the detector response for specified configurations. We are also doing calculations for a proof-of-principle experiment to be performed in the ORNL I and C Division.

In the area of radiation transport through shields, the Division has continued to apply the DOT discrete ordinates and the MORSE Monte Carlo transport codes to predict the penetration of reactor radiations through shields, and, to a lesser extent, to predict the transport of weapons radiations through the atmosphere. In the reactor shield analyses, channel theory has been included as an integral part of the calculations.

Our analyses for the FFTF were completed in 1976 with the reporting of two additional series of calculations: One was an extension of an earlier study to determine effects on the maintenance floor dose rates of pipe and duct penetrations through a concrete shield installed in the upper region of the reactor cavity, and the other was designed to determine whether radiation streaming upward through gaps in the head compartment shield would enhance the dose rates on the operating deck. Both calculations indicated a need for additional shielding.

The first three series of shielding calculations for the CRBR were also reported. The first series concentrated on the lower axial shield and indicated that a substantial reduction in the shield thickness could be realized. The second and third series were "full-assembly" series in which dose rates on the reactor head were calculated for several reactor cavity shield designs. Subsequent CRBR shielding work has centered on determining the adequacy of the SLFM for monitoring the beginning-of-life (BOL) core during shutdown. Since the SLFM will be located outside the reactor vessel with considerable intervening shielding and the inherent source of the BOL core will not have yet built up, it will be very difficult for the SLFM to "see" neutrons during shutdown.

In our support of the shielding analysis program for the 300-MW(e) demonstration model of the Gas Cooled Fast Breeder Reactor proposed by General Atomic Corporation (GA), we have performed calculations for a new reference model to analyze both the proposed radial shield and the lower axial shield. These have included two-dimensional cross-section sensitivity analyses with the SWANLAKE-VIP system and also channel-theory calculations that showed how neutrons from the core were transported to and into the PCRV (prestressed concrete reactor vessel). In addition, these calculations, which will be used by GA as a basis for new reference designs on the radial and lower axial shields, yielded gamma-ray heating and dose-rate information at the tendon position beyond the liner. Future efforts will be directed at establishing the GCFR neutron leakage spectra both at the bottom of the core and at the top of the grid plate. These, then, will become the sources of lower and upper axial shielding calculations, respectively.

Finally, the MORSE Monte Carlo code and the time-dependent ANISN code (TDA) have been used to calculate time-independent and time-dependent radiation fields produced in the atmosphere by weapons bursts for DNA. These follow earlier two-dimensional DOT calculations of the neutron and secondary gamma-ray fields, including cross-section sensitivity studies, produced by weapons at varying heights above the ground and sea water.

Information Analysis and Distribution

In addition to performing research per se, the Neutron Physics Division long ago recognized that in order to avoid unnecessary duplication of research efforts and/or lost time in library searches, it was mandatory that the results of research in a given field be organized in a usable and readily available manner. It was this recognition that prompted the establishment within the Division of the Radiation Shielding Information Center (RSIC) over a decade ago and the subsequent organization of the Biomedical Computing Technology Information Center (BCTIC) in 1975.

In the previous reporting period, it was announced that the Division was also initiating a safety analysis computerized reactor data base (SACRD) which would provide input for approximately 100 fast reactor safety analysis codes now in existence. In the ensuing period (about 1-1/2 years) the process of collecting and collating the data and putting it on ORNL-IBM machines has resulted in the handing over to a Safety Analysis Data Coordinating Group (SADCG) of Version 0 of the base, which contains thermophysical properties of materials important for safety analysis. The coordinating group, chaired by the member of our Division who leads the safety data base effort, is composed of representatives from industry, universities, regulatory bodies, national laboratories, and ERDA, and it is responsible for the evaluation and approval of the data released. The data eventually to be released will describe the physical properties of reactor core materials (e.g., thermodynamics, material strengths, heat transfer, mass transport) and also meteorological, chemical, dosimetry, neutronics, and aerosol transport data. In addition, SACRD will contain bibliographic and reference materials, files for commonly used correlations and variables, benchmarks for data testing, and files containing new but nonevaluated data.

The management system for the SACRD data base is the computer system JOSHUA that was developed at Savannah River Laboratory. JOSHUA is a modular code system consisting of approximately 150 separate codes linked together. Interactive terminal display capability under time-sharing operations is being implemented to facilitate easy user access to the data base. It is expected that Version I of SACRD will be released to users in late 1977.

A related but new data base which will soon be initiated by the Division is a Reliability Data base, which will include statistics on the reliability of various nuclear plant components as determined from experience with operating reactors. Specifications for this data base will be presented to ERDA on May 1, 1977, for review.

RSIC, now in its 15th year, continues to be our largest and most widely used information center. Established to promote the exchange of shielding technology, it acts as a resource base for both government and civilian agencies in the United States and in foreign countries and performs such diverse functions as providing bibliographic information; testing, assembling and distributing computer codes; preparing, testing, and distributing multigroup cross-section libraries, both fine-group and broad-group; collecting and distributing other types of data bases; holding seminars to educate the shielding community on particular techniques, especially computer techniques; helping to establish shielding benchmark problems and shielding standards; and generally providing problem assistance to requestors.

Results from some of the RSIC activities during the current reporting period have already been cited in this summary, in particular, the maintenance of an evaluated neutron cross-section file for the Defense Nuclear Agency; the development and testing, in collaboration with other Division staff members, of the DNA few-group library, the EPR 100-neutron and 21-gamma-ray group library, and the 171-neutron and 36-gamma-ray group VITAMIN-C library. The release to and through RSIC of other types of data bases has also been mentioned. RSIC has continued to participate in the work of the CSEWG Shielding Subcommittee, with the Subcommittee presently being chaired by an RSIC staff member. In addition, RSIC performed a survey of the radiation protection, radiation transport, and shielding needs of the nuclear power industry, and in April, 1976, conducted a seminar on radiation energy spectra unfolding techniques. One staff member has also been actively involved with ERDA in developing radiation protection and shielding standards from the overall management viewpoint. Another significant accomplishment during the year is that our bibliographic information (SARIS) is now on RECON, so that access to the information is now possible from all over the world.

In the area of computer codes, RSIC published its fourth volume of ORNL/RSIC-13, which includes abstracts for code packages CCC-169 through CCC-263. RSIC's insistence on an "open-code" policy has done much toward facilitating the exchange of computer codes. A similar stance on data exchange has also been taken.

During 1976, more than 4000 letter requests to RSIC resulted in 9922 separate activities (31.5/working day), which is a 26% increase. The center also filled 432 SDI requests (Selective Dissemination of Information), and mailed its monthly newsletter to ∿1600 recipients.

It is obvious that the Center's continued ability to fill the increasing number of requests without significant increases in their staff (the approximate doubling time in requests is 4 years) is to a large extent the result of the development of an efficient and supportive administrative staff. Recognizing this, the Division during this past year effectively removed the administrative staff members from solely RSIC operations and placed them in an administrative "umbrella" identified as the Technical Resource Group. This group was then requested to perform similar functions for BCTIC.

As was pointed out in our last progress report, BCTIC was established in 1975 to collect, organize, evaluate, and disseminate information on computing technology in biomedicine in general and nuclear medicine in particular.

The past year has been a growth year for BCTIC, which currently has 16 code packages and 3 data libraries available for distribution. The BCTIC newsletter, published bimonthly, now reaches a distribution of over 1200 persons, as compared to only 700 in November 1975. In an attempt to acquaint the medical community with its operation, BCTIC staff members have given presentations to several national and international organizations, including the Society for Computer Medicine, the Association for Computing Machinery, and the International Atomic Energy Agency. In addition, personal contacts have been made at several biomedical computing installations, and over 850 requests have been received for information, codes, data, or other BCTIC services.

One of BCTIC's most significant accomplishments during the past year was the publication of its first volume of a directory of computer users in nuclear medicine (Report ORNL/BCTIC-1). This document is designed to facilitate communication among current and prospective users of computers in nuclear medicine, to identify areas of common interest, and to minimize duplication of effort.

Discussions are under way with several organizations, including the National Library of Medicine and the MUMPS Development Committee, to expand BCTIC's services to other areas of the biomedical community.

As would be expected, the availability of the Technical Research Group has provided impetus to the establishment of other information centers within the Division. During the past year negotiations with the Nuclear Regulatory Commission led to an agreement for setting up an NRC Measured Data Repository (NRC-MDR) which will go into operation in 1978. This center will be an integral part of NRC's LOCA Heat Transfer Data Bank. Its functions will be to accept raw measured data from the Blowdown Heat Transfer Testing Programs, after the data have been placed in a common format elsewhere, and thus to provide a common source of raw data for NRC's regulatory and safety research areas and for the nuclear power industry.

In the meantime, a feasibility study is being performed within RSIC on the possibility of establishing a second NRC center under TRG. This center would be a computing contract support center which would accept, test, and distribute codes for safety analyses of nuclear fuel shipping casks. In the feasibility study, RSIC has already accepted the responsibility for disseminating the multigroup Monte Carlo criticality code KENO-IV. It will, in addition, work with the user community and code developers to advance the technology in this field.

Lorraine S. Abbott

1. MEASUREMENTS OF CROSS SECTIONS AND RELATED QUANTITIES

1.1 MEASUREMENT OF NEUTRON TRANSMISSIONS FROM 0.52 eV TO 4.0 keV THROUGH SEVEN SAMPLES OF ^{238}U AT 40 m*

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R. W. Ingle‡ H. Weaver

(Abstract of ORNL/TM-5256, January, 1976)

Neutron transmissions through 1.5-, 5-, 10-, 30-, 100-, 425-, and 1425-mil samples of depleted ^{238}U were measured from 0.52 eV to 4.0 keV using the ORELA pulsed electron linac neutron source and time-of-flight technique with a 1-mm ^{6}Li-glass detector with a flight path of 40 m. The measurements are tabulated and compared with transmissions calculated from the ENDF/B-IV total cross section. In addition, the 1425-mil transmission from 50 to 300 eV is compared with transmissions calculated by using multilevel formalisms, and some neutron widths are extracted with area analysis and compared with those from previous measurements.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

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1.2 PRECISE MEASUREMENT AND CALCULATION OF ^{238}U NEUTRON TRANSMISSIONS*

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(Abstract of paper published in March, 1977, issue of Nuclear Science and Engineering)

The transmissions of 0.52- to 4000-eV neutrons through 3.62-, 1.08-, 0.254-, 0.0762-, 0.0254-, 0.0127-, and 0.0036-cm-thick samples of depleted ^{238}U have been measured at 42 m with a 1.0-mm ^{6}Li-glass detector using the Oak Ridge Electron Linear Accelerator pulsed neutron source. In order to obtain resonance parameters, these seven transmissions from 0.52 to 1086.8 eV have been simultaneously least-squares shape-fitted with a multilevel Breit-Wigner cross-section formalism with "picket fence" terms to account for truncation effects. This simultaneous fit yielded a χ^2 per degree of freedom near unity. Averaged over this energy range, an s-wave strength function of $0.968 \pm 0.036 \times 10^{-4}$ and an effective radius of $0.944 \pm 0.005 \times 10^{-12}$ cm were obtained. In addition these transmission data yielded an average radiation width of 23.1 ± 1.0 meV for the 12 lowest-energy s-wave resonances with radiation widths of 23.0 ± 0.8, 22.8 ± 0.8, and 22.9 ± 0.8 meV for the 6.67-, 20.9-, and 36.8-eV resonances, respectively. The derived radiation widths for these three resonances are shown to depend on the cross-section formalism employed.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

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1.3 RESONANCE PARAMETERS OF THE 6.67-, 20.9-, AND 36.8-eV LEVELS IN ^{238}U*

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The ENDF/B-IV ^{238}U cross sections (MAT-1262) yield an effective capture resonance integral in strongly self-shielded situations which is too high.[1] This situation suggests that the ENDF/B capture widths for the first few s-wave levels may be too large. Recent ORELA measurements[2] of transmission through ^{238}U have been analyzed with a multilevel formula to determine the parameters of the 6.67-, 20.9-, and 36.6-eV levels. These three levels provide 86% of the infinitely dilute capture resonance integral.

The data consist of transmissions through 3.62-, 1.08-, 0.254-, 0.0762-, 0.0254-, 0.0127-, and 0.0036-cm metal samples. These seven transmissions were simultaneously analyzed over many resonances with the least-squares computer code SIOB[3] which contains Gaussian resolution and Doppler broadening and employs a multilevel Breit-Wigner cross-section formalism[4] with "picket fences" terms to account for distant levels, both bound and unbound. In addition to the resonance parameters, the code allows the effective radius, normalizations, backgrounds, and picket fence terms to be fitted parameters.

An effective radius of 0.949 x 10^{-12} cm was obtained, independent of the details of the bound levels, by fitting the transmission data from 55 to 500 eV. The cross section of the energy region of interest from 0.52 to 55.0 eV is sensitive to the bound levels. Both the same effective radius of 0.948 x 10^{-12} cm and a minimum in χ^2 was obtained for this lowest energy region with a picket fence of bound levels extending from -80 eV to $-\infty$. Figure 1.3.1 shows this fit.

ORNL-DWG 76-1908

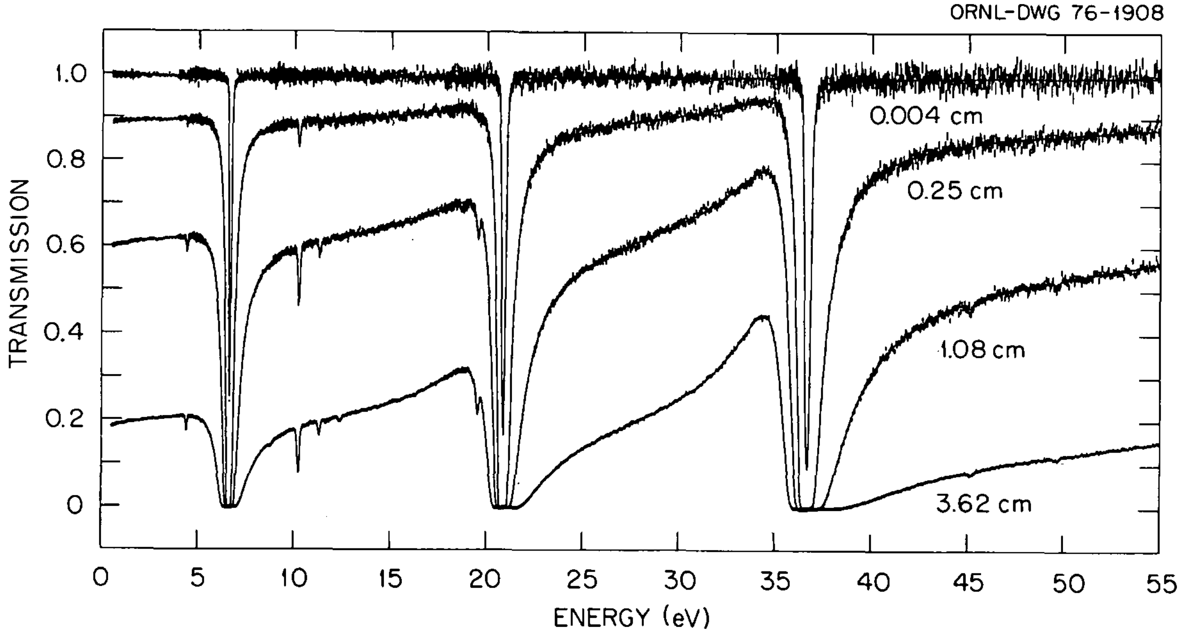

Fig. 1.3.1. Simultaneous least-squares fit with the multilevel computer code, SIOB, to neutron transmission from 0.52 to 55.0 eV through 3.62-, 1.08-, 0.254-, 0.0762-, 0.0254-, 0.0127-, 0.0035-cm samples of ^{238}U. For clarity, the fit to the data from the 0.0762-, 0.0254-, and 0.0127-cm samples is not shown.

Table 1.3.1 compares the neutron and radiation widths from this fit with those contained in ENDF/B-IV. The errors immediately following these widths are the statistical standard deviations from the fit. The numbers in parentheses are deviations corresponding to the uncertainty in the effective Doppler temperature, i.e., 300° ± 5°K The statistical standard deviations for the widths are small. However, the widths depend strongly on the cross-section model and parameters, and the statistical errors contain no estimate of the systematic errors in the data. The uncertainties associated with systematic errors in the data are estimated to be an order of magnitude larger than the statistical errors quoted in Table 1.3.1. It is concluded that radiation widths smaller than those contained in ENDF/B-IV are required to reproduce these transmission data. These smaller capture widths would significantly reduce the discrepancies between calculated and measured capture resonance integrals.[5]

Table 1.3.1. Neutron and radiation widths

	Least-squares fit		ENDF/B-IV	
E (eV)	Γ_n(meV)	Γ_γ(meV)	Γ_n(meV)	Γ_γ(meV)
6.67	1.482 ± .002 (± .006)	22.96 ± .04 (± .02)	1.50	25.6
20.9	10.19 ± .01 (± .03)	22.46 ± .04 (± .02)	8.80	26.8
36.8	33.85 ± .03 (± .05)	22.29 ± .03 (± .02)	31.1	26.0

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

†An IAEA fellow from Comision Nacional Energia Atomica, Argentina.

1. An extended discussion on this problem is contained in the various papers of *Seminar on ^{238}U Resonance Capture,* Brookhaven National Laboratory, BNL-NCS-50451, ed., S. Pearlstein, 1975.

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3. G. de Saussure, D. K. Olsen, and R. B. Perez, to be published.

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1.4 HIGH-RESOLUTION MEASUREMENT OF THE ^{238}U TO ^{235}U FISSION CROSS-SECTION RATIO BETWEEN 2 AND 25 MeV*

F. C. Difilippo† R. B. Perez G. de Saussure D. K. Olsen R. W. Ingle‡

(Reprint of Trans. Am. Nucl. Soc. 24, 449, 1976; also summary of paper presented at Specialists' Meeting on Fast Neutron Cross Sections of ^{233}U, ^{235}U, ^{238}U, and ^{239}Pu, June 28-30, 1976, Argonne National Laboratory, and published in ANL-76-90, p. 114)

There are persistent discrepancies among recent measurements of the ^{238}U fission cross section in the energy region from threshold to about 30 MeV.[1-5] Some of these discrepancies have been attributed to inconsistent energy scales. We describe here a high-resolution measurement performed at ORELA with particular emphasis on determining an accurate energy scale. A fission chamber divided into two sections, one with ^{238}U (2 ppm in other uranium isotopes) and the other with high-purity ^{235}U, was placed at a 40-m flight path. The linac was pulsed at 800 pps with 5-nsec wide neutron bursts.

A careful energy calibration was performed through five carbon resonances which allow an accurate determination of the initial time delay of the time-of-flight scale. The measurement extends over a wide energy scale; in particular, the identification of the ^{238}U subthreshold resonances at 721 eV allows a precise determination of the flight-path length.

After correcting the counting rates in both sections of the fission chamber for dead-time losses, background, and scattering in the chamber walls, the ^{238}U to ^{235}U fission cross-section ratio was obtained at each congruent energy point. This ratio was then normalized to the value of 0.432 at 2.5 MeV.[6]

A comparison with the data of Behrens et al.[2] (LLL), Coates et al.[4] (Harwell), Cierjacks[5] (Karlsruhe), and Meadows[3] (ANL) are presented in Figs. 1.4.1 and 1.4.2.

ORNL-DWG 76-10370

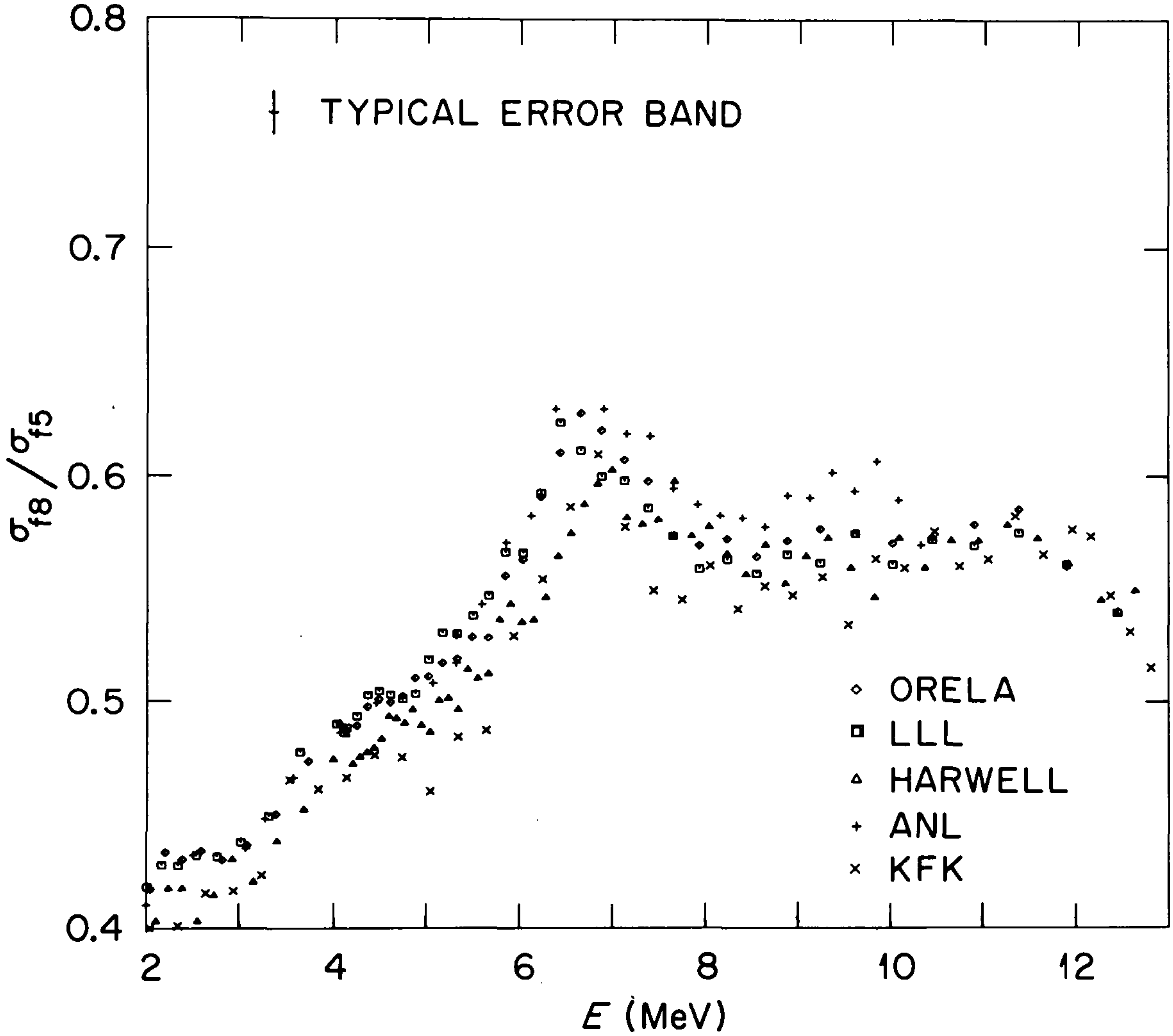

Fig. 1.4.1. Ratio of ^{238}U fission cross section to ^{235}U fission cross section.

ORNL-DWG 76-10369

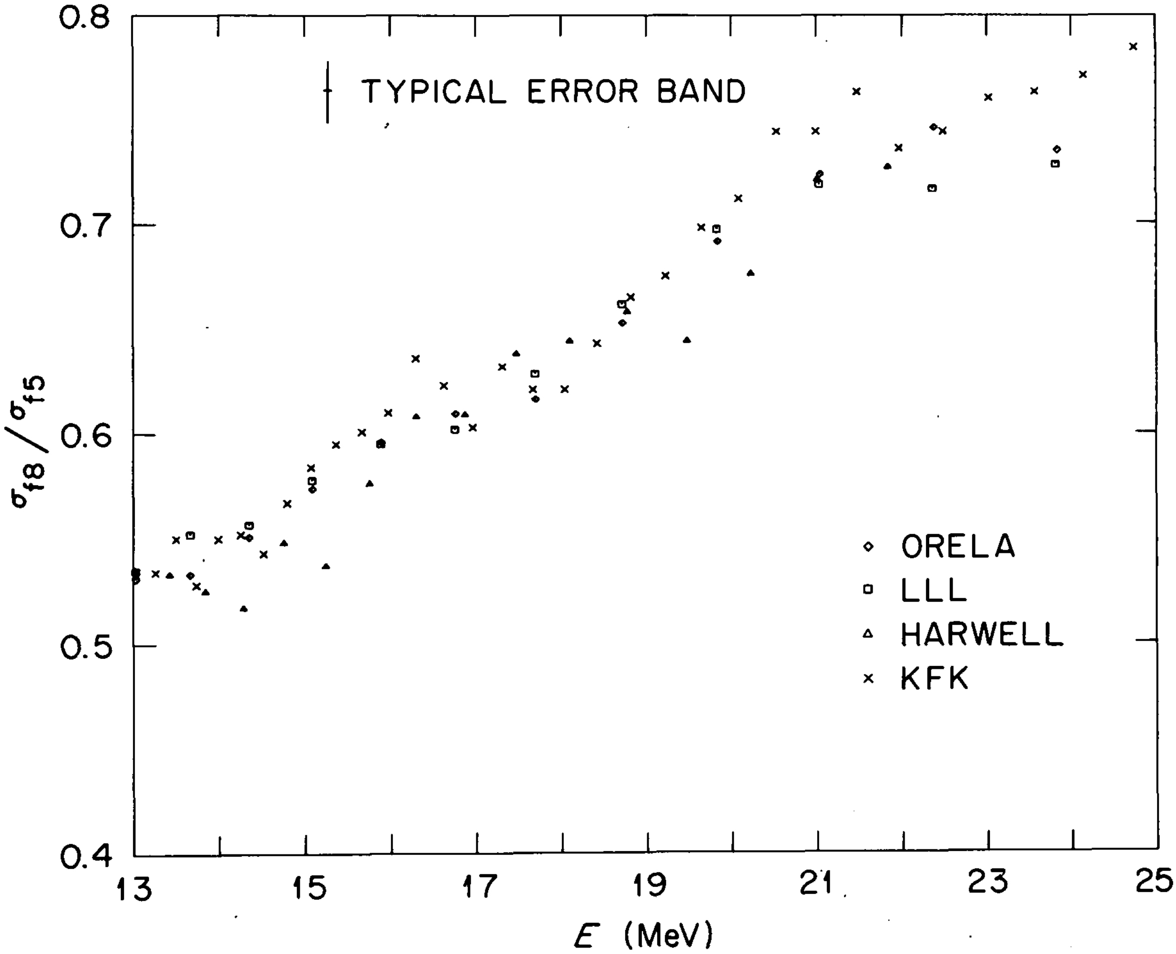

Fig. 1.4.2. Ratio of ^{238}U fission cross section to ^{235}U fission cross section.

Up to 7 MeV our ratios and those of LLL and ANL are consistent, but higher than those of Harwell and Karlsruhe. Above 12 MeV our ratios and those of LLL, Harwell, and Karlsruhe are all consistent. If the various data sets are renormalized to a common value at 2.5 MeV, the agreement below 7 MeV is improved, but the agreement at high energies is destroyed, as must be expected because of the differences in shape between the various data sets.

In conclusion, our ratios below 7 MeV agree well with those of LLL[2] and ANL.[3] Above 7 MeV, our ratios lie between the above two measurements.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

†An IAEA fellow from Comision Nacional Energia Atomica, Argentina.

‡Instrumentation and Controls Division.

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2. J. W. Behrens, G. W. Carlson, and R. W. Bauer, "Neutron Induced Fission Cross Sections of ^{233}U, ^{234}U, ^{236}U, and ^{238}U with Respect to ^{235}U," *Proc. Conf. on Nuclear Cross Sections and Technology,* NBS Special Publication 425 (1975), vol. II, p. 591. Also UCRL-76219 (1975) and private communication.

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4. M. S. Coates, D. B. Gayther, and M. H. Pattenden, *Proc. Conf. on Nuclear Cross Sections and Technology,* NBS Special Publication 425 (1975), vol. II, p. 568.

5. We are indebted to J. W. Behrens for KFK data obtained by private communication from S. Cierjacks (1976).

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1.5 THE ^{238}U SUBTHRESHOLD NEUTRON-INDUCED FISSION CROSS SECTION*

F. C. Difilippo† R. B. Perez G. de Saussure
D. K. Olsen R. W. Ingle‡

(Reprint of Trans. Am. Nucl. Soc. 23, 499, 1976; also summary of paper given at International Conference on the Interactions of Neutrons with Nuclei, July 6-9, 1976, Lowell, Massachusetts, and published in ERDA TIC report CONF-760715-P2, p. 1401)

Subthreshold fission in the ^{238}U nucleus has been measured by Silbert and Bergen,[1] Block et al.,[2] and Blons.[3] We report here recent high-resolution measurements performed at the ORELA facility for neutron energies between 600 eV and 2 MeV. The ORELA was operated at 800 pps with neutron bursts 30 nsec wide and a power of 40 kW on target. The resolution was 2 eV at 600 eV and 500 eV at 100 keV. The detector was a fission chamber divided in two sections. The first section contained 4.5 g of ^{238}U (2 ppm of ^{235}U) and the second section had 0.65 g of highly enriched ^{235}U.

The time-of-flight spectrum between 600 eV and 100 keV is shown in Fig. 1.5.1. The data were reduced to fission cross sections by a ratio measurement to the ^{235}U count rate and the ENDF/B-IV evaluation of the ^{235}U fission cross section.

The average subthreshold fission cross section between 10 and 100 keV was found to be 44 ± 6 μb, which compares well with the values of 50 ± 15 μb and 41 ± 16 μb obtained by Silbert and Bergen[1] and Block et al.,[2] respectively.

Between 600 eV and 57 eV, 28 subthreshold fission clusters were clearly identified. The fission clusters at 721 eV and 1.2 keV were resolved into five and four resonances, respectively.

The present results have been interpreted on the basis of Strutinsky's[4] double-humped fission barrier and the formalism of Weigmann[5] and Lynn.[6] The average level spacing for the Class II levels was D_{II} = 1.8 keV. This yields E_{II} = 1.8 MeV for the height of the second minimum of the fission barrier above the ground state.

The fission areas for the two resolved clusters are in good agreement with the data of Block et al.[2] A value of 1.4 ± 0.3 meV was found for the fission width of the 721-eV resonance. For the unresolved fission chambers the average fission width of the Class II levels was found to be $\langle \Gamma_f^{II} \rangle$ = 0.8 ± 0.2 meV. The distribution of the Class II fission widths was a χ^2-distribution consistent with the presence of two open fission channels. From the present value of $\langle \Gamma_f^{II} \rangle$, the Hill-Wheeler formula and Specht's[7] systematics for the life time of the shape fission isomers, one obtains a value of 6.3 MeV for the height

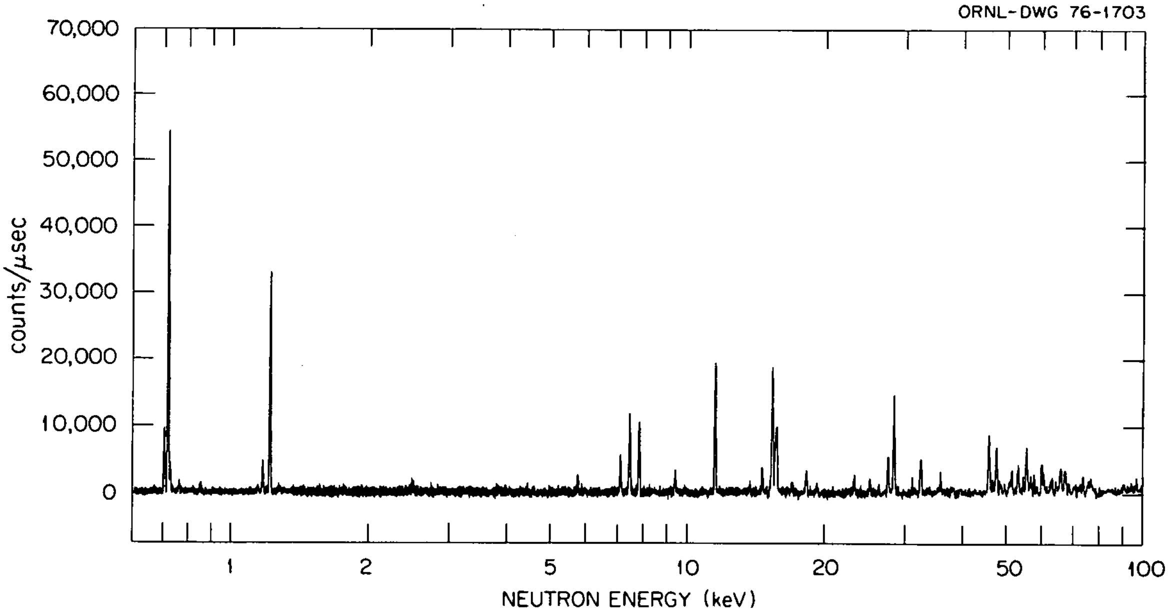

Fig. 1.5.1. Neutron-induced ^{238}U subthreshhold fission.

of the second barrier and a value of 0.7 MeV for the inverse curvature of the second barrier. Both values are in good agreement with the evaluation of Back et al.[8]

From the high-energy data and the neutron binding energy in the ^{238}U nucleus, an upper bound of the height of the fission barrier was estimated at 6.3 MeV. This indicates that the height of the first barrier is either equal to or smaller than 6.3 MeV.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

†An IAEA fellow from Comission Nacional Energia Atomica, Argentina.

‡Instrumentation and Controls Division.

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1.6 NEUTRON CAPTURE CROSS SECTION OF Plutonium-240*

L. W. Weston J. H. Todd†

(Abstract of paper to be published in Nuclear Science and Engineering)

The ^{240}Pu average capture cross section was measured from 200 eV to 350 keV. The cross section was normalized at thermal-neutron energies (0.02 to 0.03 eV) and this normalization was confirmed at the 1.06-eV resonance by the black resonance technique. The Oak Ridge Electron Linear Accelerator (ORELA) was used as the source of pulsed neutrons. The capture gamma-ray detector used was the "total energy detector" which is a modification of the Moxon-Rae detector. The shape of the neutron flux was measured relative to the ^{10}B(n,α) cross section up to 2 keV and the ^{6}Li(n,α) at higher neutron energies. The results of the measurement define the average capture cross section of ^{240}Pu over a wide neutron energy range to an accuracy of about 8%, which is significantly better than previously known. The results indicate that the ENDF/B-IV evaluation is about 25% low above 30-keV neutron energy. The cross section is important in fast Pu-fueled reactors.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

†Instrumentation and Controls Division.

1.7 NEUTRON ABSORPTION CROSS SECTION OF AMERICIUM-241*

L. W. Weston J. H. Todd†

(Abstract of Nucl. Sci. Eng. 61, 356, 1976)

The ^{241}Am neutron absorption cross section, which is predominantly capture, has been measured from 0.01 eV to 370 keV neutron energy. The Oak Ridge Electron Linear Accelerator (ORELA) was used as the source of pulsed neutrons. Resonance parameters (SLBW) have been derived for the data up to 50 eV. The capture gamma-ray detector used was the "total energy detector," which is a modification of the Moxon-Rae detector. This detector required that the events be weighted by their pulse height in the detector and that the net efficiency of the detector be low. The cross section was normalized at thermal neutron energies (0.02 to 0.03 eV), and the shape of the neutron flux was measured relative to the ^{10}B(n,α) cross section up to 2 keV and relative to the ^{6}Li(n,α) cross section at higher neutron energies. The results of the measurement indicate a lower cross section ($\sim$25%) between 0.3 and 100 eV than has been previously indicated and an appreciably higher cross section (by 100% at 100 keV) from 20 to 370 keV.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

†Instrumentation and Controls Division.

1.8 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*

R. Gwin E. G. Silver† R. W. Ingle‡ H. Weaver

(Abstract of Nucl. Sci. Eng. 59, 79, 1976)

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 ^{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 $^{10}B(n,\alpha)$ 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 Oak Ridge Electron Linear Accelerator. 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 samples of the fissile isotopes were 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 contained in ENDF/B-III for energies above 10 keV. The measured values of the ratio α, neutron capture-to-neutron fission, 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%.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

†Institute for Energy Analysis, Oak Ridge, Tennessee.

‡Instrumentation and Controls Division.

1.9 THE ENERGY DEPENDENCE OF THE NEUTRON ABSORPTION AND FISSION CROSS SECTIONS OF URANIUM-235 AND PLUTONIUM-239 BELOW 1 eV AND THE WESCOTT *g* FACTORS*

R. Gwin

(Abstract of Nucl. Sci. Eng. 61, 116, 1976)

The energy dependence of some experimentally derived neutron absorption and fission cross sections of ^{235}U and ^{239}Pu are compared with the ENDF/B-IV results over the energy range from 0.005 to 1.0 eV. Wescott g factors are presented as calculated from the experimental data. The experimental results published previously were normalized to ENDF/B-III values for σ_f and σ_a.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

1.10 REPORT OF THE WORKING GROUP ON ABSOLUTE FISSION MEASUREMENTS*

R. W. Peelle

(Introduction of paper presented at Specialists' Meeting on Fast Neutron Cross Sections of ^{233}U, ^{235}U, ^{238}U, and ^{239}Pu, June 28-30, 1976, Argonne National Laboratory, and published in ANL-76-90, p. 450)

As a result of the importance of the problem area a great wealth of direct fission cross section data has been generated within the last two decades. Yet, even for U-235 fission, some of the recent measurements by experienced experts differ one from the other by several percent. The spread originates in the great difficulty of the measurements. Values presented for the first time at this meeting should contribute to the eventual clarification of which fission cross section values should be taken as most nearly correct.

For the considerations in this report we include as "direct" measurements both the true absolute measurements and the measurements which have been performed relative to various cross section standards such as n-p scattering and, below 100 keV, the Li-6(n, alpha) and B-10(n,alpha) reaction cross sections. The committee chose to limit its attention to the energy region above 20 keV except to the extent that values obtained at lower energies determine the normalization at energies greater than 20 keV.

Several general experimental problems were discussed, and then subcommittees were formed to deal with the U-235(n,f) cross section in the various energy regions and with direct fission measurements on the other nuclides of concern.

The remainder of this committee report is organized according to the topics considered.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

1.11 NEUTRON CAPTURE CROSS SECTION OF ^{59}Co IN THE ENERGY RANGE 2.5 - 1000 keV*

R. R. Spencer R. L. Macklin

(Abstract of paper presented at International Conference on the Interactions of Neutrons with Nuclei, July 6-9, 1976, Lowell, Massachusetts, and published in ERDA TIC report CONF-760715-P2, p. 1290)

The neutron capture cross section of ^{59}Co was measured in the keV energy region using a pair of fluorocarbon liquid scintillator gamma-ray detectors in conjunction with the Oak Ridge Electron Linear Accelerator (ORELA) pulsed neutron source. With time-of-flight resolution of 0.2 to 1.0 nsec/meter, over 160 resonances below 85 keV were observed, for which capture parameters were derived. Total radiation widths were determined for 35 known s-wave resonances yielding $\langle \Gamma_\gamma \rangle_{J=3}$ = 0.66 eV (16 resonances) and $\langle \Gamma_\gamma \rangle_{J=4}$ = 0.46 eV (19 resonances). Correlations (coefficient, $\rho \simeq 0.3$) between the radiative widths and neutron reduced widths were observed at a 90% confidence level for s-wave resonances of either spin state. These correlations along with structure observed in the average capture cross section of cobalt near 500 keV and a similar correlation reported[1,2] for capture in the isotope ^{60}Ni are suggestive of intermediate structure effects in the capture process.

*Research partially sponsored by ERDA Division of Physical Research.

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2. H. Beer and R. R. Spencer, Nucl. Phys. A240, 29 (1975).

1.12 SECONDARY NEUTRON SPECTRA FROM NEUTRON INTERACTIONS IN A THICK CARBON SAMPLE*

G. L. Morgan

(Abstract of ORNL/TM-5814, March, 1977)

The spectra of secondary neutrons produced by neutron interactions in a thick (∿3 mean free path) sample of carbon have been measured as a function of the incident neutron energy over the range 1 to 20 MeV. A linac (ORELA) was used as a white neutron source with a 48-m flight path. Incident energy was determined by time-of-flight, while secondary spectra were obtained through pulse-height unfolding techniques. The results of the measurement are compared to calculations based on the current evaluated data file (ENDF/B 1274).

*Research sponsored by Defense Nuclear Agency.

1.13 CROSS SECTIONS FOR THE Al(*n*,*xn*) and Al(*n*,*x*γ) REACTION BETWEEN 1 AND 20 MeV*

G. L. Morgan F. G. Perey

(Abstract of ORNL/TM-5241, January, 1961, and of Nucl. Sci. Eng. 61, 337, 1976)

Differential cross sections for the production of secondary neutrons and photons from aluminum have been measured at 127° (lab) for incident neutron energies in the range 1 to 20 MeV. An electron linac was used as a neutron source with a white spectrum. Incident neutron energies were determined using time-of-flight techniques for a source-to-sample distance of 48 m. Secondary spectra were determined by unfolding the pulse-height distributions observed in a NE-213 scintillation counter. The results are compared to the current evaluated data file (ENDF/B-IV, MAT 1193).

*Research sponsored by Defense Nuclear Agency.

1.14 THE Cu(*n*,*x*γ) REACTION CROSS SECTION FOR INCIDENT NEUTRON ENERGIES BETWEEN 0.2 AND 20.0 MeV*

G. T. Chapman

(Abstract of ORNL/TM-5215, February, 1976)

Differential cross sections for the neutron-induced gamma-ray production from copper have been measured for incident neutron energies between 0.2 and 20.0 MwV. The Oak Ridge Linear Accelerator (ORELA) was used to provide the neutrons and a NaI spectrometer to detect the gamma rays at 125°. The data presented are the doubly 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 with higher resolution in the neutron energy is also presented. The experimental results are compared with previous measurements made at ORELA and with the Evaluated Neutron Data File (ENDF/B-IV, MAT 1295).

*Research sponsored by Defense Nuclear Agency.

1.15 THE Mo($n,x\gamma$) REACTION CROSS SECTION FOR INCIDENT NEUTRON ENERGIES BETWEEN 0.2 AND 20.0 MeV*

G. L. Morgan E. Newman†

(Abstract of ORNL/TM-5097, ENDF-220, December, 1975)

Differential cross sections for the neutron-induced gamma-ray production from natural molybdenum have been measured for incident neutron energies between 0.2 and 20.0 MeV. The Oak Ridge Linear Accelerator (ORELA) was used to provide the neutrons and a NaI spectrometer 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 Files (ENDF).

*Research sponsored by ERDA Division of Physical Research.

†Physics Division.

1.16 THE Cr($n,x\gamma$) REACTION CROSS SECTION FOR INCIDENT NEUTRON ENERGIES BETWEEN 0.2 AND 20.0 MeV*

G. L. Morgan E. Newman†

(Abstract of ORNL/TM-5098, ENDF-222, January, 1976)

Differential cross sections for the neutron-induced gamma-ray production from natural chromium have been measured for incident neutron energies between 0.2 and 20.0 MeV. The Oak Ridge Linear Accelerator (ORELA) was used to provide the neutrons and a NaI spectrometer to detect the gamma rays at the laboratory angle of 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 Files (ENDF).

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

†Physics Division.

1.17 THE V($n,x\gamma$) REACTION CROSS SECTION FOR INCIDENT NEUTRON ENERGIES BETWEEN 0.2 AND 20.0 MeV*

E. Newman† G. L. Morgan

(Abstract of ORNL/TM-5299, ENDF-221, April, 1976)

Differential cross sections for the neutron-induced gamma-ray production from natural vanadium have been measured for incident neutron energies between 0.2 and 20.0 MeV. The Oak Ridge Linear Accelerator (ORELA) was used to provide the neutrons and a NaI spectrometer 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 Files (ENDF).

*Research sponsored by ERDA Division of Physical Research.

†Physics Division.

1.18 The Mn($n,x\gamma$) REACTION CROSS SECTION FOR INCIDENT NEUTRON ENERGIES BETWEEN 0.2 AND 20.0 MeV*

G. L. Morgan

(Abstract of ORNL/TM-5531, August, 1976)

Differential cross sections for the neutron-induced gamma-ray production from natural manganese have been measured for incident neutron energies between 0.2 and 20.0 MeV. The Oak Ridge Linear Accelerator (ORELA) was used to provide the neutrons and a NaI spectrometer 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 0.22 and 10.6 MeV for coarse intervals in incident neutron energy. The integrated yield of gamma rays of energies greater than 220 keV with higher resolution in the neutron energy is also presented. The experimental results are compared with the Evaluated Neutron Data Files (ENDF).

*Research sponsored by ERDA Division of Physical Research.

1.19 A RE-MEASUREMENT OF THE NEUTRON-INDUCED GAMMA-RAY PRODUCTION CROSS SECTIONS FOR IRON IN THE ENERGY RANGE 850 keV $\leq E_n \leq$ 20.0 MeV*

G. T. Chapman G. L. Morgan F. G. Perey

(Abstract of ORNL/TM-5416, July, 1976)

Values of the gamma-ray production cross sections for neutron interactions with iron as reported by previous investigators have differed by as much as a factor of 1.5 or more at neutron energies greater than about 5 MeV. Because of this discrepancy, the measurements have been repeated at ORNL using the ORELA as a pulsed source of neutrons with energies between 850 keV and 20 MeV. The data were obtained using a NaI(Tl) gamma-ray spectrometer oriented at an angle of 125 degrees to the incident neutron beam. The sample was positioned in the beam at a distance of 47.35 meters from the neutron source. The resulting data, presented as differential cross sections ($d^2\sigma/d\Omega dE$) for gamma rays between 0.7 and 10.5 MeV, show good agreement with some previously published data, but are significantly different from previous ORNL measurements for neutron energies greater than 5 MeV.

*Research sponsored by Defense Nuclear Agency.

1.20 CROSS SECTIONS FOR GAMMA-RAY PRODUCTION BY FAST NEUTRONS FOR 22 ELEMENTS BETWEEN Z = 3 AND Z = 82*

J. K. Dickens G. L. Morgan G. T. Chapman T. A. Love
E. Newman† F. G. Perey

(Abstract of paper to be published in Nuclear Science and Engineering)

Cross sections for the production of gamma rays with 0.3 < E_γ < 10.5 MeV have been measured as a function of neutron energy over the range 0.1 < E_n < 20.0 MeV. Results were obtained for 22 elements which are commonly encountered in the calculation of radiation effects. The measurements were made using a heavily shielded NaI detector in conjunction with the white neutron spectrum from the Oak Ridge Electron Linear Accelerator. Incident neutron energies were determined by time-of-flight over a 47-m flight path while gamma-ray energy distributions were obtained from pulse-height unfolding techniques. Elemental differential cross sections are presented for Li, C, N, F, Mg, Al, Si, Ca, V, Cr, Fe, Ni, Cu, Zn, Nb, Mo, Ag, Sn, Ta, W, Au, and Pb.

*Research sponsered by the Defense Nuclear Agency and by the ERDA Division of Physical Research,

†Physics Division.

1.21 USE OF ORELA FOR SCATTERING MEASUREMENTS IN THE MeV REGION*

F. G. Perey

(Abstract of invited paper presented at American Physical Society Meeting February 2-5, 1976, New York City)

White neutron sources produced by electron linear accelerators have great potential for scattering measurements in the MeV region, where previously monoenergetic neutron sources from Van de Graaff have almost exclusively been used. The very short and intense neutron bursts available at ORELA are quite suitable for a variety of experiments using flight paths of 40 meters or more with well collimated beams. Recently experiments have been completed on elastic and inelastic scattering of neutrons up to a few MeV with incident neutron energy resolutions up to an order of magnitude better than normally done on Van de Graaffs.[1] Examples of data obtained will be shown and the implications of these data for practical applications, as well as the physical information regarding nuclear structure revealed by these data, will be discussed. In another experiment[2] neutron and gamma-ray emission spectra were observed up to 20 MeV incident neutron energy using thin samples in a ring geometry. These data span the region of 8 to 20 MeV which is difficult to cover with monoenergetic sources except at about 14.5 MeV. These experiments provide data of a type not previously available to test the validity of various nuclear models.

*Research sponsored by ERDA Division of Physical Research.

1. See paper 1.22.

2. G. L. Morgan and F. G. Perey, to be published.

1.22 HIGH-RESOLUTION NEUTRON SCATTERING EXPERIMENTS AT ORELA*

W. E. Kinney J. W. McConnell

(Abstract of paper presented at International Conference on the Interactions of Neutrons with Nuclei, July 6-9, 1976, Lowell, Massachusetts)

Sodium, silicon, iron, and carbon neutron elastic differential scattering data taken with resolutions of ∿0.2 nsec/m at the Oak Ridge Electron Linear Accelerator (ORELA) have been reduced to cross sections from 500 to ∿3000 keV in 1-keV intervals using Cσ_T as a standard. The data, acquired at a 40-m flight path at 8 angles from 24° to 155°, were taken in a two-dimensional array of time-of-flight vs pulse height. Inelastic contributions were eliminated by using data from successively higher pulse-height groups. Multiple scattering corrections were obtained from Monte Carlo calculations. After correction, the data were fitted to obtain Legendre expansion coefficients. Pertinent data acquisition and reduction details are given and results shown.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

1.23 HIGH-RESOLUTION FAST-NEUTRON GAMMA-RAY PRODUCTION CROSS SECTIONS FOR IRON UP TO 2100 keV*

W. E. Kinney F. G. Perey

(Abstract of paper submitted for journal publication)

High-resolution gamma-ray production cross sections for the 846-keV gamma ray of iron have been measured up to an incident neutron energy of 2100 keV. The measurements were performed using the Oak Ridge Electron Linear Accelerator as the neutron source and obtained by a ratio measurement to the ^{7}Li 477-keV gamma-ray cross sections. Three NE-213 detectors were used at 30, 90 and 125° to derive the total inelastic cross sections and

the angular distributions. The 1250 angular distributions measured with about 0.1 nsec/m resolution show considerable fluctuations as a function of energy over the resonances seen in the inelastic cross sections. The results are compared to the ENDF/B-IV evaluation, high resolution data at 125° and, after suitable averaging, with recent monenergetic neutron source data which average over the structure experimentally. The general consistency of the data with recent measurements, using different techniques and normalization procedures, indicates that our knowledge of this important cross section for fission reactor applications may now be known to an accuracy better than 10%. This is a significant achievement in view of the wide scatter of earlier data on such a fluctuating cross section.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

1.24 MEASUREMENT OF THE NEUTRON TOTAL CROSS SECTION OF SODIUM*

D. C. Larson J. A. Harvey† N. W. Hill‡

(Reprint of Trans. Am. Nucl. Soc. 23, 496, 1976)

Recent sensitivity analyses[1] for the CRBR upper axial shield indicate that 40% of the integrated tissue dose sensitivity to the sodium total neutron cross section comes from the interference minimum of the 300-keV resonance. With the large quantities of liquid sodium coolant present in the CRBR, the cross-section minimum for this resonance takes on new significance. Recent thick-sample measurements on sodium minima by Brown et al.[2] show a significant discrepancy with the present ENDF/B-IV evaluation[3] for the 300-keV resonance. The evaluation in this region is based on the data of Cierjacks et al.,[4] which show a much sharper minimum than revealed by the measurement of Brown. In addition to the 300-keV resonance problem, no high-resolution total cross-section data were available from ~40 keV to 300 keV, leading to large uncertainty estimates for this energy region in the evaluation.

In order to provide consistent high-resolution data for the ENDF/B-V evaluation, as well as to verify the high-resolution data of Cierjacks, we have measured the transmission of neutrons through a 8.1-cm ($1/n$ = 4.90 b/atom) sample of pure sodium from 40 keV to 20 MeV. The transmitted beam was detected by a NE-110 proton recoil detector located at the 200-m flight path of the Oak Ridge Electron Linear Accelerator (ORELA). A 5-ns electron beam burst width was used, with a repetition rate of 800 sec^{-1}. The data were corrected for dead-time effects (maximum of 9% in the cross section at 1.1 MeV), and background (varying from 0.1-0.2% between 100 keV and 2 MeV, rising to 1% at 8 MeV). The 50,000 channels of transmission data were suitably averaged to improve counting statistics while preserving the resonance structure and were then converted to cross section versus energy.

Figure 1.24.1 shows a comparison of our data (averaged over 10 channels) with the present ENDF/B-IV evaluation from 190 to 310 keV, the region of most serious disagreement. We observe eight resonances in this region, four of which have not been seen in previous transmission measurements. Resonance energies are 201.2, 214, 236.8,* 239.5, 243.1,* 298.1, 299.4* and 305.2* keV, where the asterisk labels the new resonances. In Fig. 1.24.2 the present data near the 300-keV minimum (averaged over 10 channels) are compared with data of Brown et al.[2] and with the ENDF/B-IV evaluation. Total (statistical + systematic) errors (typically ±3% for the ORELA data) are shown. The largest deviation of the present data from the evaluation in the vicinity of the minimum is -12%.

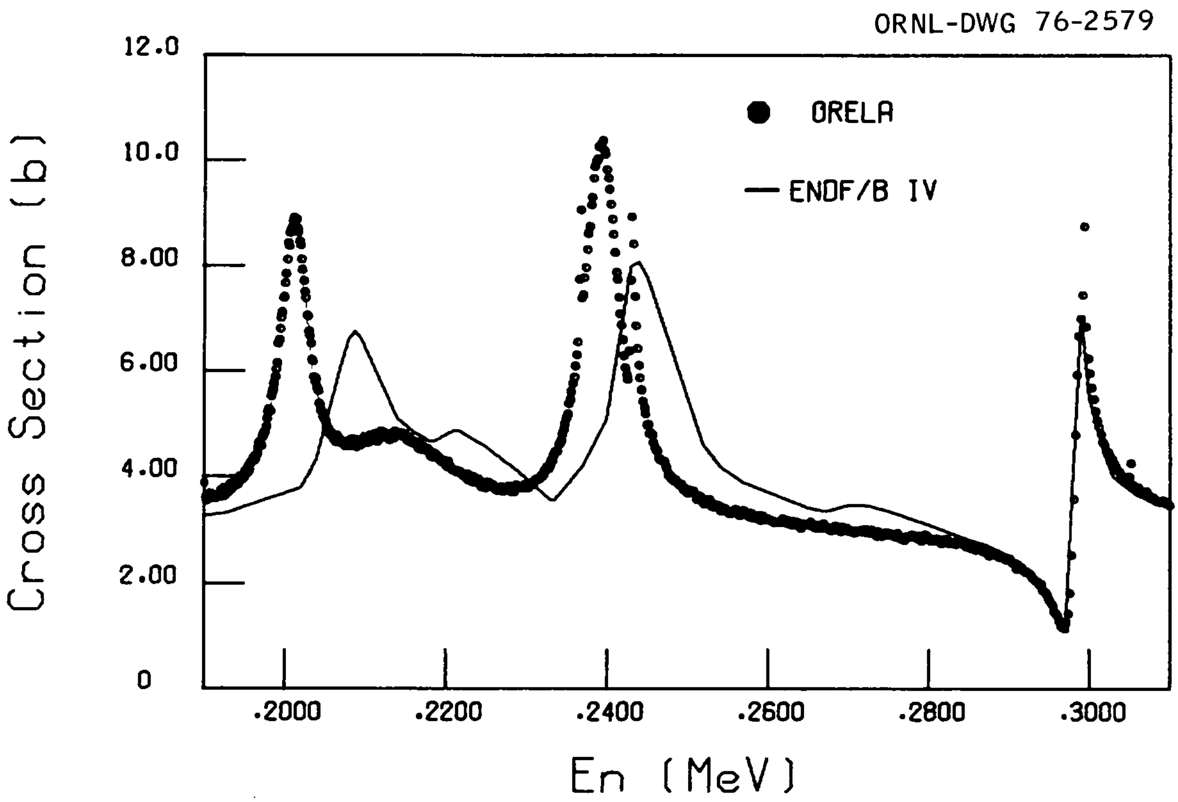

Fig. 1.24.1. The neutron total cross section of sodium from 190 to 310 keV. The data have been averaged over 10 channels. The solid line represents the ENDF/B-IV evaluation.

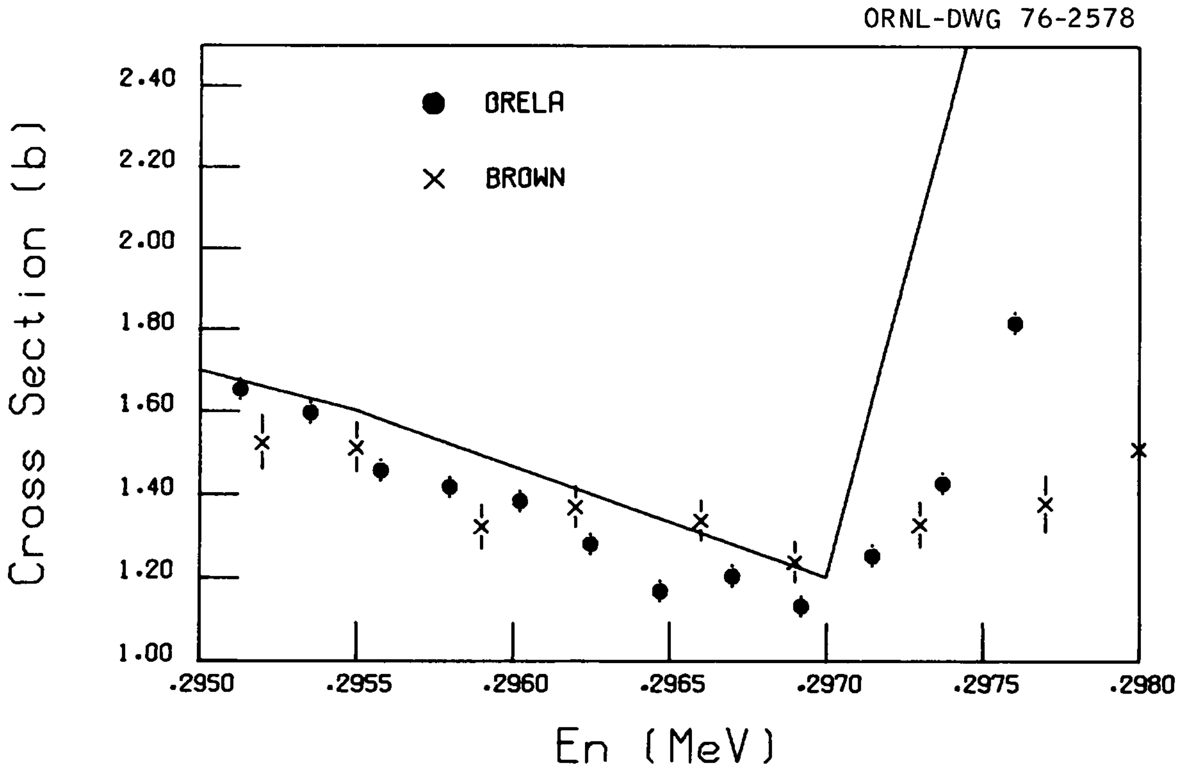

Fig. 1.24.2. The neutron total cross section of sodium near the 300-keV minimum, averaged over 10 channels. The data of Brown et al. represent their measurement through 124 cm of sodium. The solid line represents the ENDF/B-IV evaluation.

In summary, the present measurement points out several areas for improvement in the sodium evaluation for ENDF/B-V, the most important being the broadening of the minimum at 300 keV.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

†Physics Division.

‡Instrumentation and Controls Division.

1. E. M. Oblow, "Survey of Shielding Sensitivity Analysis Development and Applications Program at ORNL," ORNL-TM-5176, (January 1976).

2. P. H. Brown, B. L. Quan, J. J. Weiss and R. C. Block, "Measurement of Neutron Total Cross Sections of Sodium Near Minima," Trans. Am. Nucl. Soc. 21, 505 (1975).

3. N. C. Paik and T. A. Pitterle, "Evaluation of Sodium-23 Neutron Data for the ENDF/B Version III file," Appendix A, WARD-3045T4B-2, Westinghouse Advanced Reactors Division (April 1972), and ENDF/B-IV, MAT 1156, National Neutron Cross Section Center, Brookhaven National Laboratory, Upton, New York (1974).

4. S. Cierjacks, P. Forti, D. Kopsch, L. Kropp, J. Neve and H. Unseld, "High Resolution Total Neutron Cross Sections Between 0.5 to 30 MeV," Karlsruhe report KFK-1000 (June 1968).

1.25 MEASUREMENT OF THE NEUTRON TOTAL CROSS SECTION OF SODIUM FROM 32 keV TO 37 MeV*

D. C. Larson   J. A. Harvey†   N. W. Hill‡

(Abstract of ORNL/TM-5614, October, 1976)

The neutron transmission through a 8.1-cm sample of pure sodium has been measured for neutron energies between 32.5 keV and 37.4 MeV. The Oak Ridge Electron Linear Accelerator (ORELA) was used to provide the neutrons, which were detected at the 200-m flight path by a NE-110 proton recoil detector. The experimental results are tabulated and compared with the total cross section in the ENDF/B-IV file for sodium.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

†Physics Division.

‡Instrumentation and Controls Division.

1.26 MEASUREMENT OF THE NEUTRON TOTAL CROSS SECTION OF FLUORINE FROM 5 eV TO 20 MeV*

D. C. Larson   C. H. Johnson†   J. A. Harvey†   N. W. Hill‡

(Abstract of ORNL/TM-5612, October, 1976)

Neutron transmissions through Teflon (CF_2) and carbon have been measured to provide high-resolution transmission and cross sections for fluorine from 5 eV to 20 MeV. The Oak Ridge Electron Linear Accelerator (ORELA) was used for the neutron source. The 80-m flight path with a ^{6}Li glass detector was used for the low-energy measurements, and the

200-m flight path with a NE-110 detector was used for the higher energy measurements. The various background contributions were carefully studied and are discussed in detail. The 2389 resulting values are tabulated and compared with the current ENDF/B-IV evaluation.

*Research sponsored by ERDA Division of Physical Research and Defense Nuclear Agency.

†Physics Division.

‡Instrumentation and Controls Division.

1.27 MEASUREMENT OF THE NEUTRON TOTAL CROSS SECTION OF SILICON FROM 5 eV TO 730 keV*

D. C. Larson C. H. Johnson† J. A. Harvey† N. W. Hill‡

(Abstract of ORNL/TM-5618, November, 1976)

Neutron transmission through natural silicon samples has been measured for neutron energies between 5 eV and 730 keV. The Oak Ridge Electron Linear Accelerator (ORELA) was used to provide the neutrons. The 80-m flight path with a ^{6}Li glass detector was used for the low energy measurement, and the 200-m flight path with a NE-110 detector was used for the higher energy measurements. The 1,488 resulting values are tabulated and compared with the current ENDF/B-IV evaluation.

*Research sponsored by Defense Nuclear Agency.

†Physics Division.

‡Instrumentation and Controls Division.

1.28 ORELA PERFORMANCE*

T. A. Lewis†

(Abstract of ORNL/TM-5112, April, 1976)

This report presents the most recent information concerning the performance of ORELA that would be of interest to experimenters. Included are characteristics of the beam in terms of both time and intensity and descriptions of systems routinely used to monitor these beam characteristics. For example, with klystron power and maximum electron gun output current at nominal values and for pulse repetition rates in the range above 800 pps, output beam energies per pulse vary from 5 j for 2.5 nsec-wide pulses to ~32 j for 10 nsec pulses and 65 j for 40 nsec pulses.

*Research sponsored by ERDA Division of Physical Research.

†Instrumentation and Controls Division.

1.29 CALCULATIONS PERTAINING TO THE DESIGN OF A PREBUNCHER FOR A 150-MeV ELECTRON ACCELERATOR*

R. G. Alsmiller, Jr. F. S. Alsmiller J. Barish†

(Abstract of ORNL/TM-5419, in press)

Results derived from calculations based on a one-dimensional ballistic model are presented to indicate the extent to which a current pulse of 150-keV electrons containing 1 μC of charge and having a duration of 15 nsec (FWHM) can be bunched in a 2.5-cm-radius conducting cylinder by a combination of voltage gaps followed by a drift space. The bunched current to be useful must be accelerated by an existing accelerator (ORELA), so the calculated results include estimates (upper and lower limits) of the fraction of the bunched beam that will be accelerated. It is found that with 10 voltage caps, each having a linear voltage change from 20 kV to -20 kV over the duration of the pulse, the 15-nsec (FWHM) pulse can be reduced to a pulse of ∿4 nsec (FWHM) in a length ∿375 cm and that ∿50% of this bunched pulse will be accelerated by ORELA.

*Research sponsored by ERDA Division of Physical Research.

†Computer Sciences Division.

1.30 STATUS REPORT TO THE ERDA NUCLEAR DATA COMMITTEE*

Compiled and Edited by
F. G. Perey J. C. Gentry

(Abstract of ORNL/TM-5450, May, 1976; also abstract of ORNL/TM-5834, March, 1977)

These reports were prepared for the ERDA-NDC and cover work performed at ORNL since May 1975 and May 1976, respectively, in areas of nuclear data of relevance to the U.S. applied nuclear energy program. The reports were mostly generated through a review of abstracts of work completed to the point of being subjected to some form of publication in the open literature, formal ORNL reports, ORNL technical memoranda, progress reports, or being presented at technical conferences. As much as possible we have reproduced the complete abstract of the original publication with only minor editing. In a few cases progress reports were written specifically for this publication. The authors have selected the materials to be included in these reports on the basis of perceived interests of ERDA-NDC members and cannot claim completeness.

*Research sponsored by ERDA Division of Physical Research.

1.31 FISSION-PRODUCT BETA AND GAMMA ENERGY RELEASE QUARTERLY PROGRESS REPORT FOR OCTOBER-DECEMBER 1975*

J. K. Dickens T. A. Love J. W. McConnell J. F. Emery† R. W. Peelle

(Abstract of ORNL/TM-5272, February, 1976)

Preliminary experimental information for beta-ray energy release from fission-product decay following thermal-neutron fission of ^{235}U has been obtained for cooling times between 3 and 14,400 sec. The data were obtained as pulse-height spectra for beta energies between 0.25 and 8 MeV using a two-crystal scintillation spectrometer, and were unfolded to give beta-ray energy spectra of moderate resolution. Two irradiation times, t_i = 2.4 and 100 sec, were studied. The energy release data were obtained by integrating the observed spectra and then estimating the contribution for E_β < 0.25 MeV. Difficulties encountered in this first experiment using the beta-ray detection equipment are discussed.

Previously reported preliminary gamma-ray spectra have been compared with spectra calculated using spectral information in the ENDF/B-IV data file for ∿180 fission products. Several comparisons are presented for t_i = 100 sec. As expected the comparison is not very good for short cooling times, but is encouraging for t_{cool} ∿ 2000 sec.

*Research sponsored by Nuclear Regulatory Commission.

†Analytical Chemistry Division.

1.32 FISSION-PRODUCT BETA AND GAMMA ENERGY RELEASE QUARTERLY PROGRESS REPORT FOR JANUARY-MARCH 1976*

J. K. Dickens T. A. Love J. W. McConnell J. F. Emery† R. W. Peelle

(Abstract of ORNL/NUREG/TM-23, May, 1976)

Several important aspects of the overall program to determine fission-product energy release rates using spectroscopic methods are presented. These aspects include: (a) fabrication of a sample holder for the beta-ray measurements having an absorbing layer between sample and detector of $\sim$5 mg/cm^2; (b) a measurement of the maximum probable loss of the fission-gas product ^{133}I; and (c) a new determination of the absolute intensities of the principal x and gamma rays in the decay of ^{99}Mo in equilibrium with ^{99}Tc*.

*Research sponsored by Nuclear Regulatory Commission.

†Analytical Chemistry Division.

1.33 FISSION-PRODUCT BETA AND GAMMA ENERGY RELEASE QUARTERLY PROGRESS REPORT FOR APRIL-JUNE 1976*

J. K. Dickens T. A. Love J. W. McConnell R. M. Freestone
J. F. Emery† R. W. Peelle

(Abstract of ORNL/NUREG/TM-47, September, 1976)

Several important aspects of the overall program to determine fission-product energy release rates using spectroscopic methods are presented. These aspects include: (a) study of possible loss of fission gases and the impact upon the measured energy release rates; (b) improvement in data transfer from the PDP-15 data-acquisition computer to the PDP-10 data-reduction computer; and (c) efficiency calibration of our 90-cm^3 Ge(Li) detector. In addition, the final data-taking run for beta energy release was completed.

*Research sponsored by Nuclear Regulatory Commission.

†Analytical Chemistry Division.

1.34 FISSION-PRODUCT BETA AND GAMMA ENERGY RELEASE QUARTERLY PROGRESS REPORT FOR JULY-SEPTEMBER 1976*

J. K. Dickens T. A. Love J. W. McConnell R. M. Freestone
J. F. Emery† R. W. Peelle

(Abstract of ORNL/NUREG/TM-65, December, 1976)

Gamma-ray energy-release data for thermal-neutron fissioning of the fuel element ^{235}U were obtained for waiting times $2 \leqslant t_w \leqslant 14000$ sec. These data were processed to give modest-resolution gamma-ray energy spectra vs t_w. The resulting spectra have been integrated to give integral gamma-ray energy-release data. Beta-ray data obtained during the previous quarter were processed to give beta-ray energy spectra for waiting times $2 \leqslant t_w \leqslant 14000$ sec. Some of these spectra have been compared with existing data; the quality of agreement varies between poor and very good. The beta-ray spectra have been integrated to give integral beta-ray energy-release data.

Total energy-release data were obtained by summing the gamma-ray energy-release data with the beta-ray energy release data. The total energy-release data have been compared (a) with the current ANS Decay Heat Standard, (b) with results of computations using current best files of fission-product data, and (c) with results of other recent experiments. These comparisons suggest likely improvement to the current standard for $t_w \leqslant 400$ sec, as well as reduction of the uncertainties assigned to the standard.

*Research sponsored by Nuclear Regulatory Commission.

†Analytical Chemistry Division.

1.35 DECAY HEAT OF ^{235}U FISSION PRODUCTS BY BETA- AND GAMMA-RAY SPECTROMETRY*

J. K. Dickens T. A. Love J. W. McConnell R. W. Peelle

(Abstract of paper presented at Fourth Reactor Safety Information Meeting, September 27-30, 1976, Gaithersburg, Maryland)

Heat release from the decay of fission products is an important source of after-shutdown power for which accurate estimates are required. Benchmark experiments we report on are required to establish the credibility of summation calculations based on present or future fission-product files and perhaps to allow refinements of the files in time regions for which the present files contain approximations.

The fast-rabbit facilities of the Oak Ridge Research Reactor were employed to irradiate 1- to 10-μg samples of ^{235}U for nominal periods of 1, 10, and 100 sec. Energy release was measured as early as 2 sec following the end of the irradiation. The released power was observed using the techniques of nuclear spectroscopy to permit separate observations of emitted beta-ray and gamma-ray spectra in successive time intervals following the end of irradiations. To obtain the total decay heat, the resulting spectra were integrated over energy and the beta- and gamma-ray results were summed together. The differential gamma-ray spectra are valuable (a) for sensing unexpected backgrounds, (b) for providing the ability to compute the transport of gamma-ray energy from a fuel element, and (c) for suggesting specific cases for which the fission-product decay file should be improved.

*Research sponsored by Nuclear Regulatory Commission.

1.36 FISSION-PRODUCT BETA AND GAMMA DECAY-ENERGY RELEASE RATES FOR THE TIME PERIOD 2 sec TO 4 hr FOLLOWING THERMAL-NEUTRON FISSION OF ^{235}U*

R. W. Peelle T. A. Love J. W. McConnell J. K. Dickens

(Summary of paper to be presented at American Nuclear Society Topical Meeting on the Safety Technology of Light-Water, Heavy-Water and Gas-Cooled Nuclear Reactors, July 30 - August 4, 1977, Sun Valley, Idaho)

Both the reactor community and the public are concerned about a hypothetical reactor accident in which there is postulated a catastrophic failure in the flow of light-water coolant to an operating commercial power reactor. Of critical concern is the energy being fed into the shut-down core by the fission-product gamma- and beta-ray emitters trapped in the fuel for several thousand seconds following shutdown. Present licensing procedures utilize the current ANS 5 standard[1] multiplied by 1.2 (referred to as "ANS + 20%") to determine these energy-release rates in safety analyses of light-water reactors. Recent experimental results[2-4] indicate that ANS + 20% is quite conservative; in addition, there is concern over the lack of precision associated with the present standard.

Our experiment was designed to obtain spectral distributions of gamma and beta radiation (separately) as a function of time following short (1 to 100 sec) irradiation periods of ^{235}U by thermal neutrons. These data emphasize contributions from short-lived fission products which have not been well-studied by radiochemical techniques.

All spectra have been integrated to obtain the total energy release as a function of irradiation time (t_i), time following fission to the initiation of the measurement (t_w), and the counting time (t_c). This measured total energy-release function is designated as $E(t_i, t_w, t_c)$. For the cases where $t_w \gg t_i$ and $t_w \gg t_c$, $E(t_i, t_w, t_c)/t_c$ represents the energy-release rate at $t = t_w + 0.5\ (t_i + t_c)$ following an essentially instantaneous burst of fissions. In this format our data have been compared to similar data obtained from summation calculations[5] of fission-product decay energy release for gamma rays and beta rays (separately). Our gamma-ray data are larger than calculated data for $t < 20$ sec, agree well with calculations for $25 < t < 900$ sec and $t > 9000$ sec, and are smaller than calculation for $900 < t < 9000$ sec; the largest difference, ~20%, occurs at $t = 2.7$ and t ~2500 sec. Our beta-ray data are larger than calculated data for $t < 10$ sec, smaller by ~15% for $20 < t < 500$ sec, and in good agreement for $t > 800$ sec.

The measured data have been summed to represent total decay power as a function of time t_w following shut-down of a $(4 - t_w)$ hour irradiation. By adding the ANS-standard fission-product decay power for time-after-shutdown of 14400 sec to our data results in a direct comparison of our data to the present ANS standard. This comparison is shown in Fig. 1.36.1. Note that for $t \leq 1000$ sec the measured portion of the ordinate dominates.

ORNL-DWG 76-17608

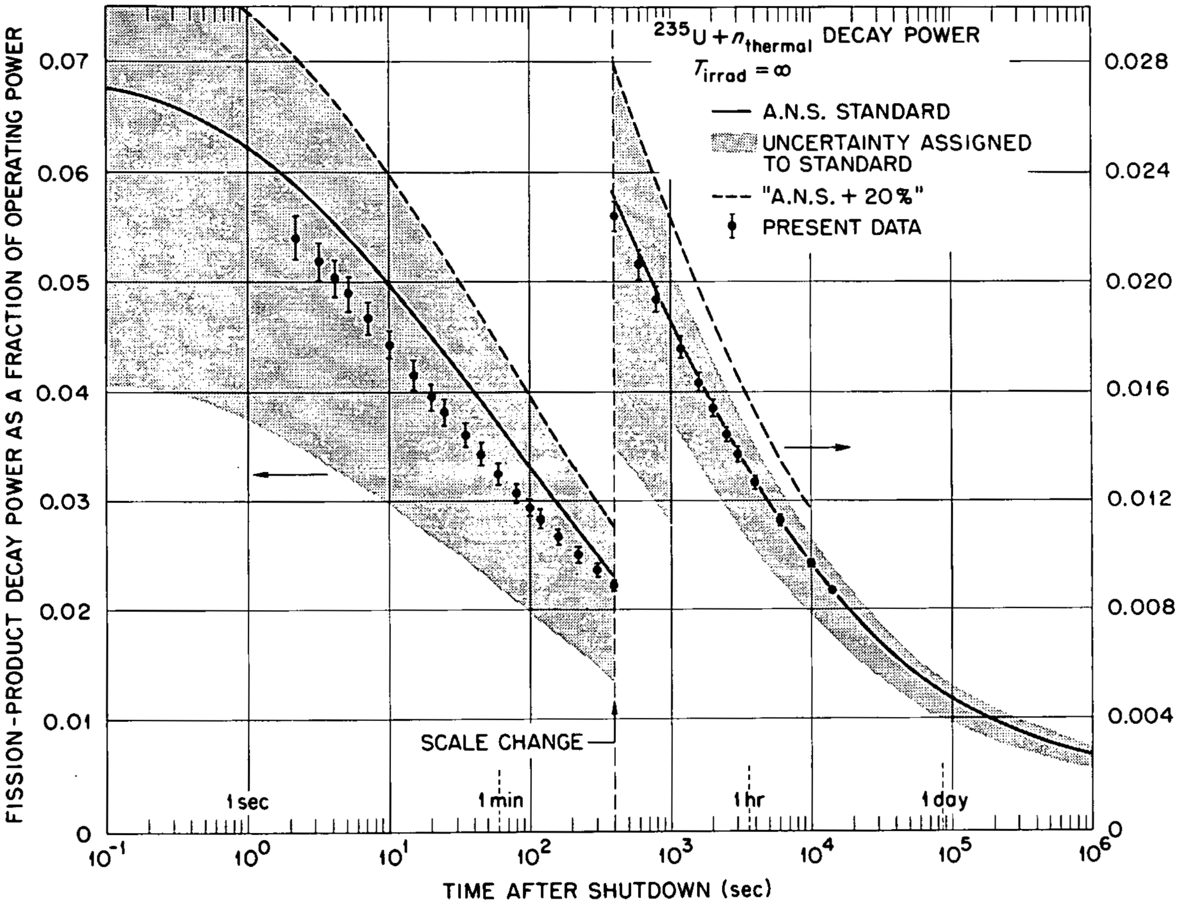

Fig. 1.36.1. Comparison of present experimental results with the current ANS standard for fission-product decay heat for thermal-neutron fission of ^{235}U.

Results of two other recent experiments[3,4] are very similar to ours when properly compared to the ANS standard. We conclude that the presently used "ANS + 20%" is an extremely conservative representation of the fission-product decay power for ^{235}U thermal-neutron fission.

*Research sponsored by Nuclear Regulatory Commission.

1. "Decay Energy Release Rates Following Shutdown of Uranium-Fueled Thermal Reactors," Proposed ANS Standard, Approved by Subcommittee ANS-5, October 1971, Revised October 1973.

2. J. K. Dickens, T. A. Love, J. W. McConnell, R. M. Freestone, J. F. Emery, and R. W. Peelle, "Fission Product Beta and Gamma Energy Release Quarterly Progress Report for July-September 1976," ORNL/NUREG/TM-65 (December 1976).

3. S. J. Friesenhahn, N. A. Lurie, V. C. Rogers, and N. Vagelatos, "U-235 Fission Product Decay Heat from 1 to 10^5 Seconds," EPRI NP-180, Project 392-1 Final Report, Prepared for EPRI by IRT Corporation (February 1976).

4. J. L. Yarnell and P. J. Bendt, "Decay Heat by Calorimetry," Los Alamos Scientific Laboratory Report No. LA-UR 76-2036 (undated); report submitted to the Fourth Water Reactor Safety Research Information Meeting held September 1976.

5. These calculations were carried out using the ORIGEN code for an irradiation period of 10^{-4} sec. A description of the ORIGEN code is given in M. J. Bell, "ORIGEN - The ORNL Isotope Generation and Depletion Code," ORNL-4628 (May 1973). R. Schenter and T. R. England have performed similar calculations obtaining essentially identical results with the ORIGEN calculation (private communication, 1976).

1.37 A SIMPLE METHOD FOR DETERMINING ABSOLUTE DISINTEGRATION RATES FOR SOME RADIONUCLIDES*

J. K. Dickens

(Abstract of paper submitted for journal publication)

A method is described for determining absolute disintegration rates for certain electron-capture isotopes using x-ray-gamma-ray summing in a single high-resolution detector. The method does not require knowledge of detector efficiencies nor of gamma-ray branching ratios. Results obtained for ^{65}Zn are presented in which a source of 10500 disintegrations/sec was calibrated to an estimated accuracy of 0.9% using a 200 mm^2 intrinsic Ge detector.

*Research sponsored by Nuclear Regulatory Commission.

1.38 THE DECAY OF ^{99}Mo*

J. K. Dickens   T. A. Love

(Abstract of ORNL/NUREG-11, January, 1977)

Relative intensities for K x-rays and gamma rays emanating from ^{99}Mo in equilibrium with its $^{99}Tc^*$ daughter have been measured using several Ge photon detectors. Combining these intensities with an evaluated set of electron-conversion coefficients has provided a set of absolute intensities for the observed gamma rays. The absolute intensity for the dominant 140.5-keV gamma ray in ^{99}Tc was determined to be $90.7 \pm 0.6/100$ ^{99}Mo disintegrations for ^{99}Mo decay in equilibrium with decay of the $^{99}Tc^*$ daughter.

*Research sponsored by Nuclear Regulatory Commission.

1.39 NEUTRON-INDUCED GAMMA-RAY PRODUCTION IN LEAD-208 FOR INCIDENT-NEUTRON ENERGIES BETWEEN 4.9 AND 8.0 MeV, AND IN BISMUTH-209 FOR INCIDENT-NEUTRON ENERGY OF 5.4 MeV*

J. K. Dickens

(Abstract of paper to be published in Nuclear Science and Engineering)

Interactions of neutrons with ^{208}Pb have been studied by measuring photon production cross sections. Gamma-ray spectra were obtained at incident-neutron energies of 4.9, 5.4, 6.4, 6.95, 7.45 and 8.0 MeV with a γ-ray detector system utilizing a 48-cc Ge(Li) detector. Nearly monoenergetic neutrons were obtained from the D(d,n) reaction using deuterons obtained from the (pulsed) Oak Ridge National Laboratory 5-MV Van de Graaff accelerator. Time of flight was used to discriminate against pulses due to neutrons and background radiation. Extracted differential and total inelastic cross sections have been compared with previous comparable measurements and with data from the ENDF/B evaluation with generally satisfactory results. New information on the level structure of ^{208}Pb is reported.

Data were also obtained at E_n = 5.4 MeV for a sample of natural bismuth, and new information on the level structure of ^{209}Bi was obtained. Differential γ-ray production cross sections were obtained, and the total inelastic cross section at E_n = 5.4 MeV was deduced from these data.

*Research sponsored by Defense Nuclear Agency.

2. CROSS-SECTION EVALUATIONS AND THEORY

2.1 AN EVALUATION OF NEUTRON AND GAMMA-RAY-PRODUCTION CROSS-SECTION DATA FOR LEAD*

C. Y. Fu F. G. Perey

(Abstract of Atom. Data & Nucl. Data Tables 16, 409, 1975)

A survey was made of the available information on neutron and gamma-ray-production cross-section measurements of lead. From these and from relevant nuclear-structure information on the Pb isotopes, we prepared recommended neutron cross-section data sets for lead covering the neutron energy range from 0.00001 eV to 20.0 MeV. The cross sections are derived from experimental results available to February 1972 and from calculations based on optical-model, DWBA and Hauser-Feshbach theories. Comparisons which show good agreement between theoretical and experimental values are displayed in a number of graphs. Also presented graphically are smoothed total cross sections, Legendre coefficients for angular distributions, and a representative energy distribution of gamma rays from resonance capture.

*Research funded by Defense Nuclear Agency.

2.2 CALCULATED NEUTRON CROSS SECTIONS FOR Cu AND Nb UP TO 32 MeV FOR NEUTRON DAMAGE ANALYSIS*

C. Y. Fu F. G. Perey

(Abstract of paper submitted for journal publication)

Cross sections for neutron interaction with Cu and Nb, with emphasis on spectra of light particles from binary reactions, are calculated for neutron energies from 4 to 32 MeV for estimating recoil probability densities for the analysis of damage experiments with a Be(d,n) neutron source. Nuclear model parameters were adjusted to reproduce the available cross-section data around 14 MeV. Helium production cross sections were also calculated for ^{63}Cu for neutrons below 20 MeV, as an illustration of the Hauser-Feshbach method for calculating tertiary reaction cross sections.

*Research sponsored by ERDA Division of Physical Research.

2.3 CONSISTENT CALCULATIONS OF (n,x) AND (n,$x\gamma$) CROSS SECTIONS FOR Ca-40, E_n = 1 - 20 MeV

C. Y. Fu

(Abstract of Atom. Data & Nucl. Data Tables 17, 127, 1976)

Cross sections of neutron interaction with ^{40}Ca and the subsequent production of gamma rays are calculated and compared with experiments. Various nuclear models are judiciously applied for the calculation. The Hauser-Feshbach theory for binary reactions is extended to include tertiary reactions, which are important for ^{40}Ca from 10 to 20 MeV. Continuum-level spins and parities are included in the gamma-ray-production calculation to conserve angular momentum. An extensive measurement of gamma-ray-production cross sections, available after all model parameters were fixed, is used to test the predictability of the models, particularly in the high-energy range, where tertiary reactions contribute significantly.

*Research sponsored by Defense Nuclear Agency.

2.4 COMPILATION OF PHENOMENOLOGICAL OPTICAL-MODEL PARAMETERS, 1954-1975*

C. M. Perey F. G. Perey

(Abstract of Atom. Data & Nucl. Data Tables 17, 1, 1976)

Presented here is a compilation, with bibliography, of optical-model parameters determined by fitting elastic-scattering angular distributions for various incident particles including heavy ions. It includes parameters from previous compilations back to 1954 and from an extensive literature search in the leading journals and publications in nuclear physics up to June 1975 inclusively.

*Research sponsored by ERDA Division of Physical Research.

2.5 THE ENDF/B-IV REPRESENTATION OF THE URANIUM-238 TOTAL NEUTRON CROSS SECTION IN THE RESOLVED RESONANCE ENERGY REGION*

G. de Saussure D. K. Olsen R. B. Perez

(Abstract of Nucl. Sci. Eng. 61, 496, 1976)

The ENDF/B-IV prescription fails to represent correctly the ^{238}U total (and scattering) cross section between the levels of the resolved range. We show how this representation can be improved by properly accounting for the contribution of levels outside the resolved region to the cross section at energies inside the resolved region, and by substituting the more precise multilevel Breit-Wigner formula for the presently used single-level formula. We illustrate the importance of computing accurately the minima in the total cross section by comparing values of the self-shielded capture resonance integral computed with ENDF/B-IV and with a more accurate cross-section model.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

2.6 AN EVALUATION FOR ENDF/B-IV OF THE NEUTRON CROSS SECTIONS FOR ^{235}U FROM 82 eV TO 25 keV*

R. W. Peelle

(Abstract of ORNL-4955, ENDF-233, May, 1976)

Capture and fission cross sections for ^{235}U in the "unresolved resonance" energy region were evaluated to permit determination of local-average resonance parameters for the ENDF/B-IV cross-section file. Microscopic data were examined for infinitely dilute average fission and capture cross sections and also for intermediate structure unlikely to be reproduced by statistical fluctuations of resonance widths and spacings within known laws. Evaluated cross sections, averaged over lethargy intervals greater than 0.1, were obtained as an average over selected data sets after appropriate renormalization. Estimated uncertainties are given for these evaluated average cross sections. The "intermediate" structure fluctuations common to a few independent data sets were approximated by straight lines joining successive cross sections at 120 selected energy points; the cross sections at the vertices were adjusted to reproduce the evaluated average cross sections over the broad energy regions. Data sources and methods are reviewed, output values are tabulated, and some modified procedures are suggested for future evaluations.

Evaluated fission and capture integrals for the resolved resonance region are also tabulated. These are not in agreement with integrals based on the resonance parameters of ENDF/B versions III and IV.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

2.7 REQUIREMENTS ON EXPERIMENT REPORTING TO MEET EVALUATION NEEDS*

R. W. Peelle

(Abstract of paper presented at Specialists' Meeting on Fast Neutron Cross Sections of ^{233}U, ^{235}U, ^{238}U, and ^{239}Pu, June 28-30, 1976, Argonne National Laboratory, and published in ANL-76-90, p. 421)

To define the requirements placed by the evaluation of nuclear cross sections upon the reporting of experimental results, a model of part of the evaluation process is presented. The model is a straightforward application of nondiagonal weighted least-squares estimation to average cross sections in the energy regions where the shape of the cross section is not given by theory. To combine in a logical way the existing evaluated information with one or more new sets of experimental results, the estimated covariance matrix of each experimenter's results needs to be known on an appropriate mesh. The liklihood that each experimenter may underestimate the uncertainties in his results does not remove the need for him to record for users the estimated magnitudes and correlation patterns of these uncertainties.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

2.8 SUR, A PROGRAM TO GENERATE ERROR COVARIANCE FILES*

F. C. Difilippo†

(Abstract of ORNL/TM-5223, March, 1976)

Covariance matrices were calculated for the ^{238}U, ^{241}Pu, and ^{239}Pu fission cross sections and for the ^{238}U, ^{240}Pu, ^{241}Pu, and ^{239}Pu capture cross sections. A computer program was written which uses the evaluated ENDF/B data files and the measured or evaluated (from other evaluations) cross sections for the calculation of the uncertainty files. An effort has been made to make the output of the program consistent with the ENDF/B error files format. A user's manual for the present code and references utilized in the covariance matrix calculations are given.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

†On assignment from Comisión Nacional de Energia Atómica, Argentina.

2.9 THE ANALYTICAL CONTINUATION OF THE PROPAGATOR*

R. B. Perez

(Abstract of paper presented at the International Conference on the Interactions of Neutrons with Nuclei, July 6-9, 1976, Lowell, Massachusetts, and published in ERDA TIC report CONF-760715-P2, p. 1458)

In neutron interaction studies, the Neumann series expansion of the Lippmann-Schwinger equation for the Green's function (propagator) of a system leads to the classical expansion for the resolvent kernel, $R(\vec{x}|\vec{x}')$

$$R(\vec{x}|\vec{x}') = \delta(\vec{x} - \vec{x}') + f(\vec{x}|\vec{x}') + \int d\vec{x}'' \, f(\vec{x}|\vec{x}'') \, f(\vec{x}''\vec{x}') + \dots \tag{1}$$

with

$$f(\vec{x}|\vec{x}') = \varepsilon H_i(\vec{x})\; G_o(\vec{x}|\vec{x}') \;, \tag{2}$$

where ε is the coupling constant, $H_i(\vec{x})$ the interaction operator and $G_o(\vec{x}|\vec{x}')$ the free propagator. The functional series (1) can be recast in terms of Volterra's powers by composition:[1]

$$f^{\star(n)}\;(\vec{x}|\vec{x}') = \int d\vec{x}^{(n)}\; \ldots\; f[\vec{x}|\vec{x}^{(n)}]\; \ldots\; f(\vec{x}''|\vec{x}') \tag{3}$$

in the form

$$R = i^{\star} + f^{\star(1)} + f^{\star(2)} \;\ldots\; f^{\star(n)} \tag{4}$$

with

$$i^{\star}(\vec{x}|\vec{x}') = \delta(\vec{x} - \vec{x}')\;. \tag{5}$$

The series (4) is a functional hypergeometric series, convergent whenever $||f^{\star}|| < 1$. The analytical continuation of the hypergeometric series (4) leads to

$$R = -[f^{\star(-1)} + f^{\star(-2)} + \;.\;.\;.\; f^{\star(-n)}]\;,$$

where $f^{\star(-n)}$ is a negative power by composition, satisfying the relation $f^{\star(n)}\;.\;f^{\star(-n)} = i^{\star}$. We define the strong coupling operator, Q, by the relation

$$Qf^{\star} = i^{\star}\;, \tag{6}$$

so that the resolvent kernel is given from Eqs. (4) and (6) by

$$R = -\,Q(I - Q)^{-1}\;. \tag{7}$$

The definition (6) of the strong coupling operator is equivalent to the integral equation

$$\int d\vec{x}''\; Q(x|\vec{x}'')\; H_i(\vec{x}'')\; G_o(\vec{x}''|x') = \varepsilon^{-1}\;\delta(\vec{x} - \vec{x}')\;. \tag{8}$$

From (8) and (7) the propagator is given as a power series of the inverse of the coupling constant. The generalized resolvent equation, Mockel,[2]

$$\frac{\partial}{\partial\varepsilon} R = -\,R\,H_1 R \tag{9}$$

is satisfied both by the series expansion of the propagator (1), valid for small values of the coupling constant, and by the strong coupling expansion (7). Hence, the latter is the analytical continuation of the usual Neumann series for the propagator.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

1. V. Volterra, *Theory of Functionals*, p. 99, Dover Publications Inc., 1959, New York.

2. A. Mockel, J. Math. Phys. 8, 2318 (1967).

2.10 A NEW PERTURBATION FORMALISM FOR THE COMPLEX WIDTHS AND POLES OF THE TRANSITION T-MATRIX*

R. B. Perez G. de Saussure

(Abstract of paper presented at the International Conference on the Interactions of Neutrons with Nuclei, July 6-9, 1976, Lowell, Massachusetts, and published in ERDA TIC report CONF-760715-P2, p. 1457)

The T-matrix of nuclear reaction theory can be written in the form[1]

$$T_{cc'} = i \sum_{v} \frac{g_{vc} g_{vc'}}{\varepsilon_v - E} \,, \tag{1}$$

where g_{vc} is the complex width for channel c and ε_v the complex poles of the transition T-matrix. We have shown that, in terms of a parameter τ ($0 < \tau \leqslant 1$), the complex widths and poles of the T-matrix are obtained via the solution of the two coupled Volterra equations

$$g_{vc}(\tau) = g_{vc}(0) + \sum_{v' \neq v} \int_0^{\tau} d\tau' \frac{P_{vv'}(\tau') g_{v'c}(\tau')}{\left\{W_{0v'v} - \int_0^{\tau'} d\tau'' [P_{v''v''}(\tau'') - P_{vv}(\tau'')]\right.} \,. \tag{2}$$

and

$$\varepsilon_v(\tau) = \varepsilon_v(0) - \int_0^{\tau} d\tau' \, P_{vv}(\tau') \,, \tag{3}$$

where $g_{vc}(0)$ is the real partial width of R-matrix theory[2] and $\varepsilon_v(0)$ is given in terms of R-matrix parameters by

$$\varepsilon_v(0) = E_v - \frac{i}{2} \Gamma_v \,. \tag{4}$$

The matrix elements, $P_{vv'}$, are functions of the complex widths, g_{vc}, and the interaction between the initial R-matrix states. In the present formalism the interaction contains both changes in the hamiltonian operator and the boundary conditions. The convergence of the method depends on the ratio of the elements of the interaction matrix to the spacing of the complex poles rather than on the same ratio expressed in terms of the spacing of the R-matrix poles on the real axis, as is the case in the usual perturbation approach.

THe present method has been applied to neutron cross section calculations in cases of large level interference and to the study of intermediate structure phenomena.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

1. R. B. Perez and G. de Saussure, Phys. Rev. C10, 187 (1974).

2. E. P. Wigner and L. Eisenbud, Phys. Rev. 72, 29 (1944).

2.11 CALCULATED NUCLEON SPECTRA AT SEVERAL ANGLES FROM 192-, 500-, 700-, AND 900-MeV CARBON-12 ON IRON-56*

H. W. Bertini   R. T. Santoro   O. W. Hermann†

(Abstract of ORNL/TM-5161, February, 1976, and of Phys. Rev. C14, 590, 1976)

Neutron spectra were calculated as a function of angle between 0 and 110° for ^{12}C on ^{56}Fe at 192, 500, 700, and 900 MeV. Proton spectra were calculated for the same angular range but for only 192-MeV ^{12}C on ^{56}Fe. The most significant property of these spectra is that there is an appreciable number of neutrons emitted with energies greater than the incident energy per nucleon at all angles investigated.

*Research sponsored by ERDA Division of Physical Research.

†Computer Sciences Division.

2.12 COMPARISON OF MEASURED NEUTRON SPECTRA WITH PREDICTIONS OF AN INTRANUCLEAR-CASCADE MODEL*

Aaron Galonsky†   R. R. Doering‡   D. M. Patterson‡
H. W. Bertini

(Abstract of Phys. Rev. C14, 748, 1976)

Neutron spectra resulting from bombardment of targets of ^{48}Ca, ^{90}Zr, ^{120}Sn, and ^{208}Pb with 45-MeV protons have been measured at many angles between 0° and 160°. Intranuclear-cascade Monte Carlo calculations predict too many high-energy neutrons in the forward direction and too few neutrons, particularly high-energy neutrons, at angles greater than ∿45°. Beyond 90° the underprediction is by factors of 10 to 100. For angle-integrated spectra, however, there is reasonable agreement between theory and experiment.

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‡Cyclotron Laboratory and Physics Department, Michigan State University, East Lansing, Michigan.

3. CROSS-SECTION PROCESSING, TESTING, AND SENSITIVITY ANALYSES

3.1 AMPX: A MODULAR CODE SYSTEM FOR GENERATING COUPLED MULTIGROUP NEUTRON-GAMMA LIBRARIES FROM ENDF/B*

N. M. Green† J. L. Luicus† L. M. Petrie†
W. E. Ford, III† J. E. White† R. Q. Wright†

(Abstract of ORNL/TM-3706, March 1976)

AMPX is a modular system for producing coupled multigroup neutron-gamma cross section sets. Basic neutron and gamma cross-section data for AMPX are obtained from ENDF/B libraries. Most commonly used operations required to generate and collapse multigroup cross-section sets are provided in the system.

AMPX is flexibly dimensioned; neutron group structures, gamma group structures, and expansion orders to represent anisotropic processes are all arbitrary and limited only by available computer core and budget. The basic processes provided will (1) generate multigroup neutron cross sections; (2) generate multigroup gamma cross sections; (3) generate gamma yields for gamma-producing neutron interactions; (4) combine neutron cross sections, gamma cross sections, and gamma yields into final "coupled sets"; (5) perform one-dimensional discrete ordinates transport or diffusion theory calculations for neutrons and gammas and, on option, collapse the cross sections to a broad-group structure, using the one-dimensional results as weighting functions; (6) plot cross sections, on option, to facilitate the "evaluation" of a particular multigroup set of data; (7) update and maintain multigroup cross-section libraries in such a manner as to make it not only easy to combine new data with previously processed data but also to do it in a single pass on the computer; and (8) output multigroup cross sections in convenient formats for other codes.

*Research sponsored by Defense Nuclear Agency.

†Computer Sciences Division.

3.2 PRODUCTION AND TESTING OF THE DNA FEW-GROUP COUPLED NEUTRON-GAMMA CROSS-SECTION LIBRARY*

D. E. Bartine J. R. Knight† J. V. Pace, III† R. Roussin

(Abstract of ORNL/TM-4840, February, 1977)

A state-of-the-art cross-section library has been developed for the Defense Nuclear Agency in a 37-21 neutron-gamma energy group structure that can be used for radiation transport calculations on most available computer facilities. Based on data from the DNA Working Cross Section Library and the Evaluated Nuclear Data File, the library contains cross sections for 35 nuclides that will be updated and added to regularly. The documentation presented here includes descriptions of typical sources and responses in the 37-21 structure and results of comparative calculations performed for testing purposes. The library is available through the Radiation Shielding Information Center at Oak Ridge National Laboratory.

*Research sponsored by Defense Nuclear Agency.

†Computer Sciences Division.

3.3 MODIFICATION NUMBER ONE TO THE COUPLED 100n-21γ CROSS SECTION LIBRARY FOR EPR CALCULATIONS*

W. E. Ford, III† R. T. Santoro R. W. Roussin D. M. Plaster†

(Abstract of ORNL/TM-5249, March, 1976)

The EPR ANISN-formatted 100-group neutron and 21-group gamma-ray cross-section library has been modified by the addition of data for 20 materials, by the addition of reaction cross sections for calculating tritium and helium production, and by the addition of kerma factors. The EPR 100-group master cross-section library has been modified by the addition of data for 19 materials. Procedures used to generate these cross sections and the organization of the libraries are described.

*Research sponsored by ERDA Division of Magnetic Fusion Energy.

†Computer Sciences Division.

3.4 MULTIGROUP DATA COMMONLY AVAILABLE FOR FISSION AND FUSION REACTOR SHIELDING*

R. W. Roussin

(Summary of paper to be presented at American Nuclear Society Annual Meeting, June 12-17, 1977, New York City)

Multigroup data sets commonly available for fission and fusion reactor shielding applications have both advantages and problems associated with them. A status report on multigroup sets in this particular ANS session is appropriate because recent trends are toward multigroup sets which alleviate some of the problems and are more suited for use in nuclear data assessment.

As is described at a different session[1] at this meeting, the Radiation Shielding Information Center (RSIC) has been involved since 1968 in the packaging, documentation, and distribution of multigroup data for shielding applications through its Data Library Collection (DLC).[2]

Most multigroup data sets in RSIC are of U.S. origin. However, international agreements exist which allow exchange, in most instances, of computing technology (including multigroup data). For example, the Nuclear Energy Agency Computer Program Library (NEA-CPL) in Ispra, Italy, distributes many libraries[3] obtained through RSIC, and vice versa.

A brief description of selected multigroup libraries from the RSIC Collection is given in Table 3.4.1. It can be seen that most were generated for a certain design application on analysis and were then made commonly available through a center like RSIC. A user saves substantially by avoiding the production cost if he can apply such a library to his project. Some users have no choice but to try to apply an existing set to problems for which the library was not designed. Among the common drawbacks of these problem-dependent libraries are that needed materials are missing and the wrong group structure, resonance self-shielding and temperature dependence are used.

In some older libraries, particularly coupled neutron and gamma-ray sets, inconsistencies in energy balance exist. Problems with the DLC-23/CASK and DLC-36/CLAW were recently noted by Kalra and Driscoll.[4] Recent trends are alleviating some of these problems. Later versions of the Evaluted Nuclear Data File (ENDF/B) have neutron and gamma-ray production data for many materials (not true for data sources for DLC-23 and -36), thus improving the chances for consistency in energy balance.

User needs have dictated the more flexible form of newer libraries. They contain more detailed reaction information and more versatile retrieval capability which allows the user to do his own self-shielding and temperature corrections, group collapsing, etc. With these tools, more detailed analyses can be performed, including sensitivity studies.

Examples are DLC-40/LIB-IV[5] in CCCC[6] format, DLC-43/CSRL[7] in AMPX[8] format, and the DLC-41/VITAMIN-C[9] and DLC-42/CLEAR[10] libraries in a combination of CCCC and AMPX formats.

The availability of these newer, more consistent, and more flexible libraries increases the capability of the individual user to solve a wider variety of problems. In addition, the appearance of data sets containing covariance matrices[11] (DLC-44/COVERX) and sensitivity profiles[12] (DLC-45/SENPRO) enhance his chances for participation in nuclear data sssessment.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, ERDA Division of Magnetic Fusion Energy, Defense Nuclear Agency, and Nuclear Regulatory Commission.

1. B. F. Maskewitz, D. K. Trubey, R. W. Roussin, and B. L. McGill, "The Radiation Shielding Information Center — An International Technology Resource," see in this report.

2. R. W. Roussin, "Abstracts of the Data Library Packages Assembled by the Radiation Shielding Information Center," ORNL-RSIC-30 (1972).

3. NEA-CPL, "KWIK Index of Nuclear Programs," 1976, NEA Computer Program Library, Casella Postale N.15, 21027 — Ispra (Varese), Italy.

4. M. S. Kalra and M. J. Driscoll, "Gamma Heating in LMFBR Media," COO-2250-18 (1976).

5. R. B. Kidman and R. E. MacFarlane, "LIB-IV, A Library of Group Constants for Nuclear Reactor Calculations," LA-6260-MS (1976).

6. B. M. Carmichael, "Standard Interface Files and Procedures for Reactor Physics Codes, Version III," LA-5486-MS (1974).

7. W. E. Ford, III, C. C. Webster, and R. M. Westfall, "A 218-Group Neutron Cross-Section Library in the AMPX Interface Format for Criticality Safety Studies," ORNL/CSD/TM-4 (1976).

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11. J. D. Drischler and C. R. Weisbin, "Compilation of Multigroup Covariance Matrices for Several Important Reactor Materials," ORNL/TM-5564 (1977).

12. J. H. Marable, J. L. Lucius, and C. R. Weisbin, "Compilation of Sensitivity Profiles for Several CSEWG Fast Reactor Benchmarks," ORNL/TM-5560 (ENDF-234) (1977).

Table 3.4.1. Characteristics of selected multigroup data sets from the RSIC collection

Number/Name of DLC	Contributor* (Date)	Description†	Purpose or Origin of Original Library	Validation
2/100G, 24/SINEX	ORNL (1973)	Neut. Xs, 100g, P_8; all materials; SINEX has ENDF/B-III reaction Xs	AEC/DRRD via RSIC	Reference set from ENDF/B-I-II, -III
5/HALLMARK, 14/AIR	ORNL (1971)	Coupled Xs (22, 18), P_3; air, ground	DNA air and air/ground work at ORNL	Benchmark air transport codes
8/BP-3, 8/BP-6	ORNL (1969, 1972)	Coupled Xs (22, 18), P_3; air, polyethylene	DNA methods development work at ORNL	ANS shielding benchmark Nos. 3 and 6; ORNL-RSIC-25
17/NOX, 18/NAB	ORNL (1972)	Coupled Xs (86, 33), P_3, N&O; Neut. Xs, 100g, P_3, sodium	DNA integral expt.; LMFBR integral expt.	Successful analysis of experiments
23/CASK	UCND, ORNL, SAI (1973)	Coupled Xs (22, 18), P_3; many materials	AEC/DML work at ORNL	Analysis of shipping cask
26/W-M-NRSM	WANL, NASA-MSFC (1974)	Coupled Xs (52, 13), P_3; many materials	WANL-developed data system for NASA-MSFC	Nuclear rocket shielding analysis
28/CTR	ORNL (1973)	Coupled Xs (52, 21), P_3; many materials	AEC/DCTR work at ORNL	CTR standard blanket; wide use in community
29/MACKLIB	U. WISC., ORNL (1974)	Neut. kerma, 100g; many materials	AEC/DCTR work at U. Wisc.; plus RSIC collaboration	Analysis of UWMAK fusion reactor design
31/FEWG1	ORNL (1976)	Coupled Xs (37, 21), P_3; DNA	DNA-sponsored for users	Analysis of air and concrete problems
33/MONTAGE	LASL (1976)	Neut. activation Xs, 100g; CTR structural & coolant materials	ERDA/DCTR-sponsored LASL activation library	LASL RTPR analysis; wide use in community
35/EURLIB	ESIS, IKE (1976)	Neut. Xs, 100g, P_3; many materials	European shielding benchmark program	Analysis of iron benchmark expts.
36/CLAW	LASL (1974)	Coupled Xs (30, 12), P_3; many materials	AEC/DMA weapons effects calcs. at LASL	Weapons effects analysis at LASL
37/EPR	ORNL (1975)	Coupled Xs (100, 21), P_8; many materials	ERDA/DMFE work at ORNL on experimental power reactor	Analysis of EPR fusion reactor design

Table 3.4.1 (continued)

Number/Name of DLC	Contributor* (Date)	Description†	Purpose or Origin of Original Library	Validation
40/LIB-IV	LASL (1976)	Neutron Xs, 50g; all materials; CCCC, ISOTXS, BRKOXS, DLAYXS format	ERDA/DRDD-sponsored for LMFBR core analysis	Reference set from ENDF/B-IV CSEWG; CCCC methods testing
41/VITAMIN-C	ORNL (1977)	Coupled Xs (171, 36); all materials; AMPX and CCCC formats	ERDA/DMFE-DRDD-sponsored for CTR neutronics	Selected problems in CTR community
42/CLEAR	ORNL (1977)	Coupled Xs (126, 36); all materials; AMPX and CCCC formats	ERDA/DRDD-DMFE-sponsored for LMFBR core and shield analysis	CSEWG data testing; CCCC methods testing
43/CSRL	UCND (1977)	Neut. Xs, 218g; all materials; AMPX interface format	NRC criticality safety work at Oak Ridge	Applied to U solution systems
44/COVERX	ORNL (1977)	Covariance matrices; 126g	ERDA/DRDD sensitivity program at ORNL	CSEWG fast critical benchmarks
45/SENPRO	ORNL (1977)	Sensitivity profiles; 126g	ERDA/DRDD sensitivity program at ORNL	CSEWG fast critical benchmarks

*Oak Ridge National Laboratory (ORNL), Union Carbide Nuclear Division Computing Technology (UCND), Westinghouse Astronuclear Laboratory (WANL), Marshall Space Flight Center (MSPC), U. Wisconsin (U. Wisc.), Los Alamos Scientific Laboratory (LASL), European Shielding Information Service (ESIS), Institute for Nuclear Energy, Stuttgart (IKE), Science Applications, Inc. (SAI).

†Unless otherwise noted, the library format is that used by discrete ordinates codes like ONETRAN AND ANISN.

3.5 PROGRESS ON THE VALIDATION OF THE CTR MULTIGROUP DATA PACKAGE*

R. W. Roussin C. R. Weisbin J. E. White† N. M. Greene†
R. Q. Wright† J. B. Wright

(Reprint of Trans. Am. Nucl. Soc. 23, 114, 1976)

Specifications for a general-purpose CTR Processed Multigroup Cross Section Library (PMCSL)[1,2] were developed by collaboration between CTR neutronics contractors and the Radiation Shielding Information Center (RSIC). To obtain the flexibility sought, the neutron cross section processor MINX[3] was used to generate the neutron cross sections and modules of AMPX[4] were used for the gamma-ray cross sections. The primary output format chosen was the AMPX master interface for neutron and gamma-ray cross sections with appropriate AMPX modules for manipulating, self-shielding and merging into coupled cross section libraries. Output is also provided in CCCC[5] neutron cross-section formats with coupling codes for self-shielding the CCCC neutron data and merging them with the AMPX gamma-ray data to obtain coupled cross sections. This was done by sharing production costs and computing technology with the Shielding and Reactor Physics Analysis Group at ORNL in their ERDA-DRRD sponsored cross section library generation work.[6] In order for the user to utilize the full flexibility offered, various retrieval and manipulation programs are available for merging, coupling, collapsing, editing, resonance self-shielding, and otherwise manipulating the library.

A validation effort in support of the PMCSL data package seemed appropriate because of the new types and quantities of data and retrieval programs which were written, adapted and/or modified in the course of this development. A meeting, attended by representatives of the installations listed in Table 3.5.1, was held at ORNL to discuss procedures for such a validation effort. The goals of the effort are to test the validity of the output of the neutron and gamma-ray processing codes and to demonstrate the various retrieval programs that support the library. No attempt was made to separate processing approximations from nuclear data deficiencies in the basic ENDF/B Library. The participants volunteered to perform one or more calculations chosen from the list given in Table 3.5.1, provide feedback on problems encountered with codes or data, make comparisons with calculations using other data libraries, and report the results jointly in documentation to accompany the final version of the library.

Table 3.5.1. Listing of the participants in the validation effort and the calculations which will be performed with the CTR Data Package

Participants	
Argonne National Laboratory	Oak Ridge National Laboratory
Battelle Northwest Laboratory	Princeton Plasma Physics Laboratory
Lawrence Livermore Laboratory	University of Wisconsin
Los Alamos Scientific Laboratory	Westinghouse Fusion Power Systems

Calculations to be Performed*
CTR Standard Blanket Benchmark (ORNL/TM-4177)
ANL-TEPR (ANL/CTR-75-2 and ANL/CTR/TM-51)
Japanese Experiment (Nucl. Sci. Eng., August, 1975)
Weale Experiment
LLL Pulsed Spheres (UCRL-74277, Rev. I, 1973)
RTPR Benchmark
Wyman Experiment (Texas Symposium, 1972)
TFTR
C and Fe Ring Experiments (ORNL/TM-4157, -4193)
University of Illinois Fe Sphere (Washington, D.C.)
Princeton Reference Design (MATT-1050)
UWMAK-III

*All participants will calculate the CTR standard blanket and one or more of those listed.

The wide variety of problems to be calculated will provide an extensive test of the PMCSL data package and useful comparisons of results obtained with other cross-section libraries. As an example, the first ORNL results in this effort calculated a tritium breeding ratio of 1.61 (1.03 in Li-6, 0.58 in Li-7) in the CTR standard blanket benchmark using the PMCSL compared to a value of 1.57 (0.99 in Li-6, 0.58 in Li-7) using the DLC-37/EPR[7] (generated by XLACS of the AMPX system). The reason for this difference is being investigated and the resolution of the question will be documented to provide valuable insight to the prospective user. Possible causes include differences in processing methods, high-energy weighting spectrum, and temperature used in the thermal group Maxwellian spectrum.

*Research sponsored by ERDA Division of Magnetic Fusion Energy.

†Computer Sciences Division.

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3.6 MINX, A MULTIGROUP INTERPRETATION OF NUCLEAR X-SECTIONS FROM ENDF/B*

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(Abstract of LA-6486-MS, ENDF-237, September, 1976)

MINX calculates fine-group averaged infinitely dilute cross sections, self-shielding factors, and group-to-group transfer matrices from ENDF/B-IV data. Its primary purpose is to generate pseudo-composition independent multigroup libraries in the standard CCCC-III interface formats for use in the design and analysis of nuclear systems. MINX incorporates and improves upon the resonance capabilities of existing codes such as ETOX and ENDRUN and the high-Legendre-order transfer matrices of ETOG and SUPERTOG. Group structure, Legendre order, weight function, temperature, dilutions, and processing tolerances are all under user control. Paging and variable dimensioning allow very large problems to be run. Both CDC and IBM versions of MINX are available.

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3.7 THE ROLE OF "STANDARD" FINE-GROUP CROSS SECTION LIBRARIES IN SHIELDING ANALYSIS*

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(Summary of paper to be presented at Fifth International Conference on Reator Shielding, April 18-22, 1977, Knoxville, Tennessee)

The Divisions of Magnetic Fusion Energy (DMFE) and Reactor Development and Demonstration (DRDD) of the U.S. Energy Research and Development Administration (ERDA) have jointly sponsored the development of a 171 neutron, 36 gamma-ray group pseudo-composition independent cross-section library[1-3] based upon ENDF/B-IV.[4] This library is intended to be generally applicable to fusion blanket and LMFBR core and shield analyses. The purpose of this paper is to (1) evaluate this library and its past performance[5-6] in light of ANS Shielding[7] and Reactor Physics[8] standards for cross-section data sets, (2) review the nature of the problem independence of the library in terms of group structure and weighting function selection, (3) indicate how the group-band concept[9] can be exploited to extend the range of applicability of this library, and (4) document the current limitations of and anticipated extensions to this processed data file.

Current ANS standards[8] set forth "specifications for developing, preparing, and documenting nuclear data sets." The 171/36 library was created with the intent of meeting these established procedural and technical requirements. In particular, the processing codes (and associated multigroup formats) employed have been tested and are available on both IBM and CDC computers. RSIC requirements for data packaging have ensured conformance with documentation and quality control requirements. ENDF/B-IV data were used, their primary limitations being the lack of comprehensive files for kerma factors, covariance files, and actinide and activation cross sections. The choices of group structure and weighting function were guided by the results of sensitivity studies and the details of these choices elicit the most concern regarding the range of applicability of the library. Successful results from three major ERDA-sponsored activities in data and processing methods testing by nine major laboratories and the feedback from diverse application programs indicate that the range of applicability may be broad enough to qualify this library as a generally applicable data set for fast reactor and fusion analysis.

For shielding problems characterized by a strong spatial dependence of the self-shielding factor and important energy regions with significant cross-section fluctuations, the group-band approach has been demonstrated[10] to produce more reliable results for calculated penetration. The required group-band cross sections, directly obtained from the 171/36 library, would eliminate the σ_0-ambiguity[11] inherent in a Bondarenko "standard" library by explicitly defining cross sections for any material without iteration. The usefulness of the method is tested in this study for an LMFBR shielding problem characterized by deep penetration through sodium and iron.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program, and ERDA Division of Magnetic Fusion Energy.

†Lawrence Livermore Laboratory.

‡Computer Sciences Division.

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3.8 MULTIGROUP TIME-DEPENDENT DELAYED GAMMA SPECTRA FROM FISSION AND ACTIVATION OF URANIUM AND PLUTONIUM*

O. W. Hermann† M. A. Bjerke† C. R. Weisbin

(Summary of paper to be presented at American Nuclear Society Annual Meeting, June 12-17, 1977, New York City)

The purpose of this paper is to report calculated time-dependent delayed gamma-ray spectra from neutron-induced fission and capture in isotopes of uranium and plutonium. The multigroup spectra were generated with the ORIGEN code[1] from ENDF/B-IV[2] and preliminary ENDF/B-V files in a 36-group energy structure for use in design analysis with the 126/36 LMFBR[3] and 171/36 DMFE[4] coupled cross-section libraries. Prior to this time, LMFBR reactor designers typically have not included spectral information for delayed gamma rays arising from activation of ^{238}U (i.e., neutron capture leading to ^{239}U and ^{239}Np), and have used a single ^{235}U-based spectrum for the fission gamma rays, independent of the fissioning species.[5,6] As a result of this work, isotope- and incident-energy-dependent multigroup delayed fission gamma-ray spectra, as well as delayed gamma-ray spectra from ^{238}U capture, are now in use at GE and W/ARD for the determination of in-core heating in nuclear reactor components and critical experiment analysis.

While a total gamma-ray energy yield of 6.1 MeV/fission has been used in the past by designers,[6] our multigroup library shows a broad range of energy yields for the various isotopes considered. For ^{235}U thermal fission, the total yield is still 6.1 MeV; however, we find 4.9 MeV, 6.1 MeV, 7.9 MeV, and 5.0 MeV for ^{239}Pu thermal fission, ^{235}U fast fission, ^{238}U fast fission, and ^{239}Pu fast fission, respectively. There is also a large difference between the energy yields of fission and capture, with the ^{238}U capture chain contributing 0.24 MeV/capture. Although delayed gamma-ray heating from ^{238}U capture is only $\sim$5% of the total ^{238}U heating in the LMFBR core, the assessment of ^{238}U activation is essential for blanket heating analysis.

Figure 3.8.1 is a plot of the delayed gamma-ray spectra integrals over an infinite (10^8 seconds) decay time. Various differences are evident between the delayed fission gamma-ray spectra from ^{238}U and ^{239}Pu fast fission and the previously used ^{235}U spectrum.[6]

While the ^{235}U and ^{239}Pu spectra are similar, the ^{238}U spectrum is considerably harder. The delayed gamma-ray spectrum following neutron capture in ^{238}U is also displayed in Fig. 3.8.1 and is seen to be much softer than the spectra of fission gamma rays.

The delayed gamma-ray spectra from both fission and capture are very sensitive to the limits of integration over decay time, with the mean gamma-ray energy varying by a factor of two over the time range 0 to 10^8 sec. For this reason, we have provided GE and W/ARD with separate spectra integrated over the time intervals from 0 to 10, 100, 1000, 3600, 10^5, 10^6, and 10^8 sec.

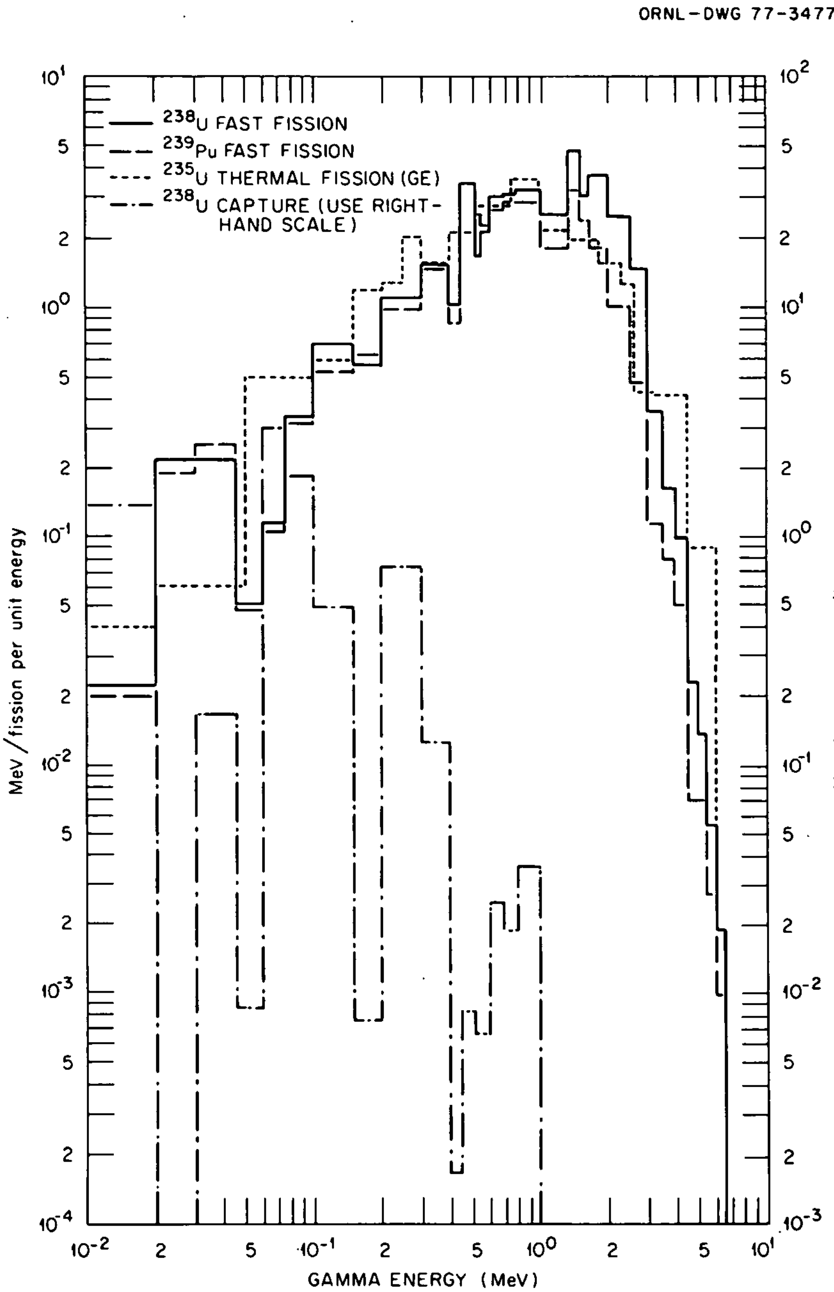

Fig. 3.8.1. Delayed gamma-ray spectra integrals over infinite time.

In conclusion, the delayed gamma-ray spectra developed in this work represent a significant addition to current multigroup cross-section libraries. It is now possible to include gamma-ray spectra separately for different decay time intervals and different fissioning or capturing species. Recent format extensions, such as the EXPOSE file, allow these data to be used in three-dimensional burnup and depletion codes such as BURNER.[7] Finally, this data set is now being applied by GE and W/ARD to in-core gamma-ray heating analyses.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

†Computer Sciences Division.

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3.9 DATA TESTING OF THE 126/36 NEUTRON-GAMMA ENDF/B-IV COUPLED LIBRARY FOR LMFBR CORE AND SHIELD ANALYSIS*

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C. R. Weisbin

(Reprint of Trans. Am. Nucl. Soc. 23, 507, 1976)

The Physics Branch of the ERDA Division of Reactor Development and Demonstration has sponsored the development of a 126/36 neutron/gamma coupled library derived from ENDF/B-IV data. The detailed specifications (e.g., selection of the tailored group structure) and processing methods have already been documented.[1] The purpose of this paper is to describe the testing program and results obtained to date in an effort to gain confidence in the library performance for a variety of important LMFBR core and shield applications. Such testing clearly involves the validity of the entire ENDF/B-IV file and the processing techniques employed; as a consequence, the validation program is rarely ever considered complete. However, since the testing for a library of this size ($\sim 3 \times 10^6$ numbers) is a prerequisite to its use and acceptance, it appears valuable to summarize the calculational experience to date which includes CSEWG Phase II data testing,[2] comparisons of processed cross sections to measurements[3,4] of gamma-ray production cross sections, RDD processing methods testing,[5] and comparison of calculations to measurements of dose through thick steel/sodium/steel shields.[6] Owing to space limitations, only some results from the first two can be described here.

Integral results for several eigenvalues and reaction rate ratios are presented in Table 3.9.1 for six fast critical assemblies. Clearly, predictions of k_{eff} for large,

Table 3.9.1. Eigenvalues and reaction rate ratios calculated using ENDF/B-IV*

	Pu Assemblies			U Assemblies		
	ZPR-6/7	ZPR-3/56B	JEZEBEL	ZPR-6/6A	ZPR-3/11	GODIVA
k_{eff} corrected						
Diffusion theory	0.9889	0.9870		0.9989	1.0080	
Transport theory	0.9885	0.9919	0.9926	0.9999	1.0053	1.0052
C/E, $^{49}f/^{25}f$	0.959	0.944				
C/E, $^{28}f/^{25}f$	0.922	0.906		0.917	1.015	
C/E, $^{28}c/^{25}f$	1.029			1.007	0.959	
C/E, $^{28}c/^{49}f$	1.073					

*These calculations do not include recent improvements in unresolved quadrature schemes suggested by ANL which have been shown to reduce the eigenvalues (e.g., k for ZPR-6/7 is decreased by ∿0.4%).

dilute Pu critical assemblies are low (∿1%) relative to those containing U. The calculated/experiment (C/E) ratio of 1.073 for $^{28}c/^{49}f$ is consistent with independently calculated overpredictions of breeding ratio (C/E breeding ratio ∿1%) relative to those containing U. The calculated/experiment (C/E) ratio of 1.073 for $^{28}c/^{49}f$ is consistent with independently calculated overpredictions of breeding ratio (C/E breeding ratio ∿1.076 of which 5.4% was due to an overprediction of $^{28}c/^{49}f$ using ENDF/B-III).[7] The central ratio and total breeding have been shown[8] to exhibit equivalent cross-section sensitivities. The integral parameters in Table 3.9.1 are also consistent with those deduced at several other laboratories.[2] (The dispersion in results between various laboratories for the eignevalue is ∿0.4%.)

As a check on the gamma-ray production cross sections in the 126/36 coupled library over the full range of incident neutron energies, a comparison was made with Fe, Na, and Ni measurements performed at the Tower Shielding Facility of ORNL. For thermal-neutron capture, there appears to be no serious discrepancy between measurements and values of gamma-ray production derived from our multigroup library. The measured and multigroup processed integrals over gamma-ray energies above 1 MeV are within the experimental error (±15%). The C/E ratios due to thermal capture in Fe, Na, and Ni were 0.991, 0.887, and 1.06 respectively. Although the integrals compare favorably with the measurements, there are deviations outside the experimental error for some gamma-ray groups. Differences well outside the estimated experimental error (±30%) are observed for gamma-ray production arising from neutron interactions above 1 MeV. The differences noted in the representative materials above are the subject of continuing study.

In addition to the fine-group composition-dependent libraries generated for each of the criticals in Table 3.9.1, a fine-group composition-dependent library for analysis of problems related to CRBR has been prepared. Also, a tailored 39/16 broad-group subset has been developed. Testing programs are currently being defined for these libraries.

The results above represent only a small sample of the experience obtained to date. In light of the generally favorable results obtained, particularly for fast reactor core analysis, the Radiation Shielding Information Center has packaged the library along with auxiliary processing programs and sample problems for general distribution.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

†Computer Sciences Division.

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3.10 GENERAL STATUS OF ACCURACY OF ENDF/IV CROSS SECTIONS FOR LMFBR SHIELDING CALCULATIONS*

R. E. Maerker

(Summary of paper to be presented at Annual Meeting of American Nuclear Society, June 12-17, 1977, New York City)

The accuracy of ENDF/IV for use in shielding calculations is far superior to that of any previous set of ENDF evaluations. Not only are the neutron total cross-section minima better represented (of the greatest importance in deep-penetration shielding calculations), but for the first time many of the more important nuclides present in LMFBR shielding configurations have gamma-ray production cross sections as well. If we limit ourselves to the nuclides of primary importance in LMFBR shielding problems, we need only consider sodium and the major constituents of inconel, mild steel and stainless steel — iron, chromium, and nickel.

At the outset we must distinguish between calculational disagreements with the results of highly integral experiments caused by using a poor group structure and/or an inaccurate transport method and using inaccurate basic data. This ambivalence is seldom easy to separate, but the group structure effect can often be ascertained by simple one-dimensional "scaling" calculations of an integral experiment using two cross section sets — one the broad-grouped set used to calculate the complete experiment and the other a fine group set sufficient to describe the important details in the basic data discovered by a sensitivity analysis. Alternatively, the basic data can be used directly in a "point" Monte Carlo calculation if the statistics in the results are adequate. Obviously, the transport calculation must have acceptable accuracy, and for the deep-penetration problems we at Oak Ridge favor the S_n method.

Generally speaking, the conclusions appearing in Table 3.10.1 are derived from simple integral experiments that do not suffer from group structure effects or transport errors in their analysis.[1-3]

Table 3.10.1. Status of ENDF/IV cross sections

Nuclide	$\sigma_{n\gamma}$ (0.0253 eV)	$\sigma_{nn'\gamma}$	σ_T
Sodium	Adqequate	Poor — needs to be re-evaluated	Inadequate near the 300-keV minimum; perhaps ∿6% too low above 1 MeV; needs to be re-evaluated*
Nickel	∿8% too high for all E_γ	Adequate	Poor; minima should be ∿20% lower in range 40 keV-4 MeV; needs to be re-evaluated
Chromium	Probably adequate	?	Poor; minima should be ∿20% lower in range 10 keV-5 MeV; needs to be re-evaluated
Iron	Generally adequate with perhaps ±20% adjustments for $1.5 \leqslant E_\gamma \leqslant 5$ MeV	Adequate	Generally adequate; perhaps ∿4% too high in the ranges 1-3.2 MeV and 4-5 MeV; minima are well represented in the range 20 keV-1 MeV

*A recent re-measurement has been performed; see D. C. Larson, J. A. Harvey, and N. W. Hill, "Measurement of the Neutron Total Cross Section of Sodium from 32 keV to 37 MeV," ORNL-TM-5614 (1976).

Despite the significant inaccuracies in the nickel and chromium cross sections indicated in Table 3.10.1, they become of far less significance when combined with iron in stainless steel because of the "filling in" of the minima by the iron. Errors in the stainless steel neutron total cross sections average from 2 to 6% high over the range 10 keV-5 MeV.

As an example of the accuracy of calculations based on ENDF/IV cross sections in a deep-penetration problem, let us consider comparisons with the upper axial shield experiment, where neutron transmissions through up to 47 cm of stainless steel followed by 460 cm of sodium and 62 cm of mild steel were measured.[4] Through up to attenuations of the order of 10^{12}, a broad-group set produced disagreements of at most a factor of 3.3, and for most of the configurations the calculations were within ∿30% of the measurements. A fine-group set reduced the maximum disagreement to only a factor of 1.7, with impact directly on inaccuracies in the basic ENDF/IV data. By folding fine-group sensitivity functions of the experiment with the error files present in ENDF/IV sodium and iron, estimates of a factor of about 2 uncertainty were obtained.[5]

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

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3.11 ACTIVITIES OF THE SHIELDING SUBCOMMITTEE OF THE ENDF/B CROSS SECTION EVALUATION WORKING GROUP*

R. W. Roussin

(Paper to be presented at Fifth International Conference on Reactor Shielding, April 18-22, 1977, Knoxville, Tennessee)

The Shielding Subcommittee of the Cross Section Evaluation Working Group (CSEWG) was established in 1968 at the request of the CSEWG chairman Sol Pearlstein to help ensure that the content of the ENDF/B cross-section library was adequate for treating shielding problems.

Early work of the subcommittee concentrated on devising formats for gamma-ray interaction and production data, as well as providing programs for testing the clerical and physics consistency of the files. The Radiation Shielding Information Center (RSIC) collaborated directly with evaluators on behalf of the National Neutron Cross Section Center (NNCSC) to begin testing and adding data sets to be fed into the official ENDF/B libraries. These efforts, which were sponsored by AEC-DRDT (now ERDA-DRDD) were augmented greatly through the Defense Nuclear Agency program of establishing a working cross-section library in ENDF format. The effort concentrated on evaluation and testing of materials of interest to DNA programs and providing these for inclusion in the ENDF/B library.

Shielding data testing efforts, as a part of the CSEWG Data Testing Program, are now also an integral part of the Shielding Subcommittee effort. Procedures for writing and approving the shielding benchmarks were devised by Shielding Subcommittee members. Data-testing benchmark experiments have been documented and analyzed, and the most recent results for ENDF/B-IV are as reported as part of ENDF-230, "Benchmark Testing of ENDF/B-IV."

Current and future activities of the subcommittee include (1) providing sensitivity coefficients for selected shielding experiments to aid ENDF/B-IV evaluators, (2) documentation and analysis of new benchmarks for fast reactor integral experiments, and (3) expanding formats and procedures as indicated by the needs of the newer community of ENDF/B users.

It is expected that much of the new activity of the Shielding Subcommittee will be on behalf of the controlled thermonuclear research (CTR) neutronics community in the area of new evaluations via the CTR Working Library and the documentation and analysis of new integral benchmarks for fusion sources and materials.

*ORNL participation sponsored by ERDA Division of Reactor Development and Demonstration and the Defense Nuclear Agency.

3.12 RECENT PROGRESS AT ORNL IN DETERMINING NUCLEAR-DATA REQUIREMENTS FOR FAST REACTOR SHIELD DESIGN USING ADVANCED SENSITIVITY TECHNIQUES*

E. M. Oblow C. R. Weisbin

(Abstract of paper presented at International Atomic Energy Agency Meeting on Differential and Integral Nuclear Data Requirements for Shielding Calculations, October 11-15, 1976, Vienna, Austria)

The sensitivity analysis of a large sodium-iron-steel integral experiment and a blanket gamma-ray heating experiment has recently been completed, allowing estimates to be made of the impact of nuclear-data uncertainties on fast-reactor shield design. Preliminary error files for ENDF/B-IV iron and sodium data were used for the first time to estimate the effect of data uncertainties on shield performance. An initial attempt was also made to adjust iron and sodium cross-section data so that the calculated and measured integral results were in agreement. Significant discrepancies in both the iron and sodium data above 1 MeV are indicated by these studies and supported by the analyses of other relevant integral experiments. Recent measurements of the sodium total cross section in the several hundred keV energy range have also had a large effect on the calculated results. A full discussion of the relevant discrepancies found in the analyses of several integral experiments will be presented in the full paper.

In the course of the above-mentioned studies new methodologies for sensitivity analyses, uncertainty estimation, cross-section adjustment, and determining nuclear data requirements for shield design were developed and tested. For sensitivity work in general, the basic concepts of "channel theory" were recently worked out to illustrate the paths taken trhough a shield system by the radiation ultimately responsible for the detector response of interest. In addition, cross-section adjustment procedures analogous to those developed in Italy have for the first time been implemented in the FORSS code system at ORNL. These procedures allow estimates to be made of the cross-section data responsible for discrepancies found in analyzing integral experiment results. Procedures have also been worked out to use adjustment to estimate the effects of integral measurements and differential cross-section data in meeting design constraints on shield performance. All of these new areas will be discussed more thoroughly in the full paper to be presented.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

3.13 FAST REACTOR SHIELD SENSITIVITY STUDIES FOR STEEL-SODIUM-IRON SYSTEMS*

E. M. Oblow C. R. Weisbin

(Summary of paper to be presented at Fifth International Conference on Reactor Shielding, April 18-22, 1977, Knoxville, Tennessee)

A sensitivity analysis of a number of integral measurements on a heterogeneous 6-m steel-sodium-iron fast reactor shield mockup was performed to determine the adequacy of the basic nuclear cross-section data for sodium and iron. The study was based on a fine-group ENDF/B-IV neutronics cross-section library and utilized the complete sensitivity analysis capabilities of the FORSS code system. Preliminary ENDF/B-V error files for sodium and iron were also used to determine calculational uncertainties for the analysis.

The initial results of the sensitivity study, given in the form of sensitivity profiles, indicated that the measurements were sensitive to a wide range of nuclear data covering an energy range from a few eV to 10 MeV. Such measurements, therefore, represent excellent benchmarks with which to test calculations of fast-reactor shield performance. Several individual cross-section features which were found to be important included: high-energy (5-10 keV) anisotropic elastic scattering in sodium and steel, the 500-keV

and 300-keV total cross-section minima in sodium, the 24-keV total cross section minimum in iron, and elastic slowing down cross sections from 1 keV to 1 eV in both sodium and iron.

In the uncertainty analysis, the sodium and iron error files were used to estimate the confidence limits within which the measured neutron transmissions could be calculated. Results here indicate that in most cases the cross-section uncertainties lead to uncertainties in the measured integral data of 100% or more. Such large uncertainties make it difficult to meet shield design constraints based solely on calculated results using differential data only; integral experiments are still needed to reduce uncertainties.

In the final phase of study, the reported disagreements between calculated and measured results were used as the basis for a trial cross-section adjustment. This step was taken to see what changes could be made in the basic nuclear data to significantly improve the agreement. Results here indicate that reasonable changes (within the reported cross-section uncertainty bounds) could be made to the high-energy elastic cross-section data for iron and sodium and the total cross sections of these two materials in the vicinity of the minima at 500 keV, 300 keV, and 24 keV. New differential cross-section measurements for sodium and integral measurements for iron in these energy ranges support these general conclusions.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

3.14 APPLICATION OF FORSS SENSITIVITY AND UNCERTAINTY METHODOLOGY TO FAST REACTOR BENCHMARK ANALYSIS*

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(Reprint of Trans. Am. Nucl. Soc. 24, 455, 1976; also summary of ORNL/TM-5563, ENDF-236, December, 1976)

FORSS[1] is a code system used to study relationships between cross sections, integral experiments, performance parameter predictions, and associated uncertainties. This paper presents the first results of applying FORSS to fast-reactor benchmarks. Specifically, for various assemblies and performance parameters, the nuclear data sensitivity is computed by nuclide, reaction type, and energy. Nuclear data accuracy requirements to meet specified performance criteria at minimum experimental cost are determined. Uncertainties induced by nuclear data are quantified using preliminary, energy-dependent relative covariance matrices (evaluated assuming ENDF/B-IV) which were not included in earlier reported work.[2-14] Results of integral experiments are incorporated resulting in improved uncertainty estimates of reactor performance for devices being designed.

The sensitivities of performance parameters for the ZPR-6/6A, ZPR-6/7, and GODIVA CSEWG fast-reactor benchmarks[15] to important partial cross sections were computed using transport theory in 126 groups. For example, the total relative sensitivity of k in GODIVA to ^{235}U fission is 0.659, while the total relative sensitivity of the central ^{238}U-capture-to-^{239}Pu-fission ratio in ZPR-6/7 to ^{239}Pu fission is -1.073. Comprehensive libraries[16] of energy-dependent coefficients in a computer retrievable format[17] have been documented and released for distribution by RSIC and NNCSC.

Recently, standard formats and procedures were established within the ENDF/B system[18] for the processing[19] of evaluated, correlated energy-dependent uncertainty information into multigroup covariance matrices. Files[20] have been processed for ^{238}U(n,f), ^{238}U(n,γ), ^{239}Pu(n,f), ^{239}Pu(n,γ), and ^{239}Pu$(\bar{\nu})$. [The current file for ^{239}Pu includes[21] estimates of the correlated uncertainties to reference standards, conservatism where the measurements are sparse, and ratio measurement correlations between ^{239}Pu(n,f) and ^{239}Pu(n,γ)].

The determination of the accuracy required of multigroup nuclear data to ensure at minimum cost a given accuracy in calculated performance parameters is characterized by the following: (1) a set of performance parameters R_j with specified accuracy requirements V_j, (2) a set of proposed experimental cross sections with estimated correlation matrix D and variable standard-deviation vector x to be determined, (3) a set of sensitivity coefficients S_j relating the cross sections to performance parameters R_j and (4) a cost function $C(x)$ which gives an estimate of the cost of experimental measurements as a function of the variable standard deviations x. The optimization calculation finds the standard deviations required of the experimental measurements in order to find a minimum cost subject to the constraints imposed by performance parameter accuracy requirements. Using an existing code for nonlinear optimization with nonlinear constraints,[22] the following problem is solved:

Find x which minimizes $C(x)$ subject to the constraints

$$\tilde{S}_j \, \overline{X} \, D \, \overline{X} \, S_j \leq V_j \; .$$

Here $\tilde{S}_j$ is the transpose of S_j and $\overline{X}$ is a diagonal matrix, the diagonal elements of which are components of the vector x. Based on our estimated correlations and standard deviations, a conclusion to be drawn is that design accuracy goals of 0.5% in k and 2% in the central $^{28}c/^{49}f$ ratio in conventional mixed oxide LMFBR cores (neutronic behavior similar to ZPR-6/7) are unlikely to be attained in the next 5-10 years if the nuclear data is based only on microscopic measurements and the level of effort on the most crucial cross sections is not increased dramatically. Incorporation of integral-experiment results is required, directly or indirectly, in calculating reactor performance.

Inclusion of integral measurements in uncertainty estimation introduces correlations (heretofore unknown) in the covariance data which reduce the overall uncertainty[6,14] for designs with similar neutronic sensitivities. Consider, for example, the measurements in ZPR-6/7 of k and of central ^{238}U-capture/^{239}Pu-fission with assigned uncorrelated standard deviation (1σ) of 1 and 2 percent, respectively. Standard deviations in calculations of these two parameters for similar systems are reduced from 3.5% and 8.6% to 0.82% and 1.8%, respectively, when the integral data are included in a cross-section adjustment, which changes the basic multigroup file less than one standard deviation. Elimination of any significant downward adjustment in the ^{238}U capture cross section can be effected only by significant upward revision in the reported ZPR-6/7 capture/fission ratio and/or its uncertainty. Changes to k may be accounted for without resort to large changes in the ^{238}-capture cross section by including additional reaction types (e.g., ^{238}U inelastic) in the adjustment.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

†Computer Sciences Division.

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3.15 THE IMPORTANCE OF NEUTRON DATA IN FISSION REACTOR APPLICATIONS*

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R. Roussin C. Weisbin

(Abstract of paper presented at International Conference on the Interactions of Neutrons with Nuclei, July 6-9, 1976, Lowell, Massachusetts, and published in ERDA TIC report CONF-760715-P2, p. 1127)

The neutron data required to completely analyze fission reactors includes many isotopes and covers a broad energy range. In both fast and thermal reactors, the neutron inventory is a fine balance determined by the fission properties of ^{235}U, ^{239}Pu, and ^{238}U and by the capture cross sections of ^{238}U, fuel materials, structural materials and coolant materials. In fast reactors, the spectrum of neutrons ranges from 1 keV to 3 MeV and is influenced by the elastic and inelastic scattering properties of ^{238}U and the structural and coolant materials. For neutron shielding applications, the important neutron data includes the total cross sections of structural and coolant materials in the MeV range.

The impact of these basic nuclear data in fission reactor applications is most suitably described by sensitivity analysis. For example, sensitivity coefficients computed for a typical large plutonium-fueled fast reactor indicate that a percent increase in the $^{239}Pu(n,f)$ cross section translates into a 0.59% increase in k, a 0.78% decrease in the breeding ratio, and 0.71% decrease in the sodium coolant reactivity worth.

Integral data tests of ENDF/B-IV on the thermal-reactor benchmarks indicates that there are no major deficiencies in the H_2O and ^{235}U cross sections for thermal systems. However, k_{eff} is underpredicted for lattices of slightly enriched systems with an indication that epithermal-to-thermal ^{238}U capture is overpredicted. Fast-reactor benchmark tests generally yield a less reactive system for plutonium-fueled reactors compared to uranium-fueled reactors and the capture rate in ^{238}U relative to the fuel fission rate is generally overpredicted in large systems. Shielding benchmark tests indicate a wide range of deficient neutron data, especially the total elastic and inelastic cross sections of iron, oxygen, and sodium in the high-energy range.

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3.16 COMPILATION OF SENSITIVITY PROFILES FOR SEVERAL CSEWG FAST REACTOR BENCHMARKS*

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(Abstract of ORNL-5262, ENDF-234, March, 1977)

Sensitivity profiles of the multiplication factor and several central reaction rate ratios for five CSEWG fast-reactor benchmarks, ZPR-6/7, ZPR-6/6A, ZPR-3/11, ZPR-3/56B, and GODIVA, are presented in graphic form and in tabular form using SENPRO format. A summary table of total relative sensitivities for each of these five benchmarks is also given.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

†Computer Sciences Division.

3.17 PERFORMANCE PARAMETER UNCERTAINTIES FOR A LARGE LMFBR*

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(Summary of paper to be presented at Annual Meeting of American Nuclear Society, June 12-17, 1977, New York City)

This paper reports uncertainties in performance parameters of a large LMFBR which result from uncertainties in both nuclear data and relevant integral experiments. The current effort is the first analysis of this type actually applied to a realistic two-dimensional model of an LMFBR using evaluated covariance files based upon ENDF/B-IV.

A working model of a 1200-MWe LMFBR was the basis of a FORSS[1] system sensitivity and uncertainty analysis. A four-group cross-section set was used in conjunction with the VENTURE[2] diffusion theory code to obtain sensitivity coefficients for the multiplication factor k and the breeding ratio. The sensitivities were obtained with respect to ^{238}U(n,γ), ^{238}U(n,f), ^{239}Pu(n,γ), ^{239}Pu(n,f) and the neutron yield $\bar{\nu}$ in ^{239}Pu. Previous studies[1] have shown these to be among the most important cross sections for these two performance parameters.

Calculational and experimental results from six integral experiments designated as CSEWG benchmarks[3] were included in the analysis. These were ZPR-6/7 k, (^{28}c/^{49}f), (^{28}f/^{49}f), and ZPR-3/56B k, (^{28}f/^{25}f), (^{49}f/^{25}f), where the parentheses imply central reaction rate ratios for fission (f) and capture (c) respectively. Sensitivity coefficients for these were obtained in the four-group structure by collapsing previously obtained 126-group coefficients.[4]

Covariance files for the five nuclear data types as determined from basic differential cross-section measurements were used to obtain standard deviations for the multiplication factor and breeding ratio (BR) of the 1200-MWe model as shown in Table 3.17.1. One-standard-deviation estimates for k and BR due exclusively to uncertainties in the five nuclear data types are 3.1 and 6.9% respectively. After including the integral experiment results from ZPR-6/7 and ZPR-3/56B in an adjustment procedure which minimized the weighted differences between calculated and measured values, a new covariance matrix was obtained. By folding the sensitivity profiles with the revised covariance matrix, the calculated standard deviations for k and BR of the 1200-MWe model were reduced to 0.97 and 3.2% respectively. The considerable reduction in the calculational uncertainty depends strongly on the uncertainties assigned to the integral experiments (taken here as 1% for k and 2% for central reaction rate ratios).

Table 3.17.1. Relative standard deviations of performance parameters in percent

Reactor System	Multiplication Factor	Breeding Ratio[a]	(^{28}c/^{49}f)	(^{28}f/^{49}f)	(^{28}f/^{25}f)	(^{29}f/^{25}f)
ZPR-6/7	3.67		8.19	4.96		
ZPR-6/7[b]	0.66		1.71	1.33		
ZPR-3/56B	3.62				3.49	6.23
ZPR-3/56B[b]	0.61				1.41	1.44
1200 MW(e) model	3.14	6.93				
1200 MW(e) model[b]	0.97	3.23				

[a]Breeding ratio is here defined as (capture in fertile)/(absorptions in fissile): (^{28}c + ^{40}c)/(^{49}a + ^{41}a + ^{25}a).

[b]Based on covariances of cross sections adjusted to include the data of the integral experiments.

Conclusions to be drawn from these calculations are: (1) Integral experiments are presently required in the design of large LMFBR-type reactors in order to achieve acceptable standard deviations in performance parameters; (2) error analysis of the integral experiments are extremely important and have a direct bearing on the standard deviations of the calculated performance parameters; and (3) based on current estimated covariance files for differential and integral data and based on our calculated 2-D sensitivity coefficients, the uncertainties in k and BR performance for the 1200-MWe LMFBR model are 1% and 3%, respectively.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

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3.18 SENSITIVITY ANALYSIS OF TRX-2 LATTICE PARAMETERS WITH EMPHASIS ON EPITHERMAL ^{238}U CAPTURE*

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(Summary of paper to be presented at Annual Meeting of the American Nuclear Society, June 12-17, 1977, New York City; also summary of report to be published as EPRI-RP 612)

Overprediction of epithermal ^{238}U capture in water-moderated lattices using ENDF/B data has been a long-standing problem[1] basic to the prediction of neutron economy and conversion ratio in light-water reactors.

The objective of this paper is the determination of a recommended representation of the ^{238}U capture cross section based upon available differential and specific integral data and the quantitative determination of the sensitivity of thermal uranium lattice (TRX-2) performance parameters[2] to the cross-section shape, magnitude, and representation with emphasis on the first four resolved s-wave resonances. Sensitivity profiles and covariance matrices developed for ^{238}U capture and ^{235}U fission permit a quantitative assessment of performance parameter uncertainties due to concomitant uncertainties in nuclear data.

The ENDF/B-IV ^{238}U cross sections[3] were modified based on recent measurements and analysis not available to the ENDF/B-IV evaluators.[4,5,6] The change from ENDF/B-IV to the recommended cross sections is described below.

The values of the capture widths of the levels at 6.67 eV, 20.9 eV, and 36.8 eV were changed from their ENDF/B-IV values of 25.6 mV, 26.8 mV, and 26.0 mV, respectively, to 23.0 mV. The values of the neutron widths of the levels at 20.9 eV and 36.8 eV were changed from their ENDF/B-IV values of 8.8 mV and 33.1 mV to 10 mV and 33.5 mV, respectively. Minor modifications were done to the scattering and capture smooth files,[6] and the resolved s-waves levels were treated with the multilevel Breit-Wigner formula[7] rather than the single-level formula.

A tentative adjusted set of cross-section data was obtained, which is consistent with the microscopic measurements and minimizes the weighted differences between our calculated and the measured values of the performance parameters for TRX-2. The actual adjustments to the recommended data were minimal such that the recommended evaluation, based solely upon differential measurements, could be used without adjustment while maintaining consistent agreement with calculated TRX-2 performance.

In Table 3.18.1 we compare the experimental values of the main TRX-2 performance parameters with values computed with ENDF/B-IV and with the "recommended" and the "adjusted" sets of cross sections.

Table 3.18.1. Experimental and calculated values of TRX-2 performance parameters

Parameter	Experimental	ENDF/B-IV	Recommended Data Set	Adjusted Data Set	Sample Methods Difference
k_{eff}	1.0000	1.0012	1.0046 ± 0.0035	1.0003 ± 0.0009	±0.009
$^{28}\delta$	0.837 ± 0.016	0.867	0.843 ± 0.007	0.845 ± 0.006	±0.02
$^{25}\delta$	0.0614 ± 0.0008	0.0602	0.0602 ± 0.0012	0.610 ± 0.0006	±0.0009
$^{28}\delta$	0.0693 ± 0.0035	0.0698	0.0695 ± 0.0003	0.0698 ± 0.0001	±0.003
CR	0.647 ± 0.006	0.645	0.637 ± 0.005	0.643 ± 0.003	±0.006

The uncertainties given for the computed quantities are due to uncertainties in nuclear data and were obtained from the FORSS system[8] by folding the sensitivity profile with the error covariance matrix.[9,10] A "sample methods difference" was obtained by comparing two independent calculations[11] using the same data set and may be interpreted as an estimate of the error in the calculation due to "methods" (geometric modeling, multigroup treatment, etc.).

The main conclusions of this study are that the results of recent measurements suggest a modification of the ENDF/B-IV representation of the low-energy cross sections of ^{238}U; calculations of TRX-2 performance parameters made with the recommended cross sections yield results in fair agreement with the measurements; and uncertainties due to methods are as large or larger than those due to nuclear data.

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3.19 ACTINIDE TRANSMUTATION: CROSS SECTIONS, METHODS AND REACTOR SENSITIVITY STUDIES*

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(Reprint of Trans. Am. Nucl. Soc. 23, 552, 1976)

One of the potential techniques of disposal of long-lived radioactive actinide wastes from fission reactors is the transmutation of these nuclides into fissile isotopes in a reactor. Subsequent fission yields highly radioactive but relatively short-lived fission products which can be disposed of using conventional storage techniques. In order to evaluate the feasibility of this method of disposal, one needs to quantify the effect of the transmutation on the actinide hazard and the sensitivity of the fission reactor performance parameters to the actinide recycle.

Some preliminary investigations in the area of actinide transmutation have been done previously.[1-6] However, much of the nuclear data available at the time of these calculations can best be described as incomplete. In addition, the calculations used a simplified reactor model. Recently the Savannah River Laboratory[7] has prepared a new evaluation of actinide nuclear data which provides considerably more information. The purpose of this paper is to describe the results of a feasibility study of this actinide transmutation technique using this more recent evaluation. In particular, we:

1) Develop multigroup actinide libraries based upon ENDF/B-IV supplemented by the recent SRL evaluated data.

2) Specify the reactor models to be utilized for detailed transmutation and sensitivity studies.

3) Compare actinide generation and production results using different reactor models (i.e., the zero-dimensional approximation employed by ORIGEN[8] versus the more rigorous space-dependent depletion capability of CITATION[9]).

4) Perform sensitivity analysis of the effect of actinide recycle on reactor performance parameters such as k_{eff}, breeding ratio, etc.

Two multigroup cross-section libraries were developed using MINX,[10] a 126-group set for fast reactor studies and an 84-group thermal reactor library. Isotopes ranging from ^{232}Th to ^{253}Es were considered. The cross sections have ben checked both visually and by comparison with existing ORIGEN files. [Resonance integrals were verified where applicable.] It must be pointed out, however, that even these libraries are clearly preliminary in nature since there are many energy regions for which experimental data are nonexistent.

The effect of actinide recycle on reactor operating performance parameters was analyzed using the generalized linear perturbation capability of the FORSS[11] system. The effects of both the quantity of recycled actinides and their placement within the "transmutation" reactor were considered. Figure 3.19.1 illustrates the sensitivity of the breeding ratio in a simplified (two-region) fast-reactor model to the replacement of ^{238}U by ^{241}Am on an atom for atom basis. [Similar energy-dependent profiles are available for all nuclides in the actinide chain.] The total fractional change in the breeding ratio (per kg ^{241}Am substituted) amounts to -8.0×10^{-5}, with 93% of the effect attributable to the ^{238}U removal. Results for other reactor parameters as well as other actinide nuclides have also been obtained.

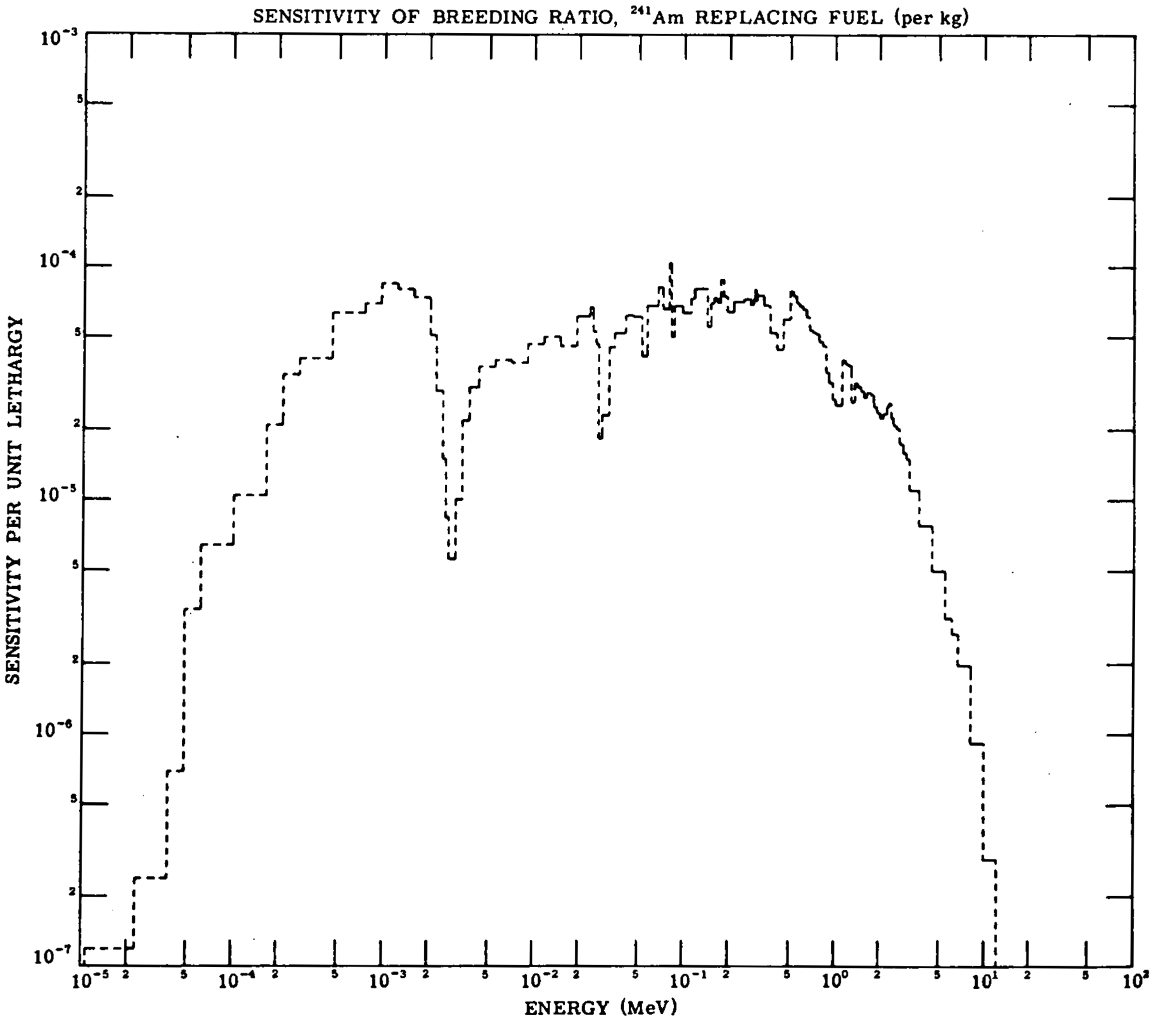

Fig. 3.19.1. Breeding ratio sensitivity profile. Solid lines indicate negative values.

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3.20 PROPAGATION OF UNCERTAINTIES IN FISSION CROSS SECTION STANDARDS IN THE INTERPRETATION AND UTILIZATION OF CRITICAL BENCHMARK MEASUREMENTS*

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(Abstract of paper to be presented at International Specialists Symposium on Neutron Standards and Applications, March 28-31, 1977, National Bureau of Standards, Gaithersburg, Maryland)

This work explores the constraints imposed on the ^{235}U(n,f) standard (proposed ENDF/B Version V) by information deduced from clean integral measurements and demonstrates how uncertainties in fission cross section standards propagate in an uncertainty analysis and interpretation of those experiments. The question of what a significant improvement in the accuracy of the ^{235}U(n,f) standard would accomplish is addressed in the limited context of analyses of GODIVA and JEZEBEL measurements.

The CSEWG integral benchmark results and uncertainties were updated in accordance with more recent information. Sensitivity coefficients were developed and used to estimate calculated results which should be obtained using the subsequent release of ^{238}U(n,f), ^{235}U(n,f), and ^{239}Pu(n,f) at version V status. Covariance files were evaluated and processed for all important cross sections with the sole exception being inelastic scattering for all levels and the continuum.

Uncertainties due to the $^{235}U(n,f)$ standard were estimated to comprise more than half of the calculated uncertainty for criticality and $\langle {}^{28}\sigma_f / {}^{25}\sigma_f \rangle_c$ spectral index in JEZEBEL as well as GODIVA, though the JEZEBEL assembly contained no ^{235}U. We are not able, at this time, to predict criticality or $\langle {}^{28}\sigma_c / {}^{28}\sigma_f \rangle_c$ to anywhere near the accuracy obtained by direct measurements, and therefore the integral results are significant to our analysis capability. Inclusion of integral information from GODIVA and JEZEBEL in an adjustment procedure was effective in reconciling all parameters other than $\langle {}^{28}\sigma_f / {}^{25}\sigma_f \rangle_c$ measurement in JEZEBEL for which current calculations and measurement are in disagreement. The adjustment procedure made changes of less than one standard deviation in the cross sections for $^{235}U(n,f)$, $^{235}U(n,\gamma)$, $^{238}U(n,f)$, $^{238}U(n,\gamma)$, and $^{239}Pu(n,f)$, including an increase of $\sim$1.5% for the $^{235}U(n,f)$ cross section above 1.3 MeV. This specific adjustment result could change with inclusion of inelastic covariance files and must be viewed cautiously at this time.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

3.21 FUTURE DIRECTIONS FOR SENSITIVITY-BASED DATA ASSESSMENT*

C. R. Weisbin E. M. Oblow J. H. Marable

(Summary of paper to be presented at Annual Meeting of American Nuclear Society, June 12-17, 1977, New York City)

The purpose of this paper is to identify priority development requirements for sensitivity-based uncertainty estimation and data assessment. The key research areas discussed arise from near-term needs in the ORNL FORSS[1] development effort but are obviously common to investigations under way at RPI,[2] ANL,[3] LASL,[4] and elsewhere. The application areas are broad and include fission and fusion, core and shield analysis.

Significant expansion of the evaluated[5] and processed[6] covariance files is essential for further development in this field. ENDF/B-V is expected to include uncertainties for a large number of materials as well as cross-section uncertainties induced through ratio measurements relative to standards and uncertainties in individual resolved resonance parameters. Covariance file data and formats are needed[7] for secondary energy and angular distributions as well as unresolved resonance region parameters. Effective uncertainty analysis absolutely requires[1] that a complementary effort be undertaken to quantify uncertainties associated with integral experiments. In addition, cost/benefit analysis of new differential and integral measurements calls for[8,9,1] quantitative estimates and functional dependence of the projected cost for each new measurement. Reasonable results for the specification of cross-section measurement priorities will only be achieved if there are clearly stated design criteria and the knowledge deduced from integral experiments is factored into the "inverse problem" (ref. 1) solution.

Intensified effort is required in the identification and separation of methods and data uncertainties. Techniques for generating sensitivities to nuclear data file parameters[10] (rather than group-averaged quantities) will receive increasing attention, as will the definition and inclusion[11] of methods bias in the adjustment process. Production use of 2-D and even 3-D sensitivity analysis is vital for application to real design problems. The methodology for collapsing sensitivity profiles and covariance matrices needs to be expanded. Moreover, it may be possible to develop production capability which extends the accuracy of sensitivity analysis beyond first-order perturbation theory.[12] This should be explored. Methods for generating sensitivity profiles including constraints (i.e., k-reset) for existing systems (e.g., critical assemblies) need to be investigated.

With the development of such a powerful analytical tool, the desire to broaden the scope of its application to related, and more complicated problems, is inescapable. Extension of sensitivity analysis into the time domain[13,2] for fuel cycle analysis is clearly important. Automated shield optimization techniques are being developed[12] and will be extended to core analysis. Sensitivity theory is now being applied to the optimal selection of multigroup energy boundaries.[14,15] Finally, no discipline would be complete without adequate documentation of its computer codes, sample problems, standard libraries, etc. An associated difficulty is simply coping and preserving the huge amount of information being generated and analyzed.

Sensitivity and uncertainty analysis has expanded greatly in the last five years. Much has been done, and much remains to be done.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

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4. INTEGRAL EXPERIMENTS AND THEIR ANALYSES

4.1 ANALYSIS OF THE TSF-FFTF INCONEL EXPERIMENT: CALCULATIONS OF NEUTRON TRANSPORT AND SECONDARY GAMMA-RAY PRODUCTION AND TRANSPORT IN INCONEL*

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(Abstract of ORNL/TM-4959, May, 1976)

This report describes the analysis of an experiment performed at the ORNL Tower Shielding Facility to investigate the neutron-attenuating and gamma-ray-producing properties of inconel and to provide an experimental base for testing the cross sections of nickel, the primary constituent of inconel. In the experiment, rectangular slabs of inconel 2-1/2 and 5 in. thick were positioned in a reactor beam behind a spectrum modifier and neutron and gamma-ray measurements were made at various locations beyond the configurations. In the analysis, the configurations were mocked up in cylindrical geometry, with the cylinder axis coinciding with the beam axis. Calculations of the neutron count rates and energy fluxes were performed with the DOT discrete ordinates code in combination with the analytic first-collision code GRTUNCL and the last-collision code FALSTF. The calculated Bonner ball count rates were 7 to 37% higher than those measured on the axis 24 in. behind the configurations, the difference being greater for the thinner configuration. Similar discrepancies occurred between the calculated and measured hydrogen counter energy fluxes 12 in. behind the configurations, but for the higher energies covered by the NE-213 spectrometer, the calculated neutron energy fluxes for a position 19 ft behind the configurations were lower than the measured fluxes and the discrepancies were greater for the thicker configuration.

Calculations of gamma-ray energy fluxes, performed for an NaI spectrometer location 45 deg from the reactor beam, were divided into two parts: DOT calculations of the secondary gamma rays produced within the configurations; and OGRE Monte Carlo calculations of core gamma rays scattered from the configurations. The calculated and measured total fluxes for both configurations agreed within ±30% except at high energies, where the calculations were 60 to 65% too high. This indicated that the gamma-ray production cross sections for nickel were adequate except for the 8.5-MeV and/or 9-MeV capture gamma rays.

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4.2 ANALYSIS OF THE TSF THREE-DIMENSIONAL STORED-FUEL EXPERIMENT FOR THE CRBR*

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(Abstract of ORNL-5187, September, 1976)

This report describes the analysis of a Tower Shielding Facility experiment performed to study the transport of radiation through a mockup of an early shield design for the Clinch River Breeder Reactor, the radiation source being a collimated beam of reactor neutrons modified to have the energy spectrum of a fast reactor. The mockup covered the radial regions between the CRBR core and the reactor vessel wall, including a region outside the radial shield in which fuel assemblies were to be stored temporarily. Of particular concern in this design was the enhancement of the fast-neutron fluxes outside the vessel wall due to fissions in the stored fuel. The experiment yielded direct information on the degree of enhancement and its analysis served as a test of the calculational technique and nuclear data used to predict stored-fuel contributions to the neutron fluxes in the CRBR reactor cavity. In the technique the two-dimensional discrete ordinates code DOT is coupled to the three-dimensional Monte Carlo code MORSE, the latter required to accurately describe the geometry of the stored-fuel region. When this approach was applied to the experimental configuration, the calculated and measured fluxes

were in general agreement. However, for some configurations the calculated fluxes were 30% lower than the measured fluxes, and the cause of this inconsistency has not yet been determined.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

†Computer Sciences Division.

4.3 TSF MEASUREMENTS OF NEUTRON TRANSMISSION THROUGH AN AI-LMFBR LOWER AXIAL SHIELD MOCKUP*

C. E. Clifford F. J. Muckenthaler P. N. Stevens†

(Abstract of ORNL-5178, in press)

This report describes the results of an experiment performed at the ORNL Tower Shielding Facility to measure neutron transmission through a mockup of a blanket-shield assembly designed by Atomics International for the lower axial region of the Liquid Metal Fast Breeder Reactor demonstration plant. The neutron source for the experiment was a collimated beam from the Tower Shielding Reactor II modified to simulate the energy spectrum of a fast reactor. The mockup contained a UO_2 blanket region and a sodium-flow transition region, both of which were penetrated by a cylindrical plug simulating various conditions in a central control-rod channel, plus upper and lower grid plates separated by a sodium plenum region. The objectives of the experiment were (1) to provide an experimental base to evaluate the adequacy of calculational techniques for predicting neutron streaming through control rod penetrations and (2) to determine the neutron attenuation in the unperturbed lower axial shield. Measurements were made of the fluence and energy spectra of neutrons both within and emerging from several different configurations comprised of sections of the mockup. The complete set of data obtained in the experiment are presented here.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

†The University of Tennessee.

4.4 ANALYSIS OF TSF EXPERIMENT WITH AI-LMFBR LOWER AXIAL SHIELD MOCKUP*

R. L. Childs† V. C. Baker‡ F. R. Mynatt Lorraine S. Abbott

(Abstract of ORNL-5179, July, 1976)

This report describes the analysis of an experiment performed at the ORNL Tower Shielding Facility with a mockup of a blanket-shield assembly designed by Atomics International for the lower axial region of the Liquid Metal Fast Breeder Reactor demonstration plant. The neutron source for the experiment was a collimated beam from the TSF reactor modified to simulate the energy spectrum of a fast reactor. The mockup contained a UO_2 blanket region and a sodium-flow transition region, both of which were penetrated by a cylindrical plug simulating various conditions in a central control-rod channel, plus upper and lower grid plates separated by a sodium plenum region. Measurements of the neutron energy flux were made with various detectors beyond the blanket and transition regions and at locations within the full configuration. The analysis was performed with the DOT-III discrete ordinates transport code in two-dimensional cylindrical geometry. Calculated detector responses were in good agreement with the measured responses in the energy region from 10 keV to 1.4 MeV and were in reasonable agreement at higher energies (within 30%). At lower energies, however, the calculated responses were lower than the measured responses by factors of two to five, and the reason for the disparity has not yet been determined. Applying stainless steel damage factors to the calculated fluxes

beyond two of the configurations indicated that neutron streaming through a central sodium-filled control-rod channel in the lower UO_2 blanket region of a reactor could considerably increase the radiation damage along the axis of a stainless steel grid plate.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, CRBR Program.

†Computer Sciences Division.

‡The University of Tennessee.

4.5 EXPERIMENTAL STUDIES OF RADIATION HEATING IN IRON AND STAINLESS STEEL SHIELDS FOR THE CRBR PROJECT*

C. E. Clifford F. J. Muckenthaler R. E. Maerker P. N. Stevens† R. K. Abele‡
G. G. Simons§ T. J. Yule‖ M. J. Driscoll#

(Abstract of ORNL-5161, February, 1977; preliminary report issued as ORNL/TM-4998, January, 1976)

This report describes a series of radiation heating measurements in heavy-metal shield configurations performed at the Oak Ridge National Laboratory. Simultaneous measurements were made by personnel from the Argonne National Laboratory, from the Massachusetts Institute of Technology and from the Oak Ridge National Laboratory. The experimental techniques being evaluated involved the use of calcium fluoride (CaF_2) and lithium fluoride (^7LiF) thermoluminescent dosimeters (TLDs) and a high-pressure argon-krypton filled ion chamber designed and built by the Oak Ridge National Laboratory Instrumentation and Controls Division. Intercomparison of the experimental results and comparison with calculation provides a meaningful evaluation of these experimental techniques for measuring the energy depositions.

It was shown that the ^7LiF-TLD measurements and the ion chamber results were about 14% higher than the CaF_2-TLD measurements. The calculations were found to be about 2-5% lower than the CaF_2-TLD measurements for the iron configuration and about 15-20% lower than the CaF_2-TLD measurements for the stainless steel configuration. Even greater disparities were found between calculation and measurement for the ^7LiF-TLD and ion chamber dosimeters.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

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4.6 ANALYSES OF TSF EXPERIMENTS ON RADIATION HEATING IN IRON AND STAINLESS STEEL CRBR RADIAL SHIELD MOCKUPS USING A 15-1/4-in.-DIAMETER COLLIMATED BEAM SOURCE*

R. E. Maerker T. J. Burns

(Abstract of ORNL/TM-5420, June, 1976)

This report describes the results of discrete ordinates calculations using a standard design cross section set that were made of the neutron and gamma-ray fluxes throughout several of the iron and stainless steel configurations in the CRBR radial shield mockup

radiation heating experiments performed at the Tower Shielding Facility in 1974. Comparisons of the calculated heating rates with the ORNL-CaF_2 TLD measured values show agreement to within ±10% throughout the iron configuration and ±25% throughout the stainless steel configuration. Comparisons of the Bonner ball counting rates and gamma-ray spectra behind each configuration also show similar agreement. A sensitivity analysis of the iron configuration indicated that the gamma-ray heating was most sensitive to neutron transport through the spectral modifier and blanket that immediately precede the iron, and that the gamma-ray heating for each detector was extremely localized, arising from gamma rays produced within about 5 cm of the detector.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

4.7 MEASUREMENT AND CALCULATION OF SECONDARY GAMMA RAYS RESULTING FROM EXPOSURE OF Fe, Pb, AND H_2O TO THE ARERR-1 SPECTRUM*

A. S. Makarious† W. E. Ford, III‡ K. R. Turnbull‡

(Abstract of ORNL/TM-5678, in press)

Integral experiments were performed to measure the angular distribution of secondary gamma rays produced when various thicknesses of Fe, Pb, and H_2O samples were exposed to bare and to B_4C-filtered neutron beams from the Research Reactor of Egypt. For selected experiments, multigroup coupled neutron-gamma cross sections and a discrete ordinates transport theory code (DOT4P1-M) were used to calculate the secondary gamma rays and the transport of primary gamma rays. Integral comparisons between the calculated and measured spectra were favorable.

Graphical comparisons of the measured flux for various angles of incidence of the neutron beams on the samples, for various angles of exit on the transmitted side of the samples, and for various sample thicknesses are shown. The comparisons show that the angular distribution of secondary gamma rays for the three materials changes slightly with a change in the angle of beam incident on the sample but increasing the angle between the normal to the sample and the detector by 60° decreases the measured secondary gamma-ray flux up to a factor of two.

An investigation was made to determine the consequences of using single scatter Compton theory versus using discrete ordinates transport calculations to estimate the primary gamma-ray contribution to the measured photon spectra.

*ORNL participation sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

†Atomic Energy Establishment of Egypt.

‡Computer Sciences Division.

4.8 SECONDARY GAMMA RAY PRODUCTION IN IRON AND NATURAL THORIUM FROM CALIFORNIUM FISSION SPECTRUM NEUTRONS*

A. S. Makarious†

(Abstract of ORNL/TM-5675, March, 1977)

Measurement of the secondary gamma-ray spectra from the interaction of a ^{252}Cf fission neutron spectrum with iron samples of different thicknesses and a ^{232}Th sample have been performed at the Tower Shielding Facility at ORNL. A 5-in. by 5-in.-diam NaI (Tl) scintillation spectrometer was used. The measured or 0.846-MeV gamma ray from neutron inelastic scattering in iron was compared with calculations using the differential

values of cross sections as a function of incident neutron energy. This is an integral check for those differential cross sections. Thorium, as far as we know, had not been investigated before for this fast-neutron energy range. Even though a peak at 0.184 MeV was found, it was felt that the resulting large cross-section value obtained made it highly improbable that the gamma ray came from a fast-neutron interaction. Search for other possible neutron-induced gamma rays was very difficult due to the high intensity of naturally radioactive gamma rays from thorium itself.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

†Atomic Energy Establishment of Egypt.

4.9 NEUTRON TOTAL CROSS SECTION CHECKS FOR IRON, CHROMIUM, NICKEL, STAINLESS STEEL, SODIUM AND CARBON*

R. E. Maerker   C. E. Clifford   F. J. Muckenthaler

(Abstract of ORNL-5013, April, 1976)

Utilizing detectors in good geometry at the Tower Shielding Facility behind various thicknesses of iron, chromium, and nickel provided experimental data for checking neutron total cross sections of these elements. In addition, data behind a type 304L stainless steel and carbon were also obtained, as were some data behind sodium. These data were then compared with data from the total cross-section files of both versions III and IV of ENDF/B. Results of these comparisons indicate version IV to be generally superior to version III, but further improvement in version IV is possible for all elements tested. In particular, minima in the total cross sections for both chromium and nickel are apparently still poorly represented in version IV for all energies above 10 keV.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

4.10 DETERMINATION OF THE NEUTRON ENERGY AND SPATIAL DISTRIBUTIONS OF THE NEUTRON BEAM FROM THE TSR-II IN THE LARGE BEAM SHIELD*

C. E. Clifford   F. J. Muckenthaler

(Abstract of ORNL/TM-5225, January, 1976)

The TSR-II reactor of the ORNL Tower Shielding Facility has recently been relocated within a new, fixed shield. A principal feature of the new shield is a beam port of considerably larger area than that of its predecessor. The usable neutron flux has thereby been increased by a factor of ∿200.

The bare beam neutron spectrum behind the new shield has been experimentally determined over the energy range from 0.8 to 16 MeV. A high level of fission-product gamma-ray background prevented measurement of bare beam spectra below 0.8 MeV; however, neutron spectra in the energy range from 8 keV to 1.4 MeV were obtained for two simple, calculable shielding configurations. Also measured in the present work were weighted integral flux distributions and fast-neutron dose rates. Results are presented in detailed tabular form and, where feasible, in the form of graphs.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

4.11 MEASUREMENTS AND CALCULATIONS OF NEUTRON FLUXES THROUGH A SIMULATION OF THE CRBR UPPER AXIAL SHIELDING*

R. E. Maerker F. J. Muckenthaler

(Reprint of Trans. Am. Nucl. Soc. 23, 611, 1976; also summary of paper to be presented at Fifth International Conference on Reactor Shielding, April 18-22, 1977, Knoxville, Tennessee)

Measurements, using a 4-in. Bonner ball, have been made of the neutron fluxes penetrating a simulation of CRBR upper axial shielding at the Tower Shielding Facility. This shielding is one of the determining factors in the biological dose to personnel on the top deck. The simulation consisted of a 45.7-cm-thick slab of SS-304 followed by a series of sodium tanks having a total thickness of 457 cm followed by slabs of carbon steel up to 61.0 cm thick. Measurements were made behind the stainless steel, behind intermediate thicknesses of 152 cm, 305 cm, and 457 cm of sodium (with the stainless steel in place), and behind various thicknesses of the carbon steel following both 305 cm and 457 cm of sodium (also with the stainless steel in place).

Since the attenuation of the full configuration is of the order of 10^{12}, the source beam from the TSR-II had to be intensified over that from an earlier arrangement.[1] By replacing the old collimator arrangement with a shorter one and opening up the collimator to the full diameter of the core (~86 cm), the intensity was increased by a factor of 200. The maximum operating power level capability was also increased from 100 kW to 1 MW. Thus, maximum surface integrated fluxes were enhanced by about a factor of 2000 over those using the old collimator.

The neutron source emerging from the new large beam shield can be represented as a disc source. The energy, angular, and spatial distribution of this source was determined by normalizing ANISN calculations of the pressure vessel leakage to free-field measurements using the large beam shield.[2,3] The adjusted disc source reproduced almost all of the free-field measurements to within ±10%.

Calculations of the fluxes[4] through each configuration of the upper axial shield using the disc source were performed using DOT.[5] A 51-group P_3 representation of ENDF/IV cross sections was employed. A forward biased 100-angle quadrature set was used in lieu of a first-collision source procedure. The geometry of each calculation included the collimator and also the shielding around the sides of the configuration. A comparison of the calculated and measured results is shown in Table 4.11.1.

From Table 4.11.1 it appears that, as anticipated by preanalysis, the 51-group calculations underpredict the Bonner ball reaction rates for neutrons transmitted through the larger thicknesses of iron following the 457 cm of sodium. For these thicknesses, sensitivity analysis has shown[6] the transport to be governed by three processes — neutron transmission above 3 MeV through the stainless steel, neutron transmission through the 300-keV window in sodium, and neutron transmission through the 24-keV window in iron. The 51-group set is known to be deficient in the latter two regions because of the broad-group structure, and from comparisons with NE-213 measurements made behind the stainless steel, it is apparently deficient in the first region as well.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

1. R. E. Maerker and F. J. Muckenthaler, "The Absolute Neutron Spectrum Emerging Through a 15-1/4 in. Collimator from the TSR-II Reactor at the Tower Shielding Facility," ORNL-TM-4010 (1972).

2. C. E. Clifford and F. J. Muckenthaler, "A Determination of the Neutron Energy and Spatial Distributions of the Neutron Beam from the TSR-II in the Large Beam Shield," ORNL-TM-5225 (1976).

Table 4.11.1 Comparison of calculated and measured 4-in. Bonner ball counting rates

Configuration[a]	Counts/sec/watt Calculated	Counts/sec/watt Measured	Calc./Meas.
45.7 cm SS	2.25(2)[b]	2.73(2)	0.83
45.7 cm SS + 152 cm Na	1.14(2)	1.57(2)	0.73
45.7 cm SS + 305 cm Na	1.70(0)	2.70(0)	0.63
45.7 cm SS + 305 cm Na + 15.2 cm Fe	2.87(-2)	3.43(-2)	0.84
45.7 cm SS + 305 cm Na + 30.5 cm Fe	7.35(-4)	8.08(-4)	0.91
45.7 cm SS + 305 cm Na + 45.7 cm Fe	1.37(-4)	1.90(-4)	0.72
45.7 cm SS + 305 cm Na + 61.0 cm Fe	5.42(-5)	8.58(-5)	0.63
45.7 cm SS + 457 cm Na	1.39(-2)	2.03(-2)	0.69
45.7 cm SS + 457 cm Na + 15.2 cm Fe	1.34(-4)	1.14(-4)	1.17
45.7 cm SS + 457 cm Na + 30.5 cm Fe	1.49(-6)	9.78(-7)	1.53
45.7 cm SS + 457 cm Na + 40.6 cm Fe	1.54(-7)	4.73(-7)	0.33
45.7 cm SS + 457 cm Na + 50.8 cm Fe	4.85(-8)	3.08(-7)[c]	0.16
45.7 cm SS + 457 cm Na + 61.0 cm Fe	2.67(-8)	7.93(-8)[c]	0.34

[a]All configurations except the 45.7 cm SS were backed by a 12.7-cm void followed by a 30.5-cm-thick slab of lithium hydride when the measurements were performed. The detector was placed in the center of the void. For the 45.7 cm SS configuration, the measurements (including other detectors not mentioned here) were made in air ∿168 cm behind the slab.

[b]Read: 2.25×10^2.

[c]One of these measurements is probably in error. Judging from the slopes of the two curves behind 305 cm and 457 cm of sodium, one might guess the measurement behind 61.0 cm of iron is about a factor of 2 too low. Many of these measurements will be repeated at a latter time.

3. R. E. Maerker and F. J. Muckenthaler, "The Absolute Neutron Spectrum Emerging Through the Large Beam Collimator from the TSR-II Reactor at the Tower Shielding Facility," ORNL-TM-5183 (1976).

4. Calculated fluxes were converted to 4-in. Bonner ball counting rates using the response function appearing in R. E. Maerker, L. R. Williams, F. R. Mynatt, and N. M. Greene, "Response Functions for Bonner Ball Neutron Detectors," ORNL-TM-3451 (1971).

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

6. F. R. Mynatt, private communication.

4.12 COMPARISONS OF ENDF-III AND ENDF-IV DATA WITH EXPERIMENTS*

R. E. Maerker

(Introduction of paper included in BNL-NCS-21118, ENDF-230, Vol. 1, March, 1976)

The benchmarks used to test ENDF-IV data, as well as earlier ENDF-III data, are the following: SDT1-5, the "broomstick" series which test the neutron total cross section in the range 1 - 10 MeV for iron, oxygen, nitrogen, sodium, and stainless steel; SDT6, which tests the gamma-ray-production cross sections arising from thermal-neutron capture in iron, stainless steel, nitrogen, and sodium; SDT7, which tests the gamma-ray production cross sections averaged over a known fast-neutron spectrum in the range 1 - 15 MeV for iron, stainless steel, oxygen, and sodium; SDT10, which tests the neutron cross sections between 2 and 14 MeV by measuring time-of-flight spectra through several mean free paths of a large number of materials; and SDT12, which tests the neutron cross sections between 1 eV and 15 MeV for sodium by measuring neutron spectral fluxes and weighted integrals of these fluxes behind various thicknesses of sodium up to and including 15 ft. These benchmarks are described in ENDF 166, 167, 168, 169, and 170 for SDT1-5 respectively, in ENDF 176 and 177 for SDT6-7, in UCID 16372 for SDT10, and in ENDF 189 for SDT12. The participants in this data testing include BNL and ORNL for SDT1-5, LASL and ORNL for SDT6-7, LLL for SDT10, and ORNL for SDT12.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

4.13 SB2. EXPERIMENT ON SECONDARY GAMMA-RAY PRODUCTION CROSS SECTIONS ARISING FROM THERMAL-NEUTRON CAPTURE IN EACH OF 14 DIFFERENT ELEMENTS PLUS A STAINLESS STEEL*

R. E. Maerker

(Abstract of ORNL/TM-5203, ENDF 227, January, 1976)

The experimental and calculational details for a CSEWG integral data-testing shielding experiment are presented. This particular experiment measured the secondary gamma-ray-production cross sections arising from thermal-neutron capture in iron, nitrogen, sodium, aluminum, copper, titanium, calcium, potassium, chlorine, silicon, nickel, zinc, barium, sulfur and a type 321 stainless steel.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

4.14 SB3. EXPERIMENT ON SECONDARY GAMMA-RAY PRODUCTION CROSS SECTIONS AVERAGED OVER A FAST-NEUTRON SPECTRUM FOR EACH OF 13 DIFFERENT ELEMENTS PLUS A STAINLESS STEEL*

R. E. Maerker

(Abstract of ORNL/TM-5204, ENDF 228, January 1976)

The experimental and calculational details for a CSEWG integral data-testing shielding experiment are presented. This particular experiment measured the secondary gamma-ray-production cross sections averaged over a fast-neutron spectrum for iron, oxygen, sodium, aluminum, copper, titanium, calcium, potassium, silicon, nickel, zinc, barium, sulfur, and a type 321 stainless steel. The gamma-ray production cross sections were binned into $\sim$0.5-MeV wide gamma-ray energy intervals.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

4.15 ANALYSIS OF A FUEL-PIN NEUTRON-STREAMING EXPERIMENT TO TEST METHODS FOR CALCULATING NEUTRON DAMAGE TO THE GCFR GRID PLATE*

C. O. Slater M. B. Emmett†

(Summary of paper to be presented at Fifth International Conference on Reactor Shielding, April 18-22, 1977, Knoxville, Tennessee; early part of analysis summarized by C. O. Slater and D. E. Bartine in paper presented at Annual Meeting of American Nuclear Society, June 13-18, 1976, Toronto, Ontario, Canada, and in GCR-76/37, November, 1976)

In shielding analyses for General Atomic's Gas Cooled Fast Reactor,[1] a primary concern is the damage to the grid plate caused by neutrons streaming through the long, narrow, low-density coolant passages between the reactor's fuel pins. This type of problem severely challenges presently developed transport methods since an accurate three-dimensional description of the geometry is possible only with the Monte Carlo method and even then a detailed description, plus the time requirements, would probably exceed current computer capabilities.

To provide an experimental base for testing less-detailed streaming calculations, an experiment designed by GA and ORNL was performed at the ORNL Tower Shielding Facility for a fuel-pin lattice having a void fraction comparable to that of the GCFR. The experimental array, positioned in a reactor beam modified to simulate a fast-reactor neutron spectrum, consisted of 894 1.3%-enriched UO_2 fuel pins (11.5-mm-diam by 1371.6-mm-long) arranged on a triangular pitch inside a 0.459-m-ID steel pipe surrounded by water. The pins were supported by three 2.77-mm-thick steel grid plates. Neutron flux measurements made along an arc 9.144 m beyond the array indicated a strong streaming effect.

Two-dimensional discrete ordinates analyses of the experiment were performed, first with a homogeneous model and second with a heterogeneous model that consisted of annular rings of void and fuel. As expected, the first calculation underpredicted the experiment. The second overpredicted by factors of 2 to 4, indicating that upper limits of streaming can be established by this method. Subsequent work on refining the mesh and source distribution, together with better techniques for obtaining radiation levels at points outside the assembly, have improved the results.

A limited Monte Carlo analysis also overpredicted the experiment for the one detector point calcualted, but less so than the two-dimensional calculation. Further study of the Monte Carlo method indicates that with a simplified geometry and specially developed calculational techniques closer agreement with the experiment can be achieved.

*Research sponsored by ERDA Division of Nuclear Research and Applications.

†Computer Sciences Division.

1. Bruno Pelland, "Physics Design of a Gas Cooled Fast Breeder Reactor Demonstration Plant," GA-10509, General Atomic Company (August 1971).

4.16 ANALYSIS OF NEUTRON SCATTERING AND GAMMA-RAY PRODUCTION INTEGRAL EXPERIMENTS ON NITROGEN FOR NEUTRON ENERGIES FROM 1 TO 15 MeV*

S. N. Cramer E. M. Oblow

(Abstract of ORNL/TM-5220, March, 1976, and of paper to be published in Nuclear Science and Engineering)

Monte Carlo transport calculations were made to analyze the results of two integral measurements of neutron scattering and gamma-ray production from liquid-nitrogen samples. The experimental data from Intelcom Radiation Technology and Oak Ridge National Laboratory were given as angular-dependent NE-213 detector count rates of neutrons and gamma rays scattered from a spherical nitrogen dewar pulsed with a 1- to 20-MeV neutron source. ORNL results also included unfolded neutron and gamma-ray spectra as a function of detector angle in broad incident neutron energy bins. Multigroup Monte Carlo calculations using the MORSE code and ENDF/B-IV nitrogen cross-section data were made to analyze all reported results. Comparisons of calculated and measured results indicate no major deficiencies exist in the ENDF/B-IV gamma-ray production data, in contrast to the conclusions drawn from studies in prior years. Deficiencies, however, were found in the neutron data, primarily in the elastic and inelastic data above 9 MeV and the elastic angular distribution data around 5 MeV.

*Research sponsored by Defense Nuclear Agency.

4.17 ANALYSIS OF A NEUTRON SCATTERING AND GAMMA-RAY PRODUCTION INTEGRAL EXPERIMENT ON OXYGEN FOR NEUTRON ENERGIES FROM 1 TO 15 MeV*

S. N. Cramer E. M. Oblow

(Abstract of ORNL/TM-5535, September, 1976)

Monte Carlo calculations were performed to analyze an integral experiment on a liquid-oxygen sample to determine the adequacy of the neutron-scatteirng and gamma-ray-production data for oxygen. The experimental results included energy- and angular-dependent NE-213 detector count rates and secondary pulse-height spectra for scattered neutrons and gamma rays. The sample was a spherical dewar of liquid oxygen pulsed with a 1- to 20-MeV neutron source. Pulse-height data were unfolded to generate secondary neutron and gamma-ray-production spectra as a function of angle in broad incident-neutron energy bins. Analysis of all the reported data was based on multigroup Monte Carlo calculations using the MORSE code. Results indicate that the current ENDF/B-IV neutron and gamma-ray-production data for oxygen above 1 MeV appear to be in good order. The only major discrepancy uncovered was related to neutron scattering and gamma-ray production from first-level inelastic-scattering interactions. Calculated results for the production of 6-MeV gamma rays from the 6-MeV first inelastic level in oxygen appear to be low by around 50% at energies above the inelastic threshold. Likewise, calculated secondary neutron spectra for incident neutron energies above 6 MeV are uniformly low at energies corresponding to neutrons having had first-level inelastic-scattering events in oxygen. Additional deficiencies in the oxygen cross-section data are indicated for inelastic scattering from the cluster of discrete levels in the 12- to 13-MeV range and for elastic scattering at very small angles at energies above 2 MeV. The size of the inelastic discrepancies are larger than the 20 to 30% order of error indicated for these cross sections in the ENDF/B-IV uncertainty files for oxygen.

*Research sponsored by Defense Nuclear Agency.

4.18 COMPARISON OF MEASUREMENTS AND CALCULATIONS FOR ORNL INTEGRAL NEUTRON SCATTERING EXPERIMENT FOR IRON*

S. N. Cramer E. M. Oblow

(Reprint of Trans. Am. Nucl. Soc. 23, 606, 1976; also summary of ORNL/TM-5548, November, 1976)

Calculations of an integral neutron-scattering experiment on an iron sample have been performed using the ENDF/B-IV data set MAT 1192 (DNA MAT 4192). The neutron source incident on the sample ranged from 20 MeV to 1 MeV. Comparisons between experimental and calculated results are given as neutron count rates and neutron secondary energy spectra.

The experimental sample consisted of six adjacent iron rings, each 0.62 cm thick and of increasing diameter away from the source of the incident neutron beam. The smallest and closest ring to the source was 15.3 cm ID and 20.34 cm OD. The largest ring was

20.34 cm ID and 25.38 cm OD, with the radii of the other rings increasing in steps of 0.504 cm. An NE-213 detector, 4.22 cm x 4.65 cm, was placed in the center of the rings (creating an approximately 90-deg scattering angle) and was shielded by a shadow bar from the incident neutron beam. More details of the experimental setup are given in ref. 1.

The calculations were done using the MORSE multigroup Monte Carlo code. The cross sections were processed by the AMPX code system into a 177-group structure of equal lethargy spacing from 0.3 MeV to 20 MeV using a P_8 expansion. Results of the count rate comparison are shown in Fig. 4.18.1. The count rates have been converted to incident energy dependence from the neutron flight times in the 47-m beam tube. Figure 4.18.2 shows flux spectra comparisons for the 12- to 13-MeV, 13- to 14-MeV, and 14- to 15-MeV incident energy bins. The spread in the calculated values indicates one standard deviation.

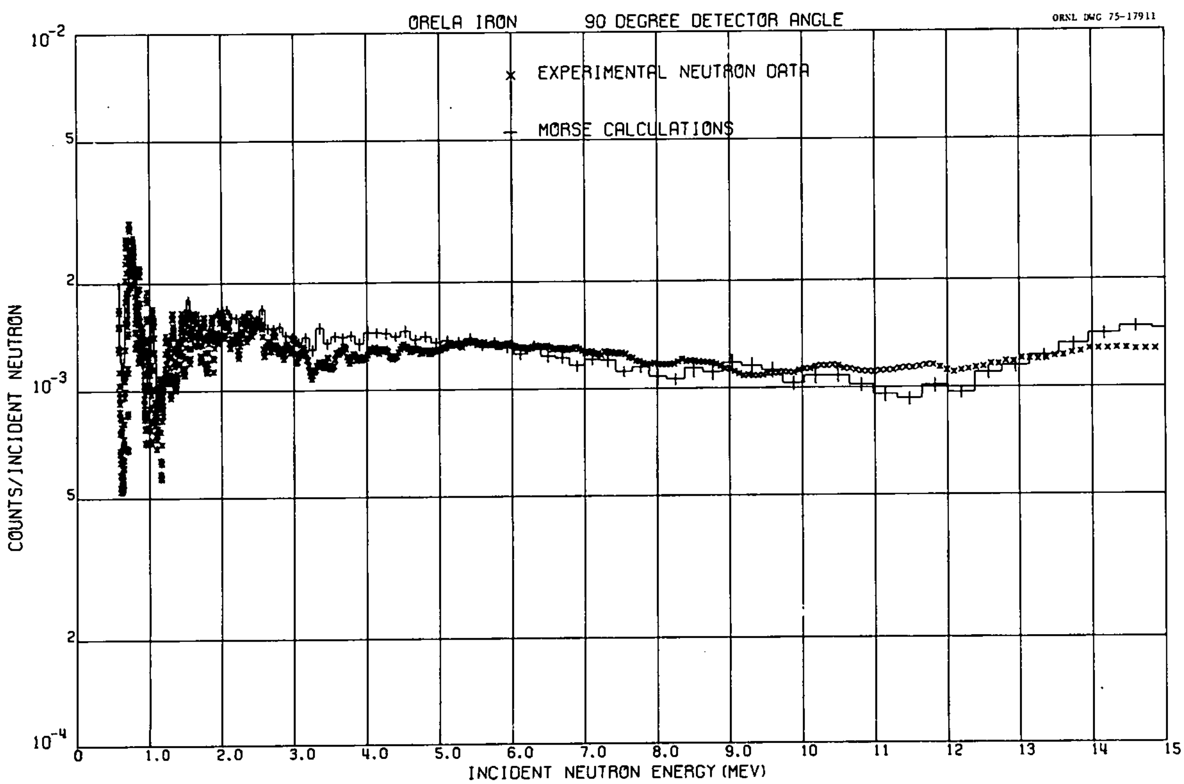

Fig. 4.18.1. Count rate comparison for the ORNL iron ring integral experiment.

ORNL DWG 75-17898

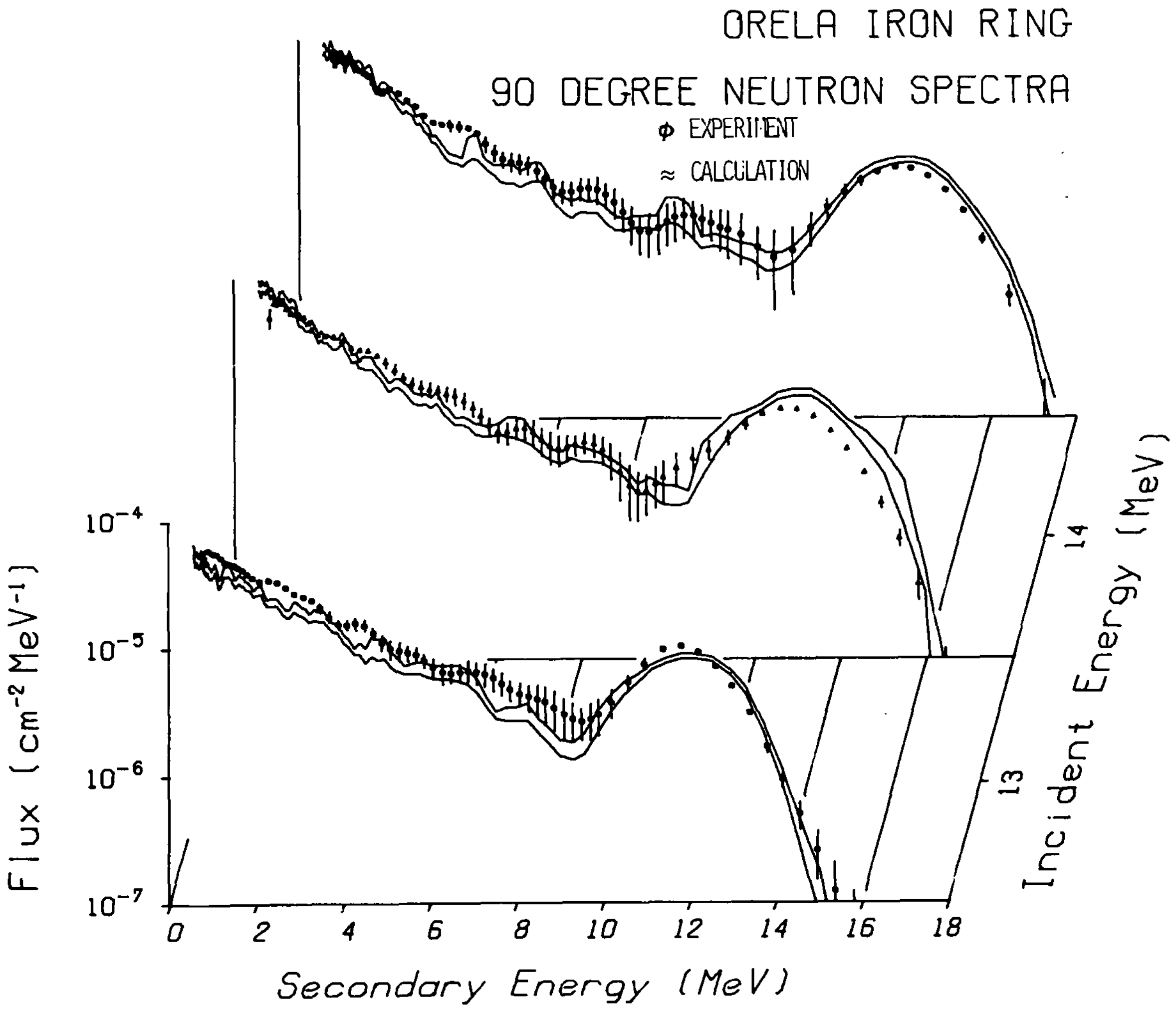

Fig. 4.18.2. Flux spectra comparisons for the ORNL iron ring integral experiment.

It is seen from the figures that there is good general agreement between the calculations and experiment. The lower incident energy spectra comparisons are similar in agreement with those shown. The count-rate calculation in Fig. 4.18.1 differs little from that using previous data evaluations. However, the high incident energy flux spectra calculations, such as shown in Fig. 4.18.2, are in much better agreement with experiment than calculations using previous iron data evaluations (see, for example, ref. 2). The earlier ENDF/B data sets grossly underpredicted the spectra in the valley between the elastic and inelastic scattering peaks for incident energies above 7 MeV. By adjustment of the secondary energy distributions in MAT 1192, these discrepancies seem to have been removed.

*Research sponsored by Defense Nuclear Agency.

1. G. L. Morgan, T. A. Love, and F. G. Perey, "Integral Neutron Scattering Measurements on iron from 1 to 20 MeV," ORNL-TM-4193 (1973).

2. S. N. Cramer, R. W. Roussin, and E. M. Oblow, "Monte Carlo Calculations and Sensitivity Studies of the Time-Dependent Neutron Spectra Measured in the LLL Pulsed Sphere Program," ORNL-TM-4072 (1973).

4.19 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*

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

(Abstract of Nucl. Instr. Methods 128, 125, 1975)

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 a NE-213 scintillation counter. Multiparameter data acquisition is done by an on-line computer for incident neutron energies from 1 to 20 MeV.

*Research sponsored by Defense Nuclear Agency.

4.20 TRANSPORT CALCULATIONS OF NEUTRON WAVE EXPERIMENTS IN SUBCRITICAL ASSEMBLIES

F. C. Difilippo*

(Abstract of paper submitted for journal publication)

A recent neutron wave experiment in a thermal multiplying assembly with and without control rods has been analyzed numerically in terms of transport theory. The code TASK was used for this purpose. The present study dealing with a highly enriched, compact multiplicative system indicates that the dispersion law of the assembly is very sensitive to transport effects and to the estimation of the leakage of the fast-neutron population. The present calculations of neutron wave propagation in multiplicative systems shows that this technique can be used as a highly sophisticated experiment for integral checks of neutron cross section sets.

*Comision Nacional de Energia Atomica.

5. DEVELOPMENT OF METHODS FOR SHIELD AND REACTOR ANALYSES

5.1 SHIELDING METHODS DEVELOPMENT IN THE UNITED STATES*

F. R. Mynatt

(Summary of invited paper to be presented at Fifth International Conference on Reactor Shielding, April 18-22, 1977, Knoxville, Tennessee)

The rate of shielding methods development in the U.S. has slowed in the period since the 1972 meeting in Paris. The current development program consists largely of long-term efforts on the major Monte Carlo and discrete ordinates codes and maintaining operation of existing code versions. Where practical (primarily in cross-section processing and sensitivity analysis), shielding needs have been combined with core physics needs.

Cross-Section Processing is primarily concerned with producing coupled neutron-gamma libraries in general-purpose or specific problem-dependent group structures. A major change in the AMPX system has been the replacement of the XLACS-NITAWL neutron modules based on the Nordheim resonance self-shielding method with the MINX-SPHINX neutron modules based on the Bondarenko resonance self-shielding method. This allows production of a single library suitable for both LMFBR core physics and shielding. LASL is developing the NJOY (MINX+1) system based on the Bondarenko approach.

Deterministic Methods rely primarily on the discrete ordinates codes ANISN and DOT. A major effort has resulted in the DOT IV two-dimensional discrete ordinates code which employs zone-dependent spatial and angular mesh and multiple-hierarchy computer storage to greatly increase problem size capability and reduce computer memory size required. The variable mesh feature allows calculations of streaming problems at much lower cost than that achievable with the existing discrete ordinates codes. LASL finite element codes ONETRAN and TRIPLET are being used more extensively. The TRIDENT code in triangular R-Z geometry may be especially useful for CTR toroidal geometry problems.

Monte Carlo Methods rely primarily on the multigroup code MORSE. The latest version MORSE-SGC employs an updated combinatorial geometry package and super-grouped cross-sections so that computer memory requirements are greatly reduced. Improved albedo features and a library of albedo data are being prepared for MORSE. DOMINO is still being used for DOT-MORSE coupling at a surface. Development of a volume coupling code is needed.

Sensitivity Analysis and Generalized Perturbation Theory methods have been developed to a high degree of competency. The original SWANLAKE capability is now part of the FORSS code system which has all sensitivity analysis capabilities, including uncertainty analysis and cross-section adjustment for one- and two-dimensional geometries using transport theory solutions. FORSS can also be used as a generalized perturbation code for shield design and optimization. Channel-theory analysis examines the flow of particles in space and energy and is very useful for design and evaluation of complex shields.

Progress in many areas seems promising. Nevertheless, much remains to be done, and the task of preparing comprehensive shield design codes or parameterized design guides has not been started.

*ORNL development sponsored by ERDA Division of Reactor Development and Demonstration and Defense Nuclear Agency.

5.2 THE DOT IV VARIABLE MESH DISCRETE ORDINATES TRANSPORT CODE*

W. A. Rhoades

(Summary of paper to be presented at Fifth International Conference on Reactor Shielding, April 18-22, 1977)

DOT IV has been added to the DOT series of discrete ordinates transport codes. The new code features completely reprogrammed logic structured to take advantage of special features of IBM-360/370 and CDC-7600 computers. Special attention has been given to inter-computer adaptability, and to compatability with other codes developed by ERDA-RDD Physics Branch.

Special features include options to allow the first space mesh to vary with the second space dimension, to allow the directional quadrature to vary with either space dimension and with energy, and an efficient blocking scheme to allow solution of problems holding only a portion of the space mesh in memory at a time. The variable mesh features concentrate the computational work in areas of greatest interest, e.g., a streaming gap, while the blocking allows very large problems to be solved as a unit without excessive memory requirements.

Other novel features include an improved formulation of the "weighted-difference" method of flux extrapolation which removes certain difficulties experienced with previous models, a method of removing negative sources generated by the finite cross-section expansion, and a "cylindrical" boundary condition intended for use in cylindrical cell problems.

On IBM computers, small assembler-language packages more than double computing speeds. Special disk-manipulation packages allow the blocked solution to proceed without undue I/O delays. On CDC equipment, the slow memory is used to hold blocks of flux and source data, which are then moved in strings to a fast-memory working area.

On problems which can also be solved by DOT III, the DOT IV central processor (CPU) speed is quite comparable to its predecessor. The code can often capture 60% of the CPU time running in competition with other jobs on the IBM-360/195, even with its space problem broken into several blocks.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

5.3 COMMENTS ON THE DOT 3.5 CODE*

W. A. Rhoades

(Reprint of paper published in SECU Bulletin N.2, Program Computer Library Newsletter 20, 9, 1976)

A new release of the DOT III code, now available from the Radiation Shielding Information Center, differs from the previous versions in such important ways that all previous versions should be replaced as soon as practical. The new version is called DOT 3.5. It was developed for these purposes:

(1) To provide a test vehicle for several new ideas to be used in DOT IV,

(2) To provide a code which would facilitate checking between DOT III and DOT IV, and

(3) To make new developments available to the user on an interim basis until DOT IV is ready for general use.

Surely, the most significant change is a marked improvement in convergence on many deep-penetration problems. As an example, a test problem which failed to converge in the zones of interest in 30 iterations on DOT III. On DOT 3.5, pointwise converged to 0.01 in 8 iterations using DOT 3.5. New features in the control input section allow some flexibility in the rebalance method. A type of damping which adds an appropriate amount to cell boundary flows to smooth the results is used.

The control parameters are input using the FIDO format, and have been reordered to agree with GRTUNCL and DOT IV. The space allocation is made at execution time, so that the code adapts itself to fill the region size available to it on IBM equipment.

In the 19$ array, a "0" entry selects the macroscopic material for each zone. Input and output data sets are handled much as before, except that the cross-section input unit is now read in. Buffer space for all data sets except card input and printed output is specified by the parameter NBUF.

The weighted difference model in this version, developed by W. W. Engle, Jr., has been thoroughly tested. At Oak Ridge, we use it for <u>all</u> deep-penetration problems. Its slightly slower computation speed is more than compensated for by the improved convergence and accuracy of results. Some users report better results using the linear model for eigenvalue calculations, however.

Fortran substitutes for all assembly-language routines are supplied. CDC users should find that the methods used to adapt previous versions of DOT III will be successful on this version also. Adaptation to UNIVAC should be trivial.

An update to ORNL/TM-4280 describing this revised version is planned. In the meantime, examining the printout from a sample problem, together with notes supplied with the code, is sufficient to indicate how to use the code.

[*] Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

5.4 THE FBSAM DATA TRANSMISSION PACKAGE FOR IBM 360/370 COMPUTERS[*]

W. A. Rhoades

(Abstract of ORNL/TM-5199, January, 1976)

The FBSAM subroutine package provides rapid movement of large blocks of data directly between user storage arrays and magnetic disk data sets. No buffer space is required. Large reductions in data transmission time and number of I/O requests can be obtained. Moves can be concurrent with each other and with computation. A form of random access is available. A demonstration program provides a thorough test of all features. The package is intended for use by FORTRAN programmers with little or no assembler language experience. It should be operable on most IBM-360/370 machines. A sample application to an I/O-bound code is discussed. The use of three concurrent functions entirely eliminated the real time associated with data transmission tasks by allowing them to proceed concurrently with other tasks.

[*] Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

5.5 USERS GUIDE TO MORSE-SGC[*]

S. K. Fraley[†]

(Abstract of ORNL/CSD-7, March, 1976)

The MORSE-SGC (<u>S</u>uper <u>G</u>rouped <u>C</u>ross Section) code is a version of the MORSE code which ...es the core size requirements to solve problems with large cross-section storag ...ments. This version of the MORSE code is available with both an improved combina geometry package and a KENO geometry package. Other changes have also been impleme .n the code to make problem formulation more convenient.

[*] Research sponsored by Defense Nuclear Agency.

[†] Computer Sciences Division.

5.6 A USER'S GUIDE FOR LAVA AS ISSUED WITH MORSE-SGC*

S. K. Fraley†

(Abstract of ORNL/CSD-11, June, 1976)

LAVA (Let ANISN Visit AMPX) is a computer code which will transform ANISN cross-section libraries to the AMPX working library format. This version of LAVA is a stand-alone program issued with the MORSE-SGC code which accepts only cross sections in the AMPX working library format.

*Research sponsored by Defense Nuclear Agency.

†Computer Sciences Division.

5.7 THE ROLES OF THE EVENT VALUE AND THE POINT VALUE IN MONTE CARLO IMPORTANCE SAMPLING*

J. S. Tang† T. J. Hoffman† P. N. Stevens‡

(Summary of paper to be presented at Annual Meeting of American Nuclear Society, June 12-17, 1977, New York City, and at Fifth International Conference on Reactor Shielding, April 18-22, 1977, Knoxville, Tennessee; also summary of paper to be published in April issue of Nuclear Science and Engineering and of report ORNL/TM-5414, June, 1976)

To obtain a Monte Carlo solution of a deep-penetration radiation transport problem, importance sampling is required. The adjoint function has been shown to be a good importance function.[1-3] However, to fully utilize the adjoint information, a distinction must be made between two adjoint functions — the event-value function and the point-value function. In this paper, the proper use of these functions in Monte Carlo importance sampling is discussed and illustrated.

The distribution of radiation particles can be described in terms of the density of particles entering events (collision density) or in terms of the density of particles leaving events (emergent particle density). Likewise the importance, i.e., the expected contribution of a particle, can be expressed as the importance of a particle entering an event (event-value function) or as the importance of a particle leaving an event (point-value function).

By requiring that the importance of a particle be conserved along its flight path, it can be shown[4,5] that the adjoint flux, i.e., the solution of the integrodifferential form of the adjoint Boltzmann equation, is the importance of a particle leaving an event. The adjoint flux, $\phi^*(\bar{r},\bar{v})$, is the dependent variable that can be obtained with most deterministic computer codes when adjoint calculations are performed, e.g., ANISN[6] and DOT[7].

The event-value function, $W(\bar{r},\bar{v})$, can be obtained from the point-value function, $\phi^*(\bar{r},\bar{v})$, by noting that the importance of a particle entering an event is the sum of the contribution of the particle at the event site and the expected contribution of event survivals, i.e.,

$$W(\bar{r},\bar{v}) = P(\bar{r},\bar{v}) + \int d\bar{v}' \frac{\Sigma_s(\bar{r},\bar{v}\to\bar{v}')}{\Sigma_t(\bar{r},\bar{v})} \phi^*(\bar{r},\bar{v}') \,, \tag{1}$$

where Σ_s and Σ_t are the differential scattering and total cross sections, respectively, and P is the collision density response function.

The random-walk procedure used in Monte Carlo calculations consists of alternately selecting the velocity of a particle emerging from an event (collision process) and selecting the position of the next event (transport process). Since in the collision process one selects parameters corresponding to the emergent particles, the point-value function should be used to bias this selection. Since in the transport process, one selects the collision sites, the event-value function is the appropriate biasing function. That is, the adjoint flux is the appropriate biasing function for the selection of the emergent particle velocity, but not the proper function for biasing the transport process. The transport process should be biased with the event-value function obtained from the adjoint flux with Eq. (1).

An indication of the variance reduction associated with the use of the event-value function to bias the transport process was obtained in the analysis of the problem shown in Fig. 5.7.1 Adjoint fluxes were obtained for the adjoint source with the discrete ordinates code DOT. A Monte Carlo calculation was performed for detector 1 with the MORSE[8] computer code in which the adjoint flux was used to bias the transport process; the same calculation was then repeated with event-value biasing. The variance was reduced by a factor of seven when the event-value function was used. These calculations illustrate the necessity of the proper use of the adjoint information in performing Monte Carlo calculations with importance biasing.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

†Computer Sciences Division.

‡The University of Tennessee.

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5.8 SPATIAL CHANNEL THEORY — A TECHNIQUE FOR DETERMINING THE DIRECTIONAL FLOW OF RADIATION THROUGH REACTOR SYSTEMS*

M. L. Williams W. W. Engle, Jr.

(Summary of paper to be presented at Fifth International Conference on Reactor Shielding, April 18-22, 1977, Knoxville, Tennessee; also summary of ORNL/TM-5467, July, 1976, and of Nucl. Sci. Eng. 62, 92, 1977)

A general description of the nature of problems encountered in reactor shield design is surprisingly unified. Simply stated, the goal of a shielding calculation is to determine the output response at some point within the reactor system due to an input source.

ORNL-DWG 76-10657

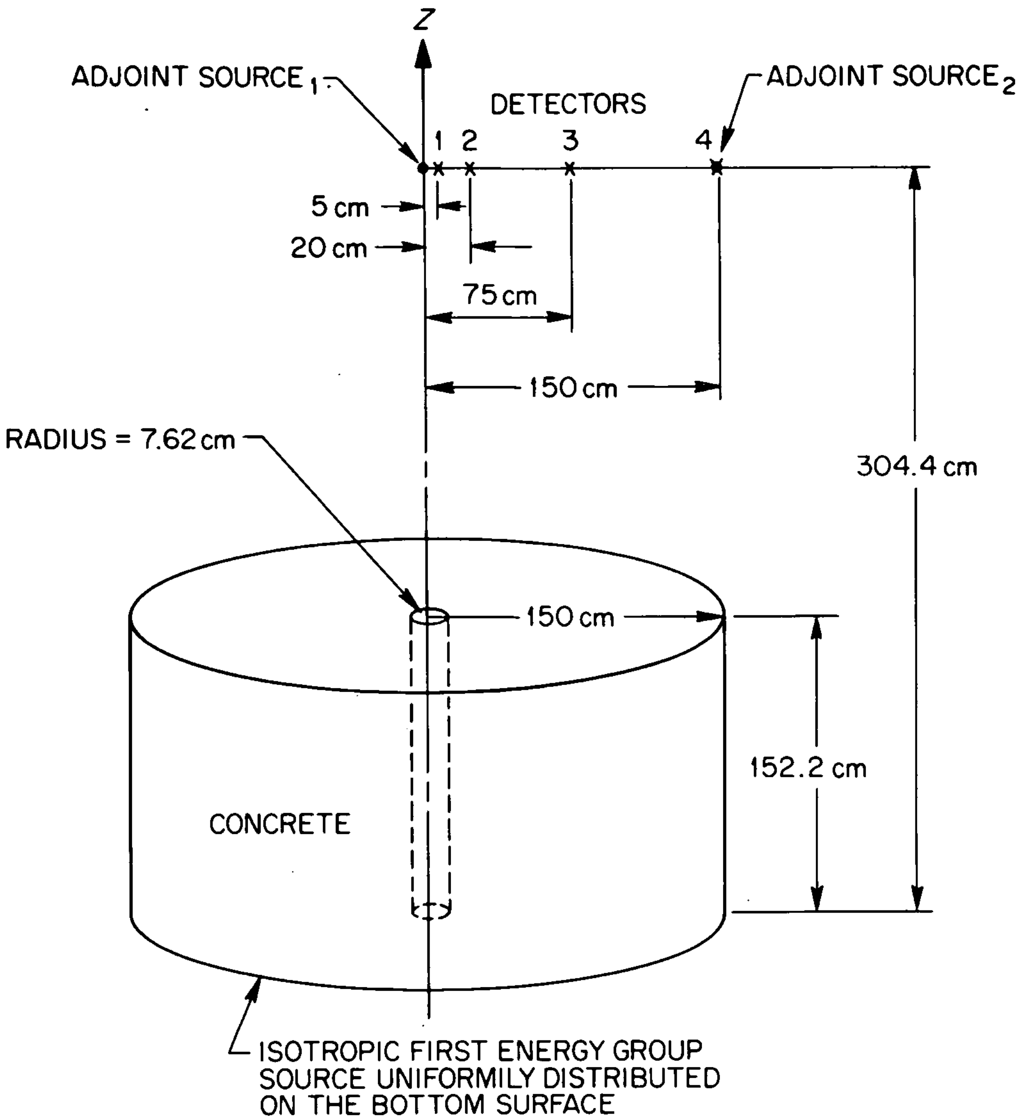

Fig. 5.7.1. Geometry of concrete cylinder with axial duct, source, detectors, and adjoint source.

The response may be any physically observable quantity such as a biological effect or an electrical impulse registered on a meter, but in every case it is merely a manifestation of some portion of the imput source that has been transformed into a phenomenon which is perceptible to the senses. Lying between the source and the response regions is an area of extreme geometrical complexity, composed of structure, pipes, control systems, and shielding. An understanding of how particles emitted from the source traverse this geometrical maze and are finally converted into an observable response is essential in ascertaining effective shield location.

For this reason, a technique known as spatial channel theory has been developed to reveal the important "channels" through which particles flow to contribute to a response. Spatial channel theory is based on two observations. First, for a given response, only a portion of the particles emitted from the source will end their histories at the response-detector, and these particles are the only ones of interest for that particular response. The others are extraneous and only tend to mask the behavior of the contributing particles, which may be a small fraction of the total population. Second, even among the contributing particles, certain ones will produce a larger physical effect than others upon the detector. Therefore, it is desirable to associate a "partial response" to each contributing particle. This can be done by applying the "importance" property of the adjoint flux to the neutron population, and, when that transformation is performed, the resulting pseudo-particle is called a *contributon,* having units of "response contribution." The flux arising from this particle is called the *response-flux,* or the *contributon flux,* and it has units of response/cm^2-sec.

Contributons have several interesting characteristics. Because they have units of partial response, a plot of the contributon flux reveals the channels followed by the response as it flows through the reactor. Also, since contributons are related to contributing particles, they can never be lost. All contributons emitted from the input source must pass through any closed surface around the source as they stream toward the response location. Furthermore, the leakage per second of contributons through the surface must equal the total response.

This technique appears to have great potential in two-dimensional analysis of complex systems, since it furnishes insight into the radiation transport process which is not available from a forward or adjoint calculation alone.

Channel-theory calculations have already been applied successfully in FFTF, CRBR, and GCFR shielding studies and have become a standard part of the ORNL shielding program.

*Research sponsored by ERDA Division of Reactor Development and Demonstration.

5.9 A LINEAR TRIANGLE FINITE ELEMENT FORMULATION FOR MULTIGROUP NEUTRON TRANSPORT ANALYSIS WITH ANISOTROPIC SCATTERING*

R. A. Lillie J. C. Robinson†

(Abstract of ORNL/TM-5281, March, 1976)

The discrete ordinates method is the most powerful and generally used deterministic method to obtain approximate solutions of the Boltzmann transport equation. However, as presently formulated, it is both restricted to orthogonal geometries and susceptible to producing ray effects.

In this work, a finite element formulation, utilizing a canonical form of the transport equation, is developed to obtain both integral and pointwise solutions to neutron transport problems. To facilitate its application to nonorthogonal planar geometries, the formulation is based on the use of linear triangles. A general treatment of anisotropic scattering is included in the formulation by employing discrete ordinates like approximations. In addition, multigroup source outer iteration techniques are employed to perform group dependent calculations.

The ability of the formulation to substantially reduce ray effects and its ability to perform streaming calculations are demonstrated by analyzing a series of test problems. The anisotropic scattering and multigroup treatments used in the development of the formulation are verified by a number of one-dimensional comparisons. These comparisons also demonstrate the relative accuracy of the formulation in predicting integral parameters.

The applicability of the formulation to nonorthogonal planar geometries is demonstrated by analyzing a hexagonal type lattice. A small high leakage reactor model is analyzed to investigate the effects of varying both the spatial mesh and order of angular quadrature. This analysis reveals that these effects are more pronounced in the present formulation than in other conventional formulations. However, the insignificance of these

effects is demonstrated by analyzing a realistic reactor configuration. In addition, this final analysis illustrates the importance of incorporating anisotropic scattering into the finite element formulation.

*Research sponsored by ERDA Division of Reactor Development and Demonstration and The University of Tennessee.

†The University of Tennessee.

5.10 A DISCRETE ANGLE — FINITE ELEMENT FORMULATION*

R. A. Lillie J. C. Robinson†

(Summary of paper to be presented at Annual Meeting of American Nuclear Society, June 12-17, 1977, New York City)

The discrete ordinates method as generally formulated is susceptible to producing flux distortions in weakly scattering media. These distortions, commonly called "ray effects," are due to the discretization of the direction variable in the divergence term of the transport equation. Ray effects are not encountered with deterministic methods that do not consider the angular domain nor with those which treat the angular variable as a continuous variable. However, these methods are most often not adequate in obtaining realistic solutions to shielding problems where streaming is of primary concern.

The purpose of this paper is to describe a multigroup finite element formulation which partially mitigates ray effects and which appears cabable of performing streaming calculations. The formulation, which is similar to the cubature approach of Kaper et al.,[1] is based upon the application of a discrete-ordinates-like approximation to the mono-energetic second-order canonical form of the transport equation. Linear Lagrange polynomials are used to represent the spatial dependence of the even and odd parity fluxes over a general triangular mesh grid.

The ability of the present formulation to substantially reduce ray effects is demonstrated for a simple one-group problem by the flux comparisons in Fig. 5.10.1. The discrete angle-finite element (DAFE) and the discrete ordinates (DOT)[2] calculations are denoted by E_N and S_N respectively. The smoothness of the E_4 curve illustrates the absence of ray effects, whereas anomalous humps or flux distortions are present in the S_4 and S_8 discrete ordinates results.

In Fig. 5.10.2, results are presented for a problem in which a flat source region is adjacent to a region containing a void. Although the E_4 and S_4 curves are comparable, the S_8 and S_{12} results indicate that both curves underestimate the flux peak. In additional calculations, a simple clustering of the available E_4 and S_4 directions in a narrow band about the X axis resulted in overestimates of the flux peak. For these cases, the discrete angle — finite element formulation appears capable of providing a viable means of reducing ray effects and predicting flux peaks in streaming gaps.

*Research sponsored by ERDA Division of Reactor Development and Demonstration and The University of Tennessee.

†The University of Tennessee.

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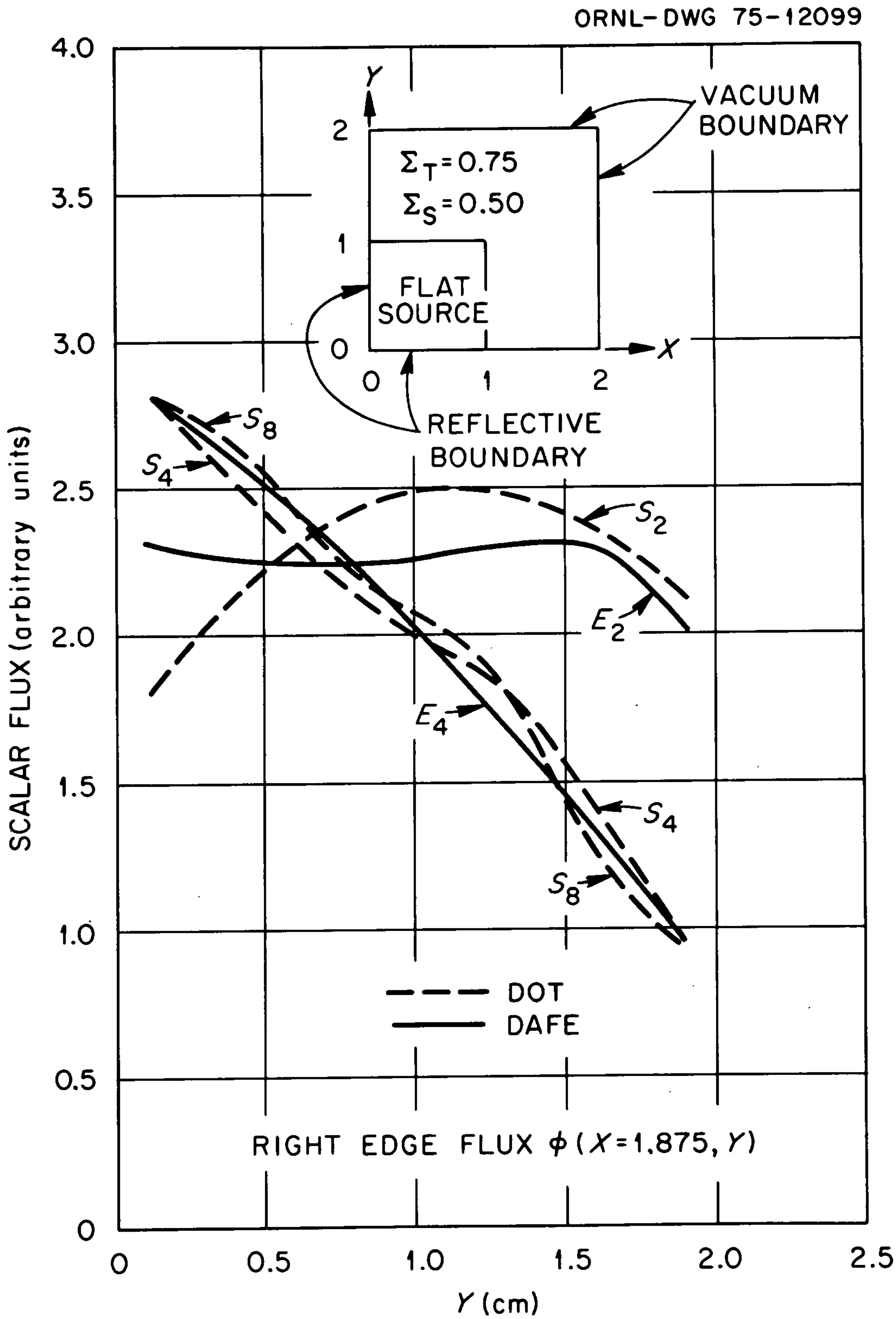

Fig. 5.10.1. Comparison of right edge (x = 1.875 cm) scalar flux in a weak scattering material.

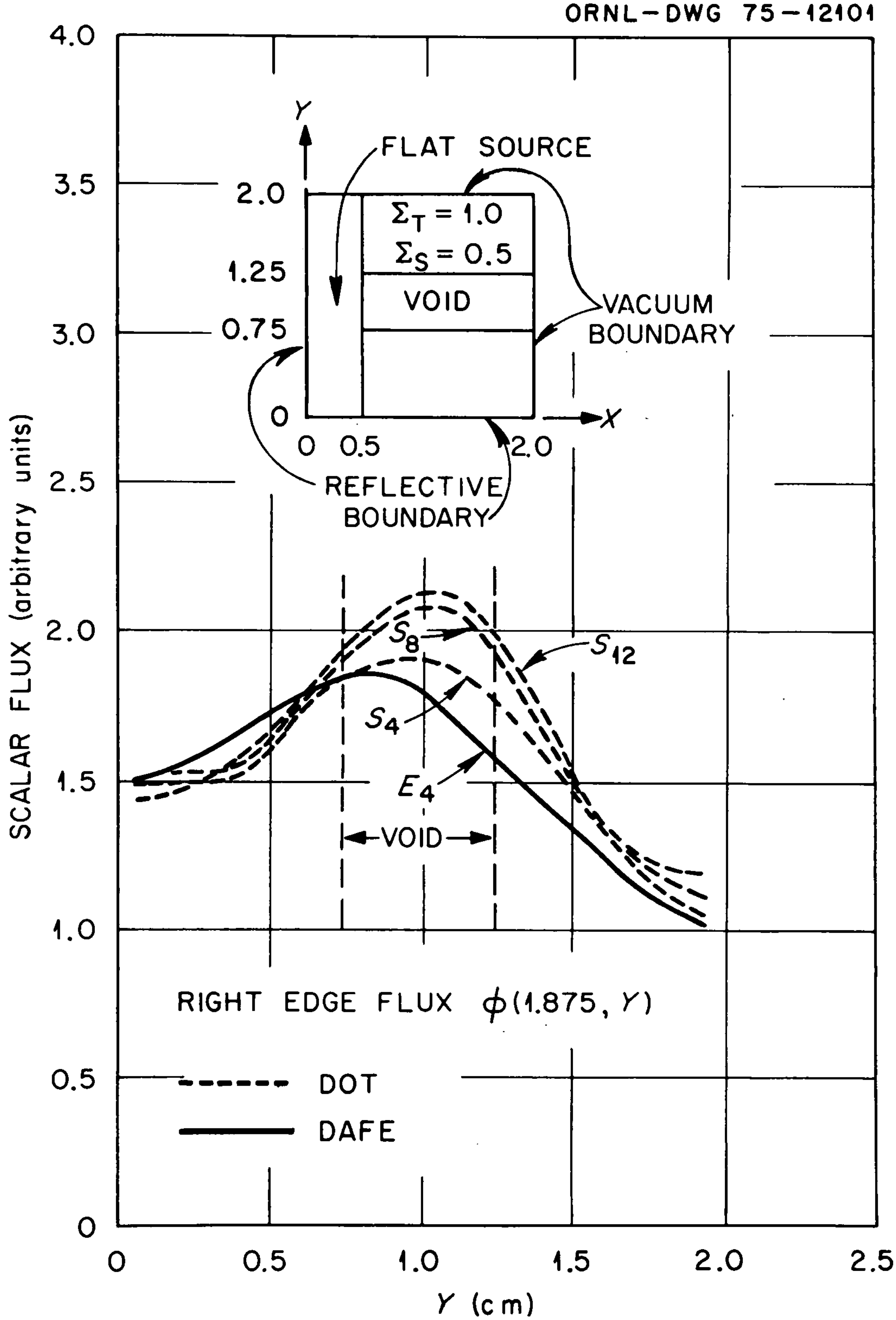

Fig. 5.10.2. Comparison of right edge (x = 1.875 cm) scalar flux in the presence of a void.

5.11 A COMPUTATION SYSTEM FOR NUCLEAR REACTOR CORE ANALYSIS*

D. R. Vondy T. B. Fowler G. W. Cunningham† L. M. Petrie†

(Abstract of ORNL-5158, in press)

This report documents a system which contains computer codes as modules developed to evaluate nuclear reactor core performance. The diffusion theory approximation to neutron transport may be applied with the VENTURE code treating up to three dimensions. The effect of exposure may be determined with the BURNER code, allowing depletion calculations to be made. The features and requirements of the system are discussed and aspects common to the computational modules, but the latter are documented elsewhere. User input data requirements, data file management, control, and the modules which perform general functions are described. Continuing development and implementation effort is enhancing the analysis capability available locally and to other installations from remote terminals.

*Research sponsored by ERDA Division of Reactor Development and DEmonstration, LMFBR Program.

†Computer Sciences Division.

5.12 INPUT DATA REQUIREMENTS FOR SPECIAL PROCESSORS IN THE COMPUTATION SYSTEM CONTAINING THE VENTURE NEUTRONICS CODE*

D. R. Vondy T. B. Fowler G. W. Cunningham†

(Abstract of ORNL-5229, November, 1976)

This report presents user input data requirements for certain special processors in a nuclear reactor computation system. These processors generally read data in formatted form and generate binary interface data files. Some data processing is done to convert from the user-oriented form to the interface file forms. The VENTURE diffusion theory neutronics code and other computation modules in this system use the interface data files which are generated.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

†Computer Sciences Division.

5.13 PROGRAMMING PRACTICES AND COMPUTER CODE DEVELOPMENT*

D. R. Vondy

(Abstract of ORNL/TM-5065, December, 1975)

This report presents recommendations to programmers developing computer codes. These are of special importance in effort on those codes which will be used at other installations on other computers. Many of the code exchange problems are addressed. Certain local routines are described which perform special tasks, many of which are not possible within the scope of the Fortran language.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

5.14 ITERATIVE SOLUTION OF THE DIFFUSION AND P_1 FINITE ELEMENT EQUATIONS*

E. T. Tomlinson† J. C. Robinson‡ D. R. Vondy

(Abstract of ORNL/TM-5224, February, 1976, and of paper to be published in June issue of Nuclear Science and Engineering)

The purpose of this work is to develop a method for obtaining solutions to the time-independent Boltzmann neutron transport equation on triangular grids with nonorthogonal boundaries and anisotropic scattering. A functional is developed from the canonical form of the multigroup transport equation. The angular variable is then removed by expanding the functional in spherical harmonics retaining only the first two moments and limiting the anisotropic scattering to be linear. The finite element method is then implemented using quadratic Lagrange-type interpolating polynomials to span the spatial domain.

The resultant set of coupled linear equations is then solved iteratively. The applicability of convergence acceleration techniques developed for the finite difference method are tested and implemented where appropriate.

Finally, a number of numerical experiments are performed to evaluate the performance of the proposed method. The results are compared to results obtained by various established methods. In all cases, agreement is excellent.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

†Computer Sciences Division.

‡The University of Tennessee.

5.15 CALCULATION OF REACTIVITY CHANGES DUE TO BUBBLE COLLAPSE*

T. J. Hoffman† L. M. Petrie†

(Summary of paper to be presented at American Nuclear Society Annual Meeting, June 12-17, 1977, New York City)

Calculations[1] based on Behrens' method[2] indicate that a substantial increase in reactivity may accompany the collapse of a large number of small bubbles in an LMFBR core. More sophisticated transport approaches[3-5] to this problem have encountered several difficulties: the large number of bubbles requires many mesh points; the desired effect can easily be masked by the movement of fuel to regions of greater (or lesser) importance; the reactivity is desired for a random distribution of spherical bubbles. This paper describes a transport approach to this problem which avoids the above difficulties by using the "subgroup"[6] or "probability table"[7,8] method.

Consider the transport of neutrons in a system that contains a uniform (random) distribution of bubbles in a compressed fuel mixture (bubble system). The probability that the neutron will travel a distance z through the compressed fuel and undergo its next collision in dz of the fuel is

$$f(z)dz = \Sigma_B e^{-\Sigma_B z} dz \ , \tag{1}$$

where Σ_B is the total cross section of the compressed fuel; i.e., $\Sigma_B = \Sigma_H/(1 - \alpha)$, where Σ_H is the total cross section of the fuel in the homogeneous system (the system in which the fuel has expanded to fill the bubbles) and α is the void fraction. Since bubbles are present, the neutron's flight path, s, will be longer than z. The lengthening of the flight path will be the product of the number of bubbles encountered along z, n, and the mean chord length of the bubbles, $\bar{c}$, i.e.,

$$s = z + n\bar{c} \ . \tag{2}$$

If Σ denotes the probability of bubble encounter per unit distance along z, i.e., $\Sigma = \alpha/\bar{c}(1 - \alpha)$, then the probability of encountering n bubbles will be a Poisson distribution:[9]

$$f_n = \frac{(\Sigma z)^n \ e^{-\Sigma z}}{n!} \ . \tag{3}$$

When s is determined with Eq. (2) by selecting z from Eq. (1) and n from Eq. (3), the probability distribution function (pdf) for the next collision site in the bubble system is

$$p_B(s) = \Sigma_B \sum_{n=0}^{s/\bar{c}} \frac{[\Sigma(s - n\bar{c})]^n \ e^{-(\Sigma_B + \Sigma)(s - n\bar{c})}}{n!} \ . \tag{4}$$

The corresponding pdf in the homogeneous system is

$$p_H(s) = \Sigma_H \ e^{-\Sigma_H s} \ . \tag{5}$$

These distributions, $p_B(s)$ and $p_H(s)$, have the same mean value, $1/\Sigma_H$. However, their second moments are different. It is this change in the second moment, and hence in the migration area, that causes the change in reactivity when the bubbles collapse.

Equation (4) can be viewed as the transport of neutrons in a system with a variable cross section, σ_B. One first selects a cross section from a distribution $p(\sigma_B)$ and then uses this cross section to select a distance from

$$\hat{p}_B(s) = \sigma_B \ e^{-\sigma_B s} \ . \tag{6}$$

If $p(\sigma_B)$ is properly distributed, then the distribution of collision sites from Eq. (6) will be the same as that from Eq. (4). Following the procedure used by Nikolaev[6] to treat neutron transport in the unresolved energy region, we let

$$p(\sigma_B) = \sum_i p_i \ \delta(\sigma_B - \Sigma_i) \ . \tag{7}$$

The distribution of collision sites will then be

$$\hat{p}_B(s) = \sum_i p_i \ \Sigma_i \ e^{-\Sigma_i s} \ . \tag{8}$$

The p_i's and Σ_i's are determined in such a manner that the moments of $p_B(s)$, Eq. (4), are conserved, i.e.,

$$\int_0^\infty s^n \ \hat{p}_B(s) \ ds = n! \ \sum_i \frac{p_i}{\sum_i^n} = \int_0^\infty s^n \ p_B(s) \ ds \ , \quad n = 0,1,2 \ ,\ldots \tag{9}$$

By reducing the distribution for collision sites in the bubble system to a set of cross sections and their probabilities, i.e., a probability table,[7] we are able to solve the problem with discrete ordinates.[8] Hence, a solution to a stochastic problem is obtained with a deterministic calculation.

The results of a one-group calculation ($\Sigma_{t} = 0.28\ cm^{-1}$, $\gamma\Sigma_{f} = 0.0075\ cm^{-1}$, $k_{\infty} = 1.488$, $\alpha = 0.2$) for a bare cylindrical core ($R = 74.25$ cm, $H = 91.44$ cm) are shown in Table 5.15.1. The reactivities calculated with this probability table approach, Δk_{pt}, were obtained with the discrete ordinates code, DOT.[10] The agreement with the reactivities predicted with Behrens' formula for "closely-spaced holes," $\Delta k_{\text{Behrens}}$, is excellent.

Table 5.15.1 Reactivity changes caused by bubble collapse

Bubble Radius (cm)	Δk_{pt}	$\Delta k_{\text{Behrens}}$
0.25	0.0027	0.0028
0.50	0.0053	0.0055
1.00	0.0105	0.0110
2.00	0.0207	0.0220

With the approach developed in this paper, the reactivity associated with bubble collapse can be calculated for reflected, multiregion, partially molten LMFBR cores as well as boiling in LWR. This approach is not limited to one-group problems.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

†Computer Sciences Division.

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6. ANALYSES FOR SPECIFIC SYSTEMS OR APPLICATIONS

6.1 THE CALCULATED PERFORMANCE OF VARIOUS STRUCTURAL MATERIALS IN FUSION-REACTOR BLANKETS*

M. L. Williams R. T. Santoro T. A. Gabriel

(Abstract of Nucl. Tech. 29, 384, 1976, and of ORNL/TM-5036, December, 1975)

The calculated nuclear performances of niobium, SS-304, and nimonic-105 as structural materials in a conceptual D-T fusion-reactor blanket model are compared. For each structural material, the tritium breeding ratio, the energy-deposition rate, the operating dose, the time dependence of the neutron-induced activity, the time dependence of the dose from the activation products, the time dependence of the nuclear afterheat, and the atomic displacement rate are calculated. Emphasis is placed on the nuclear response in the first structural wall to the selected structural material for an assumed neutron wall loading of 1 MW/m^2. Taking into account all of the nuclear responses, SS-304 appears to be a reasonable choice as the structural material for fusion-reactor application.

*Research sponsored by ERDA Division of Magnetic Fusion Energy (through ORNL Fusion Energy Division).

6.2 UNCERTAINTIES IN CALCULATED HEATING AND RADIATION DAMAGE IN THE TOROIDAL FIELD COIL OF A TOKAMAK EXPERIMENTAL POWER REACTOR DUE TO NEUTRON CROSS-SECTION ERRORS*

R. G. Alsmiller, Jr. J. Barish C. R. Weisbin

(Abstract of ORNL/TM-5198, March, 1976)

Calculated results are presented of the uncertainties in the neutron scalar flux, the energy deposition per unit volume, and the displacements per atom in the toroidal field coil of a tokamak experimental power reactor due to neutron cross-section errors in iron and carbon which are major constituents of the blanket-shield-coil configuration considered. The calculations were carried out using perturbation theory to obtain sensitivity profiles for the various cross sections of interest, and these profiles were then combined with cross-section error estimates, including correlations, to obtain the uncertainties.

Each of the three responses — the neutron scalar flux, the energy deposition per unit volume, and the displacements per atom — is found to be very sensitive to the cross sections in the energy group which contains the source ($\sim$14 MeV since a D-T source is assumed), and each of the responses is found to have a relative standard deviation of approximately 100% due to neutron cross-section errors in iron.

*Research sponsored by ERDA Division of Magnetic Fusion Energy (through ORNL Fusion Energy Division).

6.3 NEUTRONICS AND PHOTONICS CALCULATIONS FOR THE TOKAMAK EXPERIMENTAL POWER REACTOR*

R. T. Santoro V. C. Baker† J. M. Barnes‡

(Abstract of ORNL/TM-5466, in press; also abstract of paper presented at 9th Symposium on Fusion Technology, June 14-18, 1976, Garmisch-Partenkirchen, Federal Republic of Germany)

The results of one-dimensional neutronic and photonic calculations that compare the nuclear performance of blanket and shield designs proposed for use in the Tokamak Experimental Power Reactor are presented. The nuclear analysis was carried out for both

nonbreeding and tritium-breeding blanket modules to compare the spatial variations of the radiation flux and energy distributions, nuclear heating, radiation damage, and tritium breeding. Nonbreeding blanket modules that contain potassium plus SS-316 or potassium only as the energy-absorbing medium and breeding blankets that use lithium as the fertile material were evaluated as a function of the first-wall cooling scheme.

*Research sponsored by ERDA Division of Magnetic Fusion Energy (through ORNL Fusion Energy Division).

†The University of Tennessee.

‡Computer Sciences Division.

6.4 MONTE CARLO ANALYSIS OF THE EFFECTS OF SHIELD PENETRATIONS ON THE PERFORMANCE OF A TOKAMAK FUSION REACTOR*

R. T. Santoro J. S. Tang† R. G. Alsmiller, Jr. J. M. Barnes†

(Summary of paper to be presented at the Fifth International Conference on Reactor Shielding, April 18-22, 1977, Knoxville, Tennessee)

Calculations have been performed using the Monte Carlo radiation-transport code MORSE[1] to estimate the effects on the performance of a D-T burning Tokamak fusion reactor resulting from radiation that streams through penetrations in the blanket-shield assembly. The number of penetrations, as well as the size of some of these penetrations, is appreciable, so the capability of the blanket and shield in attenuating the plasma neutron and secondary-gamma radiation is reduced. The radiation that streams through these penetrations can lead to intolerable nuclear heating and radiation damage in vital reactor components, particularly the cryogenic toroidal-field coils that surround the reactor and the cryopumping surfaces inside the injector.

This paper summarizes the results of Monte Carlo calculations that were carried out for a representative fusion reactor having a rectangular neutral-beam-injector port (30 x 70 cm^2) passing through the blanket and shield. The plasma region, blanket, shield, and toroidal-field coils were represented using cylindrical geometry having dimensions and compositions corresponding to those of the Experimental Power Reactor.[2-3] The radiation transport was accomplished using coupled 35-group neutron, 21-group gamma-ray cross sections obtained by collapsing the 100n-21γ cross-section library.[4,5] Energy deposition was estimated using fluence-to-kerma factors generated by MACK,[6] and radiation damage was computed using atomic, displacement, and gas-production cross sections generated by the code RECOIL.[7]

The nuclear heating and radiation damage to the toroidal-field coils adjacent to the injector port were estimated using forward and adjoint Monte Carlo methods. The presence of the neutral-beam-injector port leads to increases in both the nuclear heating and radiation damage in the toroidal-field-coil windings by factors of 50 to 100 over the same responses in fully shielded windings.

*Research sponsored by ERDA Division of Magnetic Fusion Energy (through ORNL Fusion Energy Division).

†Computer Sciences Division.

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6.5 NUCLEAR ENGINEERING. PART 4 OF THE OAK RIDGE TOKAMAK EXPERIMENTAL POWER REACTOR STUDY — 1976*

C. A. Flanagan, Editor†

E. S. Bettis‡ H. L. Watts§ J. T. Huxford§ R. T. Santoro

J. L. Scott‖ J. S. Watson# S. D. Clinton# R. N. Cherdack¶

(Abstract of ORNL/TM-5575, December, 1976)

The Experimental Power Reactor (EPR) studies which have been performed at Oak Ridge National Laboratory (ORNL), at Argonne National Laboratory (ANL), and at the General Atomic Company (GAC) during the last two years have investigated the design and development aspects associated with a large tokamak designed to produce significant power at a high duty cycle, to be operated in the mid-1980's, and to be the precursor of a fusion Demonstration Reactor Plant (Demo).

At ORNL, the first year was devoted to scoping studies. At the end of this effort, a reference concept was selected. During the past year, the reference concept has been pursued in more depth; based on these more detailed investigations, major research and development needs have been identified and documented.

The results of the past year's design effort in the nuclear engineering areas are documented in this report. The discussion covers materials considerations, first radiation wall, mechanical design, neutronics, heat transfer, and tritium handling. Five companion reports being issued as parts of this composite EPR report present information in the other discipline areas.

*Research sponsored by ERDA Division of Magnetic Fusion Energy (through ORNL Fusion Energy Division).

†Westinghouse Electric Company.

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#Chemical Technology Divsion.

¶Burns and Roe, Inc.

6.6 NEUTRONICS CALCULATIONS FOR THE TFTR NEUTRAL BEAM INJECTORS*

R. T. Santoro R. G. Alsmiller, Jr.

(Abstract of ORNL/TM-5608, May, 1976)

Estimates, based entirely on one-dimensional transport calculations, of some of the effects of radiation on the operation and maintenance of the neutral beam injector for the Tokamak Fusion Test Reactor (TFTR) to be built at the Plasma Physics Laboratory of Princeton University are presented. Radiation effects due to 14-MeV neutrons produced by D-T reactions in the plasma and due to 2.6-MeV neutrons produced by D-D reactions in the calorimeter and in the charged-deuteron beam dump are considered. The results presented here are intended to indicate potential radiation problems rather than to be an accurate estimate of the magnitude of the actual radiation effects that will exist in the vicinity of the final injectors. This is particularly true since the results presented here are based on early injector design data, some of which are no longer applicable.

For 14-MeV neutrons, estimates are given of (1) the heating and activation of the toroidal field (TF) coils adjacent to the injector ports; (2) the activation of the injector superstructure; and (3) the heating in the cryopanel assemblies. For 2.6-MeV neutrons, estimates are given of (1) the activation of the calorimeter structure, and (2) the dose rates in the vicinity of the charged-deuteron beam dump. It is to be noted that no estimate is given here of the activation of the charged-deuteron bending magnet since at the time these calculations were performed no design for this magnet was available.

*Research sponsored by ERDA Division of Magnetic Fusion Energy (through ORNL Fusion Energy Division).

6.7 THE ELMO BUMPY TORUS REACTOR (EBTR) REFERENCE DESIGN*

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L. M. Lidsky[¶] D. A. Ehst[¶] J. S. Herring[¶]
D. L. Kaplan[¶] A. Pant[¶] R. E. Potok[¶]
P. B. Burn[#]

(Abstract of ORNL/TM-5669, November, 1976, and of Trans. Am. Nucl. Soc. 24, 42, 1976; also abstract of paper presented at American Nuclear Society Second Topical Meeting on the Technology of Controlled Nuclear Fusion, September 21-23, 1976, Richland, Washington)

The goal of the ELMO Bumpy Torus Reactor (EBTR) study is the evaluation of the EBT confinement concept as the basis for development of a commercial fusion power reactor. A multidisciplinary, self-consistent treatment of EBT reactor scaling and design has been completed and a reference design (EBIR-48) has been developed. This design, based on a realistic plasma model and relatively conservative engineering parameters (i.e., 1 MW/m^2 neutron wall loading and a 7.3 T maximum toroidal field), is a steady state, ignited-mode system with high plasma power density and aspect ratio. The total thermal power of EBIR-48, exclusive of blanket multiplication, is 4000 MW; the design is based on a standard module and the design power level for a particular plant is determined by the number of modules used. Several design variants have been investigated in detail to illustrate the effect of near-term and advanced technologies and to illustrate the design freedom offered by devices with low field and high aspect ratio. The high aspect ratio simplifies many aspects of the design, most notably those associated with remote maintenance, accessibility, and repair. It appears that a commercially successful EBTR could be constructed with only slight advances in existing technology, if the present understanding of the physics can be extrapolated to the reactor regime and does not differ markedly from the model developed for this study.

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†Exxon Nuclear Co., Inc.

‡Fusion Energy Division.

§Consultant.

‖Computer Sciences Division.

#Engineering Division.

¶Massachusetts Institute of Technology.

6.8 RADIATION-DAMAGE CALCULATIONS: PRIMARY KNOCK-ON ATOM SPECTRA, DISPLACEMENT RATES, AND GAS-PRODUCTION RATES*

T. A. Gabriel J. D. Amburgey† N. M. Greene†

(Abstract of Nucl. Sci. Eng. 61, 21, 1976, and of ORNL/TM-5160, March, 1976)

A heavy charged-particle recoil data base [primary knock-on atom (PKA) spectra] and an analysis program have been created to assist experimentalists in studying, evaluating, and correlating radiation-damage effects in different neutron environments. Since experimentally obtained controlled-thermonuclear-reactor-type neutron spectra are not presently available, the data base can be extremely useful in relating currently obtainable radiation damage to that which is anticipated in future fusion devices. However, the usefulness of the data base is not restricted to just CTR needs. Most of the elements of interest to the radiation-damage community and all neutron reactions of any significance for these elements have been processed, using available ENDF/B-IV cross-section data, and are included in the data base. Calculated data such as primary recoil spectra, displacement rates, and gas-production rates, obtained with the data base, for different radiation environments are presented and compared with previous calculations.

*Research sponsored by ERDA Division of Magnetic Fusion Energy (through ORNL Metals and Ceramics Division).

†Computer Sciences Division.

6.9 THE SPATIAL VARIATION OF THE DAMAGE ENERGY AND GAS PRODUCTION IN THE EXPERIMENTAL VOLUME OF A Li(*D*,*n*) NEUTRON RADIATION DAMAGE FACILITY*

R. G. Alsmiller, Jr. J. Barish†

(Abstract of ORNL/TM-5554, October, 1976, and of paper to be published in Nuclear Technology)

Calculated results are presented of the variation with position in the experimental volume of a Li(*D*,*n*) neutron radiation damage facility of the damage energy and helium and hydrogen production in copper and in niobium when this volume is partially filled with experimental samples. The neutron nonelastic cross-section data at the higher energies ($\gtrsim$15-20 MeV) needed to carry out the transport calculations were obtained from the intranuclear-cascade model of nuclear reactions.

*Research sponsored by ERDA Division of Magnetic Fusion Energy (through ORNL Fusion Energy Division).

†Computer Sciences Division.

6.10 THE APPLICATION OF NEUTRON TRANSPORT CODES TO THE TRANSPORT OF NEUTRAL ATOMS IN PLASMAS*

J. H. Marable E. M. Oblow

(Abstract of ORNL/TM-5164, February, 1976, and of Nucl. Sci. Eng. 61, 90, 1976)

The application of the linear Boltzmann equation as used in reactor and shielding problems to the transport of neutral atoms in a Tokomak-type plasma has been studied. The method was found to be generally valid with some limitations because of possible nonisotropy of the plasma medium.

Effective cross sections for the interaction of neutral atoms with an isotropic plasma were calculated and applied to the transport of hydrogen in a typical ORMAK plasma. The outer wall was found to have a significant effect on the hydrogen concentration.

*Research sponsored by ERDA Division of Magnetic Fusion Energy.

6.11 MODELING THE EDGE OF A TOKAMAK PLASMA*

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(Summary of paper presented at Symposium on Plasma Wall Interaction, October 18-22, 1976, Julich, Federal Republic of Germany)

A substantial fraction of the ohmic heating input power in tokamaks is radiated, primarily at the plasma edge through low-Z impurity line radiation. By using a time-dependent ionization and recombination model, we find that if low-Z impurities are stationary they probably do not radiate enough power to account for the observed power loss. Thus, we infer, in corroboration with some specific impurity transport measurements on tokamaks, that impurities must be recycling from the wall deep into the plasma and out again, on time scales not much longer than the energy containment time. The (neo)classical influx of impurities provides an inward transport mechanism, but the outward transport process is unknown. However, a recent and more exact (no mass ratio expansion) impurity transport calculation[1] has shown that a "temperature-screening" effect may inhibit or reverse the influx of low charge states of low-mass impurities. A similar quandry occurs in gas puffing experiments where the electron density at the plasma center rises rapidly. We have performed detailed calculations of the neutral transport in the plasma edge, including energy-cascading, reflection and a proper accounting of the (small) power lost via charge-exchange through an ANISN-type (XCDRN) transport code. We find that the density rise observed cannot be explained by the combination of neutral influx and particle diffusion.

In order to explain these apparent paradoxes, we hypothesize a "new" transport model based on the generalized theory of dissipative trapped-electron instabilities. In this model, the anomalous transport is in addition to neoclassical transport and affects only the "cross-field" processes of particle diffusion and electron heat conduction. In the resultant model the electron temperature is determined by the anomalous electron heat transport, as usual. However, with regard to apparent paradoxes discussed above, the particle density is determined by balancing the inward Ware pinch with the outward anomalous particle diffusion; the impurity density results from balancing the inward classical diffusion against the outward anomalous diffusion. The inward Ware pinch is found[2] to be sufficiently rapid to explain the density increases observed in ORMAK gas puffing experiments, and in ALCATOR, at least up to 4×10^{14} cm^{-3}. The impurity balance that obtains from the balancing of the classical influx (or outflux[1]) against the anomalous outflux and its dependence on the boundary conditions in the plasma edge will be discussed.

*Research sponsored by ERDA Division of Magnetic Fusion Energy.

†Fusion Energy Division.

1. K. T. Tsang and E. C. Crume, "Temperature Screening Effect in Two-Ion Species Pfirsch-Schluter Transport," ORNL/TM-5366 (1976).
2. H. C. Howe, "Evidence for a Density Pinch in Tokamaks," paper at DPP-APS, San Francisco, November 15-19, 1976.

6.12 CALCULATED PERFORMANCE OF IRON-ARGON AND IRON-PLASTIC CALORIMETERS FOR INCIDENT HADRONS WITH ENERGIES OF 5 TO 75 GeV*

T. A. Gabriel W. Schmidt†

(Abstract of ORNL/TM-5105, February, 1976, and of Nucl. Instr. Methods 134, 271, 1976)

The calculated responses of iron-argon and iron-plastic calorimeters for incident hadrons with energies of 5 to 75 GeV are presented. The responses calculated are energy resolution vs energy, energy resolution vs the thickness of the sampling plates, the angular and spatial root-mean-square deviations (i.e., the ability to determine the incident particle's entrance angle and impact point), and the spatial properties of the average and individual hardronic cascades. Some comparisons are made with experimental data; however, the main purpose of this paper is to provide specific design information for these types of calorimeters.

*Research sponsored by ERDA Division of Physical Research and by the Bündesministerium für Forschung und Technologie der BRD.

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6.13 THE CALCULATED RESPONSE OF A LIQUID-SCINTILLATOR TOTAL-ABSORPTION HADRON CALORIMETER*

R. T. Santoro J. D. Amburgey† T. A. Gabriel

(Abstract of ORNL/TM-5162, January, 1976, and of Nucl. Instr. Methods 134, 87, 1976)

The calculated performance of a large-volume, all-liquid-scintillator calorimeter for incident pions having momenta in the range of 3 to 20 GeV/c is presented. The calculated scintillation pulse-height distribution, energy deposition, leakage energy, and pulse-height resolution as a function of pion energy are given. Calculated and experimental pulse-height distributions are compared for 20-GeV/c incident pions.

*Research sponsored by ERDA Division of Physical Research.

†Computer Sciences Division.

6.14 CHARGED HADRON AND LEPTON CURRENTS PRODUCED BY LOW-MOMENTUM ($\leqslant$3 GeV/c) CHARGED PIONS IN Al, Fe, AND Pb TARGETS*

P. S. Beiser† T. A. Gabriel J. D. Amburgey‡

(Abstract of ORNL/TM-5677, December, 1976)

Calculations have been carried out to determine the spatial dependence of charged hadron and lepton currents produced by low-momentum ($\leqslant$3 GeV/c) charged pions in Al, Fe, and Pb targets. Even at the lowest momentum values considered (0.5 GeV/c), there is little difference between incident positive pions and negative pions with respect to the average spatial dependence of the charged current.

*Research sponsored by ERDA Division of Physical Research.

†Oak Ridge Associated Universities undergraduate research trainee.

‡Computer Sciences Division.

6.15 CALIBRATION CALCULATIONS OF HADRONIC CASCADES INDUCED BY HIGH-ENERGY MUONS IN IRON/PLASTIC CALORIMETERS*

T. A. Gabriel J. D. Amburgey†

(Abstract of ORNL/TM-5615, December 1976, and of paper to be published in Nuclear Instruments and Methods)

Calculations have been performed to determine the response of an iron-plastic calorimeter to muon-induced nuclear interactions. The calculated data include energy resolutions and average pulse-height signals as a function of energy transfer, energy-transfer angle, and spatial-interaction point.

*Research sponsored by ERDA Division of Physical Research.

†Computer Sciences Division.

6.16 HADRONIC- AND ELECTROMAGNETIC-CASCADE DISCRIMINATION IN A THIN LEAD-ARGON CALORIMETER*

T. Jensen† J. D. Amburgey‡ T. A. Gabriel

(Abstract of ORNL/TM-5679, February, 1977, and of paper to be published in Nuclear Instruments and Methods)

Calculations have been carried out to determine methods or a combination of methods for discriminating between high-energy ($\gtrsim$15 GeV) hadrons and leptons or photons incident on a thin lead-argon calorimeter. Some of the methods considered involve fluctuations in shower development, the centroid of the shower, and ratios of the pulse height obtained from various locations within the device. Of these, the latter appears the most promising.

*Research sponsored by ERDA Division of Physical Research and by The University of Rochester.

†The University of Rochester.

‡Computer Sciences Division.

6.17 NEUTRON KERMA FACTORS FOR H, C, N, O, AND TISSUE IN THE ENERGY RANGE OF 20 TO 70 MeV*

R. G. Alsmiller, Jr. J. Barish†

(Abstract of ORNL/TM-5702, December, 1976, and of paper to be published in Health Physics)

Calculated kerma factors (kerma per unit fluence) in the energy range of 20 to 70 MeV based on nonelastic charged-particle-production cross-section data obtained from the intranuclear-cascade model of nuclear reactions are given for H, C, N, O, and tissue.

*Research sponsored by ERDA Division of Physical Research.

†Computer Sciences Division.

6.18 PRELIMINARY REPORT ON THE PROMISE OF ACCELERATOR BREEDING AND CONVERTER REACTOR SYMBIOSIS (ABACS) AS AN ALTERNATIVE ENERGY SYSTEM[*]

F. R. Mynatt, Study Group Leader

Target Physics Studies — R. G. Alsmiller, Jr., J. Barish,[†] and T. A. Gabriel

Nuclear Engineering — D. E. Bartine and T. J. Burns

Accelerator Design — J. A. Martin[‡] and M. J. Saltmarsh[‡]

Heat Transfer and Mechanical Design — E. S. Bettis[§]

(Abstract of ORNL/TM-5750, February, 1977)

A preliminary study has been performed to evaluate the promise of accelerator breeding and converter reactor symbiotic systems (ABACS) as an alternate fission power technology which can make full utilization of the energy content of uranium and thorium ores. ABACS is, therefore, considered as an alternative to fast breeder reactors for extending our energy supply. An explanation is given of the fundamentals of accelerator breeding in which ^{233}U or ^{239}Pu fissile fuel is produced in a target/blanket system as a result of irradiation with an intense high-energy proton beam. Neutronics and heat transfer analyses are performed for three accelerator breeder concepts based on technologies of the liquid metal fast breeder, molten salt, and gas-cooled fast breeder reactors. Several converter reactors are considered, and the mass flows and economics of the complete symbiosis are presented. Particular attention is given to the potential advantages of ABACS relative to the fast breeder reactor in the areas of inherent safety and in the implementation of the ^{233}U-^{238}U denatured fuel cycle as a proliferation and diversion deterrent. Advantages and disadvantages of the present accelerator breeder concepts are summarized and development needs are indicated.

[*]Research funded by ERDA Division of Physical Research.

[†]Computer Sciences Division.

[‡]Physics Division.

[§]Consultant.

6.19 FEASIBILITY OF DENATURED LMFBR'S[*]

T. J. Burns D. E. Bartine

(Summary of paper to be presented at American Nuclear Society Annual Meeting, June 12-17, 1977, New York City)

The use of denatured fuel in nuclear reactors has been proposed as a possible means for alleviating some of the safeguard concerns of the nuclear fuel cycle.[1] In such a fuel, the fissile component would not be chemically separable from the rest of the fuel. Further, such a fuel would be subject to enrichment limitations to prevent direct use as weapons material. This paper summarizes the results of a study directed at investigating the possible use of such a fuel (^{233}U diluted with ^{238}U) in an LMFBR.

The reactor utilized for this study was based on a commercial-sized [1200 MW(e)] conceptual design. The standard two-zone Pu-fueled core surrounded with depleted uranium blankets was taken as the reference case. Preliminary calculations on the BOL core were done by substituting the denatured fuel in place of the reference core fuel. The radial blanket was also replaced by thorium oxide. The results of some of these calculations are summarized in Table 6.19.1. Also tabulated are some of the results of an analysis of a heterogeneous model in which thorium radial blanket assemblies are interspersed with core assemblies.

Table 6.19.1. Calculational results for LMFBR models

	Reference		Denatured Homogeneous		Denatured Heterogeneous	
I. Model						
a. Fuel	Pu/^{238}U		^{233}U/^{238}U		^{233}U/^{238}U	
b. Radial Blanket	^{238}U		^{232}Th		^{232}Th	
c. Internal Blanket					^{232}Th	
d. Core Fissile Inventory (kg)	2933		2512		3241	
e. Internal Blanket Fraction					33.9%	
II. Performance Parameters						
a. k	1.0369		1.0369		1.0369	
b. Core Assembly Conversion Ratio	0.917		0.836		0.355	
c. Breeding Ratio						
1. Core Assemblies	0.906	1.107	0.831	1.020	0.350	0.431
2. Axial Blankets	0.201		0.189		0.081	
3. Radial Blanket	0.222		0.214		0.310	0.783
4. Internal Blanket					0.473	
Total	1.329		1.234		1.214	

As indicated in Table 6.19.1, the primary penalty in using denatured fuel in the reference LMFBR is a significant decrease in the breeding ratio for both denatured cases given. Similarly, the doubling time for both reactors is greater than for the reference reactor, although this effect for the two-zone case is somewhat ameliorated by a smaller fissile inventory. The heterogeneous model, while requiring a larger fissile inventory, does, however, dramatically alter the isotopics of the bred fissile material (64% ^{233}U vs 17% for the two-zone case). Even for this case, it is evident that the denatured LMFBR may not be viable as a stand-alone reactor since the plutonium bred as a consequence of the denaturing process is unavailable for recycle in the denatured reactor. Moreover, even if alternate designs increase the ^{233}U breeding ratio to a value greater than unity, a source of ^{233}U for the initial core will still be required.

Owing to these considerations, a symbiotic system such as that depicted in Fig. 6.19.1 is envisioned. The denatured reactors would operate in dispersed locations. All fuel reprocessing would be confined to a safeguarded area. Also located within the safeguarded area would be reactors designed to utilize the plutonium produced in the denatured reactors (which is not allowed outside the safeguard area after reprocessing) to produce ^{233}U. The mass balances involved in such a symbiosis will be discussed, with particular emphasis on maximizing the number of outside denatured reactors supported per inside "converter" reactor.

*Research sponsored by ERDA Division of Nuclear Research and Applications.

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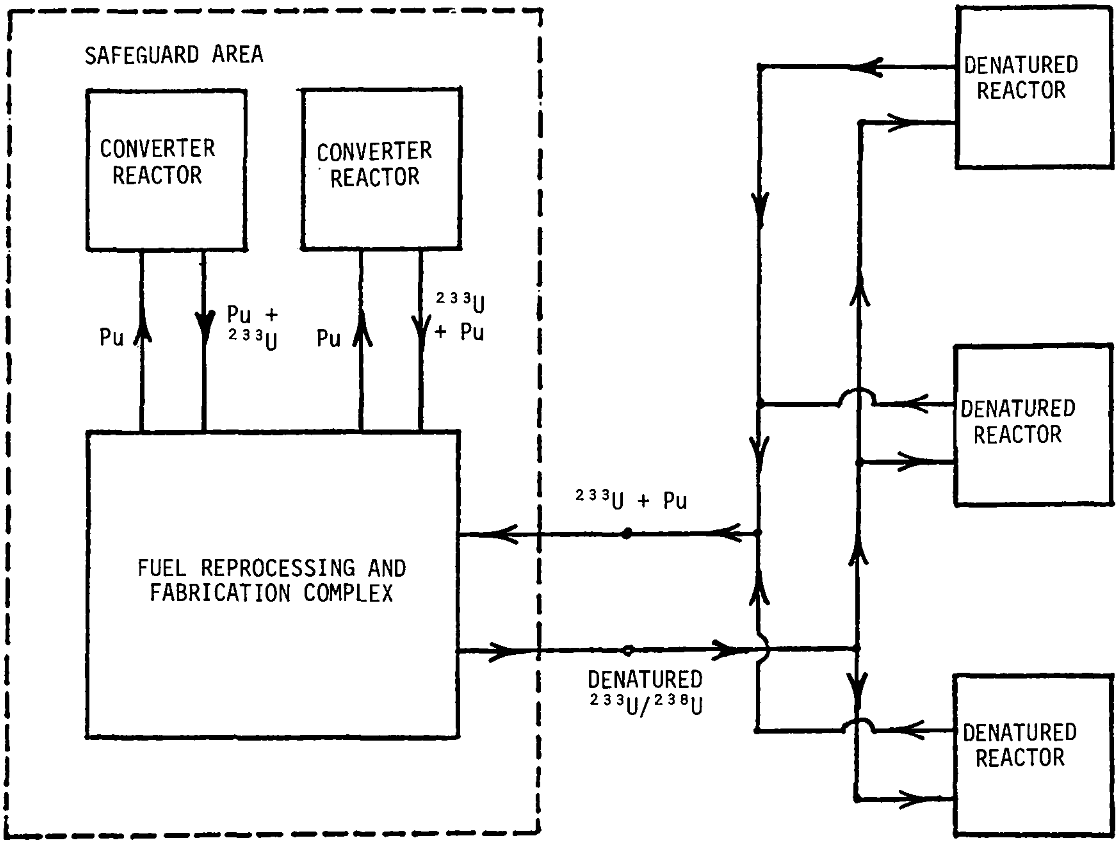

Fig. 6.19.1. Fissile flow diagram of denatured LMFBR fuel cycle with fertile thorium.

6.20 THORIUM ASSESSMENT STUDY QUARTERLY PROGRESS REPORT FOR FIRST QUARTER FISCAL 1977*

I. Spiewak,† Program Director D. E. Bartine, Program Manager

Contributors:

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(Abstract of ORNL/TM-5818, March, 1977)

The objective of this program is to contribute to the ongoing assessment of the potential role of thorium fuel cycles for alleviating safeguards concerns. Scenarios include (1) no fuel recycle permitted, (2) fuel recycle permitted only in secure regions ("energy parks") with denatured (chemically nonseparable) fuels only outside these regions, and (3) no limits on fuel recycle. A further objective is to provide nuclear mass balance data on HTGR's required by ERDA contractors for comparative cost-benefit studies.

*Research sponsored by ERDA Division of Nuclear Research and Applications.

†Engineering Technology Division.

6.21 ON NUCLEAR FUEL, MASS BALANCES, CONVERSION RATIO, DOUBLING TIME AND UNCERTAINTY*

D. R. Vondy

(Abstract of ORNL/TM-5050, November, 1976)

There is considerable interest in the performance characteristics of nuclear power plants. Concern over availability of fuel from ore leads directly to emphasis on the development of plants which breed fuel. This study addresses certain aspects of analysis of performance of and projections for nuclear reactor power plants.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

6.22 MULTIDIMENSIONAL LWR BENCHMARK PROBLEMS*

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B. Micheelsen# I. Misfeldt¶ D. R. Vondy W. Werner¶

(Reprint of Trans. Am. Nucl. Soc. 23, 211, 1976)

One of the functions of the Benchmark Problem Committee of the ANS Mathematics and Computation Division is to provide reliable solutions to well-defined mathematical benchmark problems in reactor statics. In particular, the establishment of solutions to large-scale three-dimensional (3-D) light-water reactor benchmark problems is needed to serve as standards for the verification of design codes and for the detailed error analysis of calculational methods.

The Benchmark Problem Committee has therefore adopted a 3-D two-group diffusion theory problem that was initially defined by Micheelsen and other participants of the 1971 IAEA panel on burnup physics. In May 1973, preliminary results obtained in 1972 using 17 computer codes in 7 countries were reported.[1] This paper summarizes the general characteristics of this 3-D IAEA benchmark problem and the status of the work presently being done by members of a benchmark subcommittee in an effort to generate reliable, high-precision results for this and a related 2-D IAEA problem.[2]

The problem specification[2] was chosen such that coarse-mesh and nodal codes can be easily applied. The 3-D IAEA problem represents a quarter-core of a medium-size PWR with two radial enrichment zones and a total number of 177 fuel elements (full core). The core is reflected by pure water without a steel baffle. The dimensions of the fuel subassemblies are 20 x 20 x 340 cm. A number of partially and fully inserted control rods, represented by absorber added to certain subassemblies, cause a strong nonseparable power distribution. In addition, the existence of a very large thermal flux peak in the reflector makes this a very difficult and challenging problem to solve. On the other hand, by the specification of homogenized two-group cross sections, the exclusion of nuclear feedback effects, and the choice of clean, fresh core conditions, phenomena that are not to be emphasized by this benchmark series are avoided.

During the past few years several authors have investigated the 2-D subproblem, and in 1975 a number of relatively accurate reference solutions were published.[2,3] More recently, finite-difference precision calculations with VENTURE (ref. 4) (mesh size h = 0.625 cm) and PDQ-7 (ref. 5) (h = 0.5 cm) of the 2-D IAEA problem provide a best eigenvalue estimate of 1.02958 ± 0.00001 and a peak-to-average power value of 1.5132 ± 0.0006 in the core at locations 31, 31 (cm). A second peak of ∿1.5333 is found at the core/reflector interface, position 130, 55.5.

Table 6.22.1 summarizes the data for eigenvalue and peak power density for the 3-D problem that have been obtained by members of the subcommittee using five computer codes based on different calculational methods. The generation of well-converged, reliable fine-mesh finite difference (FDM) solutions, containing in one case well over a million mesh points for the quarter core, proved to be a severe but successful test of the automated iteration and restart procedures. Good agreement, within 3 x 10^{-4}, is found in the eigenvalues obtained with mesh-centered and corner-mesh FDM codes. However, a noticeable discrepancy remains in the peak power value. This difference is consistent with earlier 2-D studies,[1] which also revealed a systematic difference between the two finite-difference approximations, indicating a somewhat better accuracy of the mesh-centered formulation for this particular problem.

Table 6.22.1. 3-D IAEA benchmark problem results

Computer Code	Method, Order of Approximation	Mesh	Eigenvalue	Peak/Average Power Value	Peak/Average Power Location
VENTURE[4]	Finite-difference method, mesh centered	17 x 17 x 19	1.02913	2.567	
		34 x 34 x 38	1.02864	2.504	
		68 x 68 x 76	1.02887	2.408	
		102 x 102 x 114	1.02896	2.378	
		Extrapolation	1.02903	2.354	
PDQ-7[5]	Finite-difference method, corner mesh	34 x 34 x 38	1.03054	2.039	35, 35, 170
		60 x 68 x 76	1.02933	2.266	32, 32, 175
FEM 3D[3]	Finite-element method, 2nd-order Lagrange interpolation	16 x 16 x 13	1.02917	2.298	32, 32, 174
				2.426	130, 56, 178
IQSBOX[6,7]	Nodal-expansion method, 4th-order polynomial	9 x 9 x 19	1.02875	2.412	30, 30, 170
		17 x 17 x 21	1.02903	2.356	30, 30, 170
	5th-order polynominal	9 x 9 x 19	1.02916	2.348	30, 30, 170
		17 x 17 x 21	1.02910	2.341	30, 30, 170
CUBOX[8]	Flux-expansion method, 3rd-order polynomial	9 x 9 x 19	1.02888	2.387	50, 30, 170
		17 x 17 x 19	1.02895	2.340	50, 30, 170

The remaining three solutions of Table 6.22.1 were calculated with higher order coarse-mesh codes with quite large mesh spacings. Definite conclusions about the quality of these solutions can only be drawn after a detailed comparison with the reference FDM solutions has been made. Nevertheless, the data in Table 6.22.1 and its convergence with mesh spacing indicate that finite element methods and improved higher order coarse-mesh techniques can produce accurate solutions for large-scale 3-D reactor problems with considerably reduced computing costs.

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‡Combustion Engineering.

§Ontario Hydro, Canada.

‖Danish AEC.

#The University of Munich, Germany.

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6.23 HTGR NEUTRONICS BENCHMARK PROBLEM*

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(Reprint of Trans. Am. Nucl. Soc. 23, 209, 1976)

The purpose of the high-temperature gas-cooled reactor (HTGR) Neutronics Benchmark Problem is to provide a static neutronics problem in triangular (hexagonal) geometry for which accurate solutions can be documented. These solutions will then provide a valuable means for checking the accuracy and efficiency of triangular-geometry static neutronics codes throughout the industry.

A simplified model of a Fort St. Vrain-size HTGR was chosen for this purpose. The salient features of that design have been retained within the model. Maintaining a realistic level of model complexity is important for code validation since the ultimate purpose of these codes is the analysis of such models.

The first level of problem definition, the Benchmark Source Situation,[1] provides a physical description of the problem. A three-dimensional (3-D) reactor model was chosen with 60-deg rotational symmetry in the horizontal plane. The reactor is modeled as a lattice of homogeneous, hexagonally shaped fuel elements surrounded by a graphite reflector. Two different core composition layers are defined axially. A control rod pattern consisting of a half-inserted central rod and a ring of six fully inserted rods is specified. Temperatures and isotopic compositions are given for initial hot conditions.

The second level of problem definition, the Benchmark Problem, proceeds from the source situation in further specifying a fully defined problem. From the given source situation, a number of benchmark problems can be defined. Two problems have been specified to date: a 2-D problem in the top half of the reactor and a 3-D problem. Both problems involve reducing the source situation by specifying four energy groups, spatial transport based on diffusion theory and a zero-flux outer boundary condition. Appropriately averaged macroscopic reaction rate cross sections are given for each composition. Contributed solutions for each problem are to have as primary results the neutron multiplication constant k_{eff}, iteration and time requirements, neutron balances and the fission rate density and fluxes both hexagon averaged and along traverses. Possible additional results for the 2-D problem are the above primary results for three other controlled states of the reactor (with the central rod removed, all rods removed, and all possible rods inserted).

The third and final level, the Benchmark Problem Solution, is the documented solution for a Benchmark Problem. At least two independently evaluated solutions (by different codes programmed by different people) are required for the acceptance of the Benchmark problem. These solutions must be sufficiently accurate and give or approach, through error refinement, the same result.

Solutions contributed to date have been for the 2-D problem. A preliminary coarse-mesh calculation on the 3-D problem has been performed, but fine-mesh 3-D calculations involving the participation of others will be needed to obtain acceptable solutions.

The 2-D problem has been solved by the authors using the finite – difference diffusion theory codes GRIMHX (Gregory, SRL),[2,3] VENTURE (Vondy, ORNL),[4] and BUG180 (Steinke, GAC).[5] GRIMHX and VENTURE solve mesh-centered difference equations, while BUG180 solves difference equations for an interface-centered mesh. Mesh element grids ranging from 1 to 54 mesh points per hexagon (mp/h) have been analyzed. K_{eff} values from the contributed solutions are given in Table 6.23.1. Comparing the 48-mp/h BUG180 solution with an estimate for the actual solution obtained by extrapolating the three BUG180 solutions to zero-mesh size indicates that its maximum flux fractional errors are <1% in the core and <5% (<2%, thermal group) in the reflector. This level of accuracy in the 48-mp/h BUG180 and 54-mp/h VENTURE solutions is felt to be sufficient for acceptance of the 2-D Benchmark Problem.

Benchmark problem specifications and their documented solutions will be distributed at the meeting.

Table 6.23.1. Neutron multiplication constant k_{eff} from contributed solutions of the 2-D HTGR benchmark problem

Computer Program		GRIMHX		VENTURE	BUG180
		Coarse Mesh[3]	Standard		
	1	1.11321	1.12725	1.12725	
	3	1.11735	1.12102		1.11672
Equivalent Number	6	1.11863	1.12028	1.12027	
Number of Mesh	12				1.11777
Per Hexagon	24			1.11929	
	48				1.11815
	54			1.11900	

*ORNL participation sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

†General Atomic Company.

‡Savannah River Laboratory.

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6.24 BENCHMARK ANALYSIS OF LMFBR NUCLEAR DESIGN METHODS*

E. Kujawski† H. S. Bailey†

(Abstract of paper to be presented at the American Nuclear Society Mathematics and Computation Division National Topical Meeting, March 28-30, 1977, Tucson, Arizona, and of paper to be published in Nuclear Science and Engineering; this paper reports LMFBR benchmark calculations performed in the Neutron Physics Division by D. R. Vondy, T. B. Fowler, and G. W. Cunningham‡)

The Large Core Code Evaluation Working Group has been organized for the prime aim of testing and validating neutronics codes and methods for use in the analysis of large fast reactors. For the initial problem static neutronics calculations were performed on a "representative" 1200-MWe LMFBR in a two-dimensional radial geometry. The initial results representing the cooperative effort of the participants are summarized. Various codes and methods using few-group diffusion theory are intercompared. The calculational methods are in substantial agreement provided the same multigroup constants are used. This effort is to be further pursued and broadened to investigate specific effects such as Na-voiding.

*ORNL participation sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

†General Electric Company, Fast Breeder Reactor Department, Sunnyvale, California.

‡Computer Sciences Division.

6.25 SUBCRITICALITY CALCULATIONS IN SUPPORT OF THE REACTIVITY SURVEILLANCE PROCEDURES EXPERIMENTS FOR THE FTR ENGINEERING MOCKUP FACILITY*

D. L. Selby G. F. Flanagan

(Abstract of ORNL-5061, October, 1976)

The results of discrete ordinate calculations are summarized. These were made at ORNL in support of the analysis of the subcritical experiments performed on the Fast Flux Test Reactor Engineering Mockup Critical located at Idaho Falls. Results indicate good agreement can be obtained between the simple modified source multiplication method and other more complex reactivity determination methods such as inverse kinetic rod drop or noise analysis provided accurate detection efficiencies can be calculated. The pointwise (zone-wise) convergence of the fluxes is given particular attention as a major problem in obtaining "accurate" detection efficiencies.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, CRBR Program.

6.26 SYSTEM FOR ASSAY OF FISSILE CONTENT OF SPENT LMFBR FUEL SUBASSEMBLIES*

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(Reprint of Trans. Am. Nucl. Soc. 23, 95, 1976)

A system was conceptually designed (Fig. 6.26.1) to nondestructively assay the total fissile content of a spent LMFBR fuel subassembly as it enters the head end of a fuel reprocessing plant. This system will help the plant operators to achieve better process control, criticality prevention, accountability, and safeguard of fissile materials. Mathematical analysis of this system, using the ORIGEN[1] and ANISN[2] codes, indicates that a real-time assay can be achieved with an uncertainty of 1-5% of the total fissile content of each subassembly for a plant throughput of 5 tons/day (2-4 subassemblies/hr). Other assay systems have been unsuitable for assaying spent LMFBR fuel subassemblies because of (1) interference by high neutron and gamma backgrounds from the fuel and (2) poor penetration of interrogating and/or signature radiations through the massive subassemblies.

The basic principle of this system is that an interrogating sub-MeV neutron source [produced in the D_2O by a (γ,n) reaction] is closely coupled to a spent fuel subassembly and the higher-energy signature neutrons, produced by fission of the fissile nuclides in the fuel, are detected by counters that are shielded by lead against gamma rays from the fuel and the gamma sources (S in Fig. 6.26.1). Neither the detectors nor the fertile fuel nuclides respond significantly to the interrogating neutrons. The subassembly is assayed as it is drawn either continuously or stepwise through the central cavity of the system (overall radius and active length each $\sim$45 cm). The arrangement is similar to that of Menlove et al.[3]

Each of the 12 ^{106}Ru-^{106}Rh sources ($t_{1/2}$ = 1 y) contains about 50 g of ^{106}Ru, the amount removed in the plant from one core-type subassembly. Gamma rays from these sources produce, in the D_2O, interrogating photoneutrons (E = $\sim$0.08 MeV) intense enough to produce a fission neutron signal in the fuel which exceeds the neutron background in the fuel. This background is mainly due to neutrons from spontaneous fission, but it also contains neutrons from (α,n) reactions and from self-interrogation by photoneutrons produced in the D_2O by gamma rays from the fuel to reduce the self-interrogation background, and the adjacent B_4C attenuates low-energy neutrons that would not properly penetrate the fuel.

The detectors (gas filled, proton recoil) are biased to set their detection threshold above 0.08 MeV, but as low as practical. This setting increases their detection efficiency and minimizes their discrimination against fission neutrons that have lost energy by scattering in escaping the fuel. The gamma-ray dose rate at the detectors is limited to a tolerable 1 r/hr by the massive outer lead shield.

Calculations indicate that the fissile assay sensitivity (counts per fissile atom) varies <4% over the entire cross section of an LMFBR subassembly, either core or blanket type. Averaged over a subassembly, the sensitivity is only 23% greater for a core vs a blanket type. The count rates are high, so for 1% counting statistics a core subassembly may require $\sim$10 sec and a blanket subassembly $\sim$1 min.

The authors conclude that this system could assay spent LMFBR fuel subassemblies for total fissile content with an uncertainty of 1-5% at the high throughput rate of a plant having a capacity of 5 tons/day. The equipment should be relatively inexpensive and practical for use in a commercial fuel reprocessing plant.

ORNL-DWG 76-2756

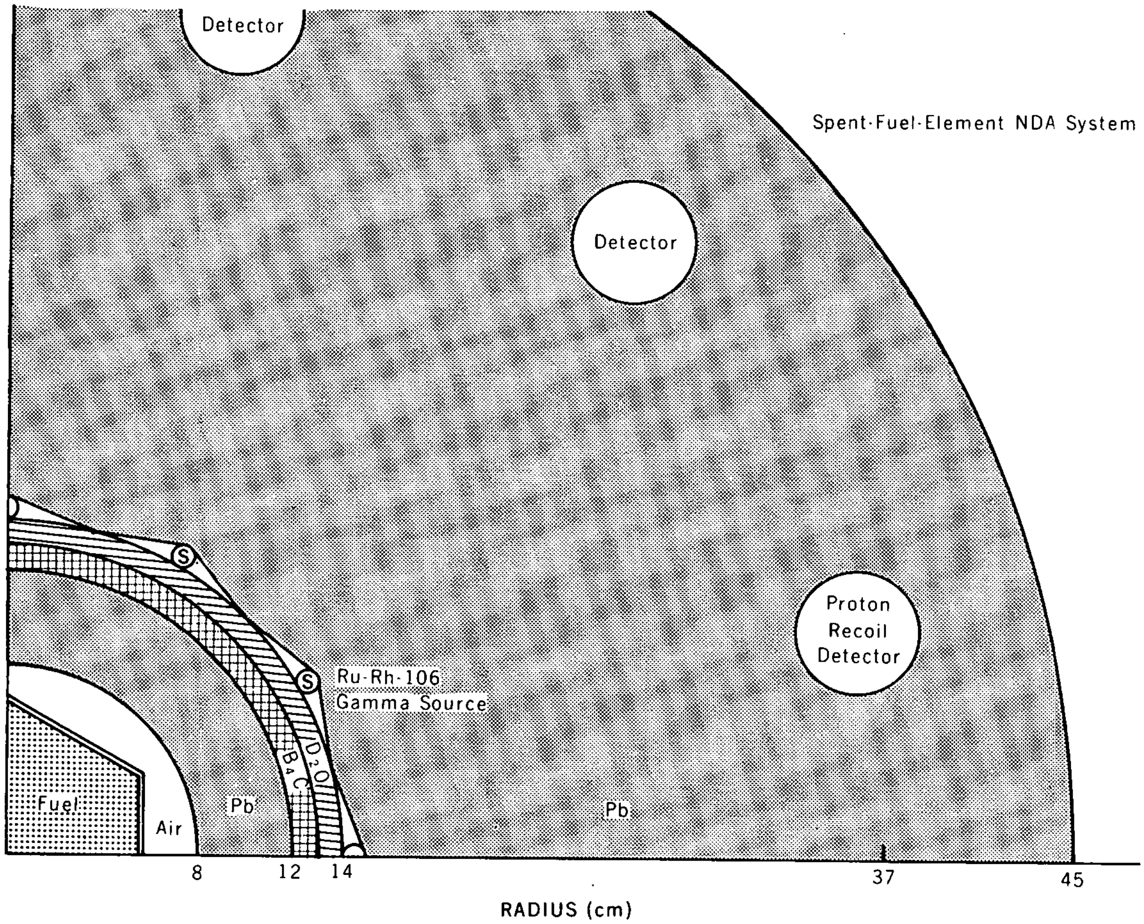

Fig. 6.26.1. Partial cross section of nondestructive assay system.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

†Instrumentation and Controls Division.

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6.27 CALCULATION OF THE FRAGMENTATION OF UO_2 BY CAPACITOR DISCHARGE: NONEQUILIBRIUM MODEL*

R. G. Alsmiller, Jr.   R. B. Perez   J. Barish†

(Abstract of ORNL/NUREG/TM-104, March, 1977)

A model based on phase-space considerations is developed to describe the fragmentation of UO_2 by capacitor discharge — i.e., to predict such quantities as the amount of gas and liquid produced, the number of liquid fragments, the number distribution of the molecules in the liquid fragments, the kinetic-energy distribution of the gas and liquid fragments, etc. The model presented here cannot give a unique numerical prediction of all of these quantities based only on the initial-state specification, but it does enable all of these quantities to be expressed in terms of the average internal energy of a gas molecule in the final state, the average binding energy of a UO_2 molecule in a liquid fragment in the final state, and the average number of molecules in a liquid fragment in the final state.

*Research sponsored by Nuclear Regulatory Commission.

†Computer Sciences Division.

6.28 WHERE ARE WE AND WHERE ARE WE GOING IN REACTOR SHIELDING

F. C. Maienschein

(Abstract of keynote address to be presented at Fifth International Conference on Reactor Shielding, April 18-22, 1977, Knoxville, Tennessee)

The field of reactor shielding shows signs of maturity and yet it retains a definite vigor. The choice between "empirical" and "transport" methods for solving shielding problems has largely been resolved in favor of the latter. Recent widespread neutron streaming problems in LWRs have illustrated rather clearly the dangers of overreliance upon "simple" methods.

Transport methods are only as good as the input nuclear data. Data files and multigroup sets are being improved and tested against integral experiments. We may note international cooperation in performing "benchmark" integral tests. Sensitivity analysis is increasingly used to define accurately additional data needs and to illustrate, for example, those neutron energy regions in sodium and iron which are of transcendent importance. Channel theory offers to illustrate with equal clarity those paths in space which determine the radiation response of interest.

Part of the needs for transport-method development have been basically met; e.g., those for one-dimensional and two-dimensional calculations (excluding streaming). Other needs, however, are barely met with minimum efficiency; e.g., streaming and other three-dimensional requirements, methods for effectively coupling calculations, and shield optimization. Sensitivity analysis falls in an intermediate category. Additional development continues but apparently at a reduced pace.

Problems remain in "balance-of-plant" shielding for LWRs, but these are not perceived to warrant the development of new methods, at least by funding agents in the U.S. More concern is expressed about the radiation levels caused by residual radioactivities which inhibit system maintenance.

For LMFBR shielding, the major problems include neutron transmission through unprecedented thicknesses of stainless steel and sodium, streaming through vessel-support regions and coolant passages for loop-type reactors, and activation of sodium in the secondary loop of pot-type reactors.

Fusion reactors appear to be inevitably quite complex with shield and blanket regions full of holes. Streaming through these holes and subsequent scattering of radiation to heat the superconducting magnets will tend to dominate the shielding problems, at least for Tokamaks. Problems of induced radioactivity will increasingly face those calculating fusion-reactor shield performance.

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

4. M. B. Emmett, "The MORSE Monte Carlo Radiation Transport Code System," ORNL-4972 (February, 1975).

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6.29 REVIEW OF METHODS FOR ANALYSIS OF NEUTRON STREAMING IN THE FAST FLUX TEST FACILITY REACTOR DESIGN*

F. R. Mynatt

(Reprint of Trans. Am. Nucl. Soc. 23, 622, 1977)

All of the important shielding problems encountered in the FFTF design were geometrically complex, and many of these problems exhibited streaming effects in which important components of the transmitted radiation were dominated by geometric attenuation.[1,2] These streaming effects present a methods paradox because their presence, when imbedded in bulk media attenuation regions, produces a complex problem, although the streaming problem when isolated is usually quite simple.

Figure 6.29.1 shows a drawing of an R-Z model of the FFTF which extends axially from the core midplane to the top of the maintenance floor and radially from the reactor center line to a depth of 30.48 cm into the concrete primary shield. The streaming zones are the reactor cavity, the reactor cavity shield zone, and the vessel support system zones. The actual models used for the latter two zones are much more detailed than shown in Fig. 6.29.1 but are not shown because of lack of space. It is useful to consider this problem as consisting of four steps:

(1) Calculation of the source entering the reactor cavity.

(2) Calculation of the reactor cavity.

(3) Calculation of the reactor cavity shield.

(4) Calculation of the vessel support system, head, and maintenance floor shield.

The actual analysis may combine some or all of these steps, but they each have different methods requirements.

Step 2 of the problem, the reactor cavity, is a large cylindrical annulus completely surrounded by steel, concrete, and sodium, and is fed from Step 1 by a source centered on the inner cylindrical surface at the midplane and having an axial height (full height at half maximum) of approximately 2 m. Experience with several alternate calculations of the reactor cavity show that it is more of a scattering cavity than a streaming path. As

ORNL DWG. 73-7953

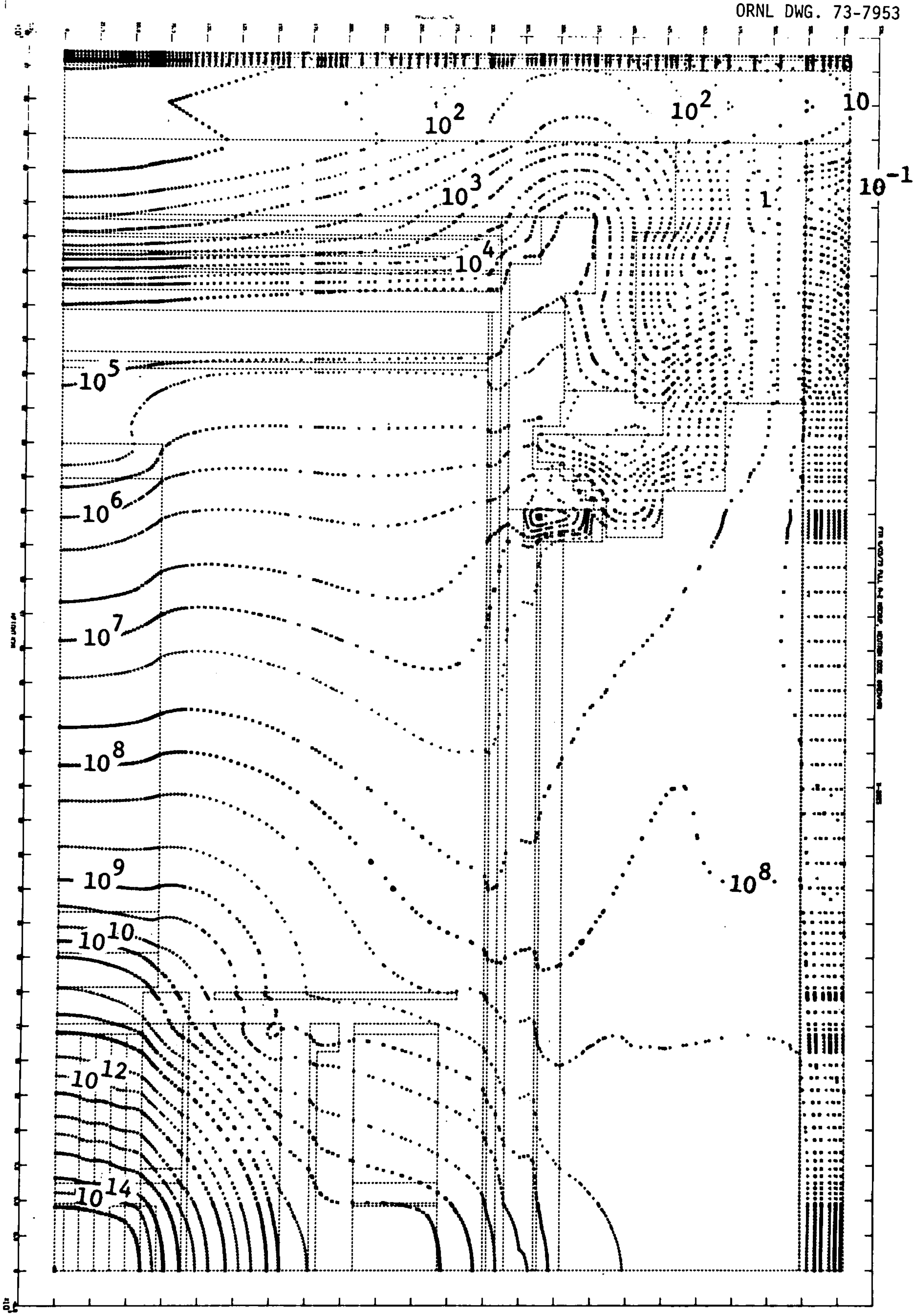

Fig. 6.29.1. Neutron dose-rate contours from step 1 of DOT r-z calculations of FFTF. Units are mrem/hr.

such, low-order discrete ordinates or albedo Monte Carlo are the most efficient methods. Because the reflection from the cavity walls significantly affect the flux levels near the surface of the vessel, the cavity problem must be incorporated in or closely coupled with the Step 1 source problem. Also, the required result from the cavity problem is an angular flux distribution for input to the next problem step. For these reasons, the cavity was included in the low-order (S_6) discrete ordinates source calculation which extended axially from 0 to 532.42 cm and radially from 0 to 555 cm. Including the cavity does not substantially increase the computing time for this problem.

Step 4 was calculated with adjoint discrete ordinates (DOT III) using a biased 166-angle quadrature to handle the streaming component. Step 3 was calculated with the three-dimensional Monte Carlo code MORSE[3,4] coupled both to the source from Step 2 and the adjoint response from Step 4 with DOMINO.[5] The three-dimensional calculation was verified in a simplified two-dimensional form with a DOT III biased 166-angle quadrature calculation. The fractional standard deviation on the MORSE calculations was typically 10% to 25% and the MORSE-DOT cases agreed within 10%.

The methods used for this problem were the most cost efficient possible. New developments such as the DOT IV code with zone and group variable quadrature and an adjoint difference coupling technique would substantially reduce the computer time needed and increase the accuracy of the solution of this problem.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

6.30 A SUMMARY OF THE ORNL SHIELD DESIGN SUPPORTING ANALYSIS FOR THE FFTF*

W. W. Engle, Jr. F. R. Mynatt Margaret B. Emmett† M. L. Williams

(Summary of paper to be presented at Fifth International Conference on Reactor Shielding, April 18-22, 1977, Knoxville, Tennessee)

From 1969 through early 1976, the Neutron Physics Division of the Oak Ridge National Laboratory provided supporting analysis for the shield design of the Fast Flux Test Facility now under construction near Richland, Washington. The lead design for the FFTF was the responsibility of Westinghouse Advanced Reactor Division (WARD). Several basic problems were apparent early in the design. Within the reactor vessel, steel and sodium provide all of the bulk attenuation between the core and the reactor head compartment. Initial calculations indicated that large errors and uncertainties existed in the cross-section data for these materials. The physical size of the system precluded a detailed description of the entire geometry for a single computer calculation. The complexity of the geometry required a verification of existing calculational techniques.

The cross-section data problem and the geometric complexity resulted in two types of experiments at the ORNL Tower Shielding Facility: the measurement of radiation transport through bulk samples (simple geometry) of important FFTF materials and the measurement of radiation transport through mockups of specific geometric regions of the FFTF. The large, complex geometry required the development of techniques for coupling both forward and adjoint discrete ordinates calculations with dissimilar mesh and angular quadrature and refinement of existing techniques for coupling three-dimensional Monte Carlo calculations with both forward and adjoint (sometimes simultaneously) discrete ordinates calculations.

The resulting data and techniques were applied to several specific problems in the FFTF. As an example, the first coupled calculations of the complete system revealed that the reactor cavity was a significant streaming path between the side of the vessel and the head compartment. The identification of this streaming path led to the realization that the source multiplication of the stored fuel inside the reactor vessel was another serious problem. As a result, a majority of the ORNL effort was expended on the analysis of various reactor cavity shield (RCS) designs. With the inclusion of the RCS and modifications to the reactor vessel support system, the dose rates above the closure head were reduced by approximately 10^4.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

†Computer Sciences Division.

6.31 APPLICATION OF AN ADVANCED SHIELDING ANALYSIS SYSTEM TO GAS COOLED FAST REACTOR DESIGNS*

D. E. Bartine L. R. Williams

(Summary of paper to be presented at Fifth International Conference on Reactor Shielding, April 18-22, 1977, Knoxville, Tennessee)

In its shielding program for General Atomic's Gas Cooled Fast Reactor,[1] ORNL has developed an advanced shielding analysis system that incorporates the latest analysis techniques for converging to a shield design compatible with other design parameters, such as those dictated by cooling and structural requirements, material compatibility, etc. Basically the analysis system consists in applying the various techniques in a logical sequence to a given design, thereby generating a large body of data to serve as an information base for a subsequent design by GA. The first step is a discrete ordinates radiation transport calculation for a two-dimensional model of the reactor system. The resulting neutron and gamma-ray fluxes are then converted to isoplots of the responses of concern (radiation damage, heating, etc.) and these are used to locate regions in the system at which those responses are higher than allowed by predetermined constraints. Next, adjoint calculations are performed for the regions of concern, and the resulting adjoint fluxes, together with the forward fluxes, are used in channel-theory calculations to determine the physical paths followed by the particles traveling from the core to those regions. In addition, the adjoint and forward fluxes are used in sensitivity calculations to determine the importance of the cross sections used in the transport calculations as functions of the shield materials and particle energies. Finally, the sensitivity results are utilized in the form of linear perturbation theory to predict the effect of changes in the shield composition and position on the various responses.

The system, using ORNL's DOT,[2] FANG,[3] and SWANLAKE[4] computer codes for the transport, channel-theory, and sensitivity calculations, respectively, has been applied to successive reference models for the GCFR. On the basis of data obtained for the first model, GA redesigned the outer radial shield and reduced its thickness by 1 ft, with a corresponding reduction in the radius of the prestressed concrete reactor vessel (PCRV). Calculations for the second model confirmed the redesign with respect to high-energy neutron fluxes and PCRV heating, but revealed problems with respect to the thermal-neutron fluxes and gamma-ray heating in the region of the lower helium channel. In addition, excessive gamma-ray heating occurred at the position of the tendon lubricant inside the PCRV wall. In all cases the problem was found to be due to the thermalization of high-energy neutrons streaming under the radial shield, and, in the case of the heating, to their subsequent capture with the concomitant emission of gamma rays. This indicated the need for extensive redesign of the lower shield region, which GA initiated upon receipt of the ORNL data. Other design-analysis-redesign iterations inherent in the analysis system — some caused by changes in basic design parameters — will continue with the expectation that they will converge upon an acceptable demonstration configuration.

*Research sponsored by ERDA Division of Nuclear Research and Applications.

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6.32 ANALYSES OF THE PRELIMINARY IN-VESSEL AND ENCLOSURE SHIELD SYSTEM DESIGNS FOR THE CLINCH RIVER BREEDER REACTOR (JULY, 1973 – JULY, 1975)*

W. W. Engle, Jr.   L. R. Williams   J. H. Swanks†   F. R. Mynatt   Lorraine S. Abbott

(Abstract of ORNL/TM-5338, March, 1976)

The first three series of calculations in ORNL's radiation shielding analysis program for the Clinch River Breeder Reactor (CRBR) are described here. The initial calculations concentrated on the neutron fluxes in the lower axial region of the reactor vessel, the results of these and subsequent calculations leading to a substantial reduction in the thickness of the lower axial shield. The second series consisted of "full-assembly" calculations for a CRBR system that closely resembled the Fast Flux Test Facility, utilizing the same vessel support system, in-vessel stored-fuel modules, and reactor cavity shields. The third series began with a CRBR design having a much-simplified vessel support system and no stored-fuel modules or reactor cavity shields, but when initial calculations yielded dose rates above the reactor head that were excessively high, a B_4C shield was reintroduced in the reactor cavity. At the same time the vessel support system was redesigned so that a portion of the support ring set on top of the concrete support ledge. A subsequent calculation for the new design with the B_4C shield showed that the dose rates were not adequately reduced, and more shielding was added in both the upper and lower sections of the support ring. With the added shielding, the dose rates were greatly reduced but were still above the criteria. A re-evaluation of the head compartment access requirements by WARD led to a relaxation of radiation criteria. The next analysis sequence will evaluate the dose levels and uncertainties based on the new criteria.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, CRBR Program.

†Operations Division.

6.33 REVIEW OF ORNL RADIATION SHIELDING ANALYSES OF THE FAST FLUX TEST FACILITY REACTOR (1975-1976)*

W. W. Engle, Jr.   Margaret B. Emmett†   Lorraine S. Abbott   F. R. Mynatt

(Abstract of ORNL-5166, ORNL-5027 Addendum, June, 1976)

This report is an addendum to an earlier document (ORNL-5027) that describes the shielding design support analyses performed by ORNL for the Fast Flux Test Facility (FFTF). Since that document was published, two additional series of calculations have been performed: one was an extension of an earlier study to determine effects on the maintenance floor dose rates of pipe and duct penetrations through a concrete shield installed in the upper region of the reactor cavity; and another was designed to determine whether radiation streaming upward through gaps in the head compartment shield would enhance the dose rates on the operating deck, which is located roughly 2.5 m above the maintenance floor. In the first series of calculations it was found that adding shielding above each of the three SISI transporter slots in the reactor cavity shield reduced the predicted dose rates at the maintenance floor level by 33% – from 2.74 ± 1.91 mrem/hr to 1.82 ± 1.27 mrem/hr. This is to be compared with a criterion of a dose rate maximum of

2 mrem/hr on the maintenance floor. In the second series of studies significant neutron streaming through gaps in the head compartment shield was observed — to the extent that the predicted maximum total dose rate at the operating deck level is 1.9 ± 1.6 mrem/hr.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

6.34 TRANSPORT CALCULATIONS AND SENSITIVITY ANALYSES FOR AIR-OVER-GROUND AND AIR-OVER-SEAWATER WEAPONS ENVIRONMENTS*

J. V. Pace, III† D. E. Bartine F. R. Mynatt

(Summary of paper presented at Symposium on Vulnerability and Survivability of Aerial and Surface Targets, October 26-28, 1976, Silver Spring, Maryland)

Two-dimensional neutron and secondary gamma-ray transport calculations and cross-section sensitivity analyses have been performed to determine the effects of varying source heights and cross sections on calculated doses. The air-over-ground calculations demonstrate the existence of an optimal height of burst for a specific ground range. They also indicate under what conditions air-over-ground calculations are conservative with respect to infinite air calculations.

The air-over-seawater calculations showed that the seawater both reduced the neutron flux and softened the neutron spectrum to a greater extent than the ground. But while the neutron dose was decreased, the gamma-ray dose was enhanced. Further investigation revealed that not only did the hydrogen content increase the gamma-ray dose, but also it slowed the neutrons to be captured in the trace element chlorine which produced large numbers of gamma rays with energies primarily in the range 6-8 MeV.

The air-over-ground sensitivity calculations gave a myriad of results and vital information. The sensitivity predictions for the ground indicate that not only could the ground thickness be decreased for calculational purposes, but also that hydrogen content was very important, thereby making the amount of water in the ground play a major role.

Additional information showed whether P_1 or P_2 cross sections were necessary to keep the results within one percent of those with P_3 cross sections. The sensitivity analysis also indicated the system height required for problems with defined source heights and ground ranges, and the effect of the degree of Legendre angular expansion of the scattering cross sections (P_ℓ) on the calculated dose.

The results from the above calculations have added tremendously to the already available knowledge of two-dimensional weapons effects calculations and should enable the users of 2-D transport codes to show a savings in time and money on future work.

*Research sponsored by Defense Nuclear Agency.

†Computer Sciences Division.

6.35 EVALUATION OF INITIAL LOADING COUNTING RATE DATA FROM EXPERIMENTS WITH THE MOCK-UP CORE FOR THE FAST FLUX TEST FACILITY*

J. T. Mihalczo† G. C. Tillett‡ D. Selby

(Abstract of ORNL/TM-5106, March, 1976, and of Nucl. Tech. 30, 422, 1976)

The modified source multiplication method was used to determine the reactivity from the count rate data as fuel assemblies were removed from the engineering mockup core for the FFTF. The count rate was monitored with a fission detector in the center of the core and in each of the three shield lobes (simulating the low-level flux monitor) as the ZPR-9 assembly was unloaded to simulate, in a reverse manner, the proposed initial loading to critical for the FFTF. Some conclusions from this interpretation are: (1) The inverse count rate from a fission counter in the center of the core is an excellent way to monitor the initial loading of the reactor. (2) The inverse count rates from each of the low-level flux monitors are not adequate for monitoring the initial loading since they were not a smooth function of the number of fuel assemblies loaded even after correction for changes in detection efficiency. (3) The reactivity versus fuel loading (obtained from the interpretation of the inverse kinetics rod-drop calibration at 0.8 dollar subcritical) was not a smooth function of the fuel loading because of difficulties in calculating the required changes in detection efficiency for detectors in the shield; however, a similar interpretation for the in-core detector showed a smooth dependence of reactivity on fuel loading. (4) The reference asymmetric loading pattern for startup does not present any interpretational difficulties with a detector in the core, and, thus, the symmetric loading pattern has no real advantages with an in-core detector and requires more time. (5) The initial startup of the FFTF should be monitored with an in-core detector. These conclusions are consistent with those obtained at the PFR in the United Kingdom, with an in-core detector, and at the Phenix reactor in France, with a detector outside the core.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, CRBR Program.

†Instrumentation and Controls Division.

‡Energy Research and Development Administration, Washington, D.C.

7. INFORMATION ANALYSIS AND DISTRIBUTION

7.1 CENTRAL COMPUTERIZED DATA BASE FOR LIQUID METAL FAST BREEDER REACTOR SAFETY CODES*

H. Alter† G. F. Flanagan N. M. Greene‡

(Summary of paper presented at International Conference on Fast Reactor Safety and Reactor Physics, October 5-8, 1976, Chicago, Illinois, and at Fifth Biennial International CODATA Conference, June 28-July 1, 1976, Boulder, Colorado)

Introduction. Since the middle sixties, the area of LMFBR safety analysis in the United States has expanded considerably. Starting with a few Bethe-Tait disassembly codes such as MARS[1] in 1967, the safety analysis field today includes well over 80 codes. These range from codes which model detailed phenomenological events such as DEFORM[2] to production-oriented codes such as VENUS.[3] As the codes become more detailed and diverse, the need for a wide variety of data increases. These data come from several sources such as universities, industry, the Department of Defense, the National Aeronautics and Space Administration, the Energy Research and Development Administration (ERDA), the Nuclear Regulatory Commission (NRC), the National Bureau of Standards, and others. Many times the search for and evaluation of data from such diverse organizations is time consuming and requires subjective conclusions on the part of the safety analyst.

For the Fast Flux Test Facility Preliminary Safety Analysis Report (PSAR) and the Final Safety Analysis Report (FSAR), the safety analyses were centered at a few national laboratories or private vendors. Even among these few organizations, the data considered "best" for a particular situation differed significantly, especially on equations of state and high-temperature properties. The problem intensified with the safety analysis for the Clinch River Breeder Reactor PSAR where several more organizations became involved. It is expected that the Prototypic Large Breeder Reactor projects will experience the same problem. These problems were brought into perspective by Dr. H. Bethe, who, while reviewing ERDA's safety program, emphasized the lack of a common data base for safety analysis. Such a data base should provide a consistent, efficient and referenceable source of data for LMFBR safety analysis.

SACRD. In the spring of 1975, Oak Ridge National Laboratory (ORNL) was asked by ERDA to define and develop this data base. It was to be computerized and user oriented so as to take advantage of the sophisticated data retrieval methods currently available. It has been named the SACRD (Safety Analysis Computerized Reactor Data) base.

SACRD will gather, evaluate, and distribute the necessary non-design-related data to perform safety analysis for all categories from initiating events through dose analysis and post-accident heat removal. The data will be structured by type, property, and material. It will be available to the user in tabular, parametric, and graphical forms.

In addition to evaluated basic data, SACRD will contain bibliographic and reference materials, as well as files for commonly used correlations and variables. Benchmarks for data testing will be specified for the user. In addition, there will be a file containing new but nonevaluated data.

The evaluated data will be coordinated by the Safety Analysis Data Coordinating Group (SADCG), which is made up of experimentalists and analysts from industry, universities, national laboratories, and government agencies. SADCG currently has six data evaluation subcommittees covering the following areas: Fuel Mechanical Properties, Structural Mechanical Properties, Aerosol Transport, Thermo-Physical Properties, Neutronics, and Radiological Dose Data. The members of the evaluation subcommittees will be responsible for creating, collecting, and evaluating the data in SACRD. In addition to the evaluations groups, there is a subcommittee to oversee computer code interfaces, terminal interactive systems, and formats.

In its early stages, SACRD will distribute the data in report and computer printout form, and later by interactive terminal display, and possibly by prepared code data input packages.

Data Management. The data management in SACRD is handled by the JOSHUA system[4] developed by Savannah River Laboratories. JOSHUA is a modular code system consisting of some 150 separate computer codes linked together by an Operating System providing data management, terminal and job execution facilities.

The data management schemes are accessible through simple FORTRAN statements which relieve the computer code developer of file searching. All data under the management schemes are catalogued in directories which are automatically scanned when data is requested. Random access input/output is employed throughout. Each record is named by a string of alphameric qualifiers which are generally chosen to describe meaningful attributes of the data. For example, an abstract record in SACRD would be named with the following qualifiers:

FILE.PROPERTY.MATERIAL.VERSION.ABSTRACT

where the periods act as delimiters, FILE is the file name, PROPERTY is the physical property name, MATERIAL is the material name, VERSION is the version of the evaluation, and ABSTRACT names the record type. A request for the abstract of the version 1 heat transfer (HT) property thermal conductivity (K) data of $^{238}U_{92}$ (U238) might then be written:

HT.K.U238.1.ABSTRACT

These simple, meaningful names serve to identify the data when it is read or written; the user never concerns himself with the actual location of the data.

Special record structures have been defined for SACRD. These include provisions to allow specifying data in matrix form to arbitrary order. Attention has been taken to ensure all data is completely identified by abstracts, names, units (SI), creation dates, etc.

Conclusions. The first version of SACRD is scheduled for completion by December 1977. However, partially evaluated data will be distributed to selected safety analysts in late 1976.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

†U.S. Energy Research and Development Administration, Washington, DC.

‡Computer Sciences Division.

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7.2 THE SACRD DATA BASE AT OAK RIDGE NATIONAL LABORATORY*

F. B. Sadler† N. M. Greene‡

(Summary of paper to be presented at Association for Computer Machinery Southeastern Regional Meeting, April 18-20, 1977, Biloxi, Mississippi)

Oak Ridge National Laboratory is presently designing a data base system for the Energy Research and Development Administration (ERDA) which contains all of the "handbook" type information needed to perform safety analyses on liquid metal fast breeder reactors (LMFBR). These data span such diverse areas as physical properties data (thermodynamics, strength of materials, heat transfer, mass transport), meteorological data, chemical data, dosimetry data, neutronics data, and aerosol transport data. The data base is required to accommodate the needs of safety analysts who must keep pace with the LMFBR program which is anticipated to grow rapidly during the next two decades. A central data base will help ensure consistency of results between various organizations and will serve to improve the state of much of the already evaluated data. When completed in 1979, the SACRD (Safety Analysis Computerized Reactor Data) system will allow a safety analyst rapid access to its data via remote time-sharing terminals.

The Data Base Management System (DBMS) chosen for the SACRD data base was a system developed at Savannah River Laboratory named JOSHUA. This system, although of the "home grown" variety, has many of the more sophisticated features incorporated in the full-scale DBMSs available on today's market. These include an efficient hierarchial data manager with record level security, a terminal monitor for time-sharing applications, and a pre-compiler for batch FORTRAN applications. However, unlike these general-purpose data-management systems, the JOSHUA system supports a relatively fixed data structure. A fixed structure can have either a good or bad overall effect on the efficiency of the implemented data base, depending on whether or not the data fit naturally into its structure. In our case, the mainly scientific type data of SACRD fit very nicely into JOSHUA's pre-defined structure; whereas, if a purchased DBMS were used, a small amount of retrieval efficiency may have had to be sacrificed.

The paper, which this writing summarizes, will give some more background of the JOSHUA data manager and the type of data for which it was designed, followed by its comparison with the general purpose type DBMS.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, LMFBR Program.

†Computer Sciences Division.

7.3 THE INFORMATION CENTER AS A TECHNICAL INSTITUTE UNIFYING A USER COMMUNITY*

Betty F. Maskewitz Betty McGill

(Excerpts from a paper presented at American Society for Information Science Annual Conference, October 4-9, 1976, San Francisco, California)

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The information analysis center concept was articulated by a select body of scientists called together by the President of the United States in 1962. Chaired by A. M. Weinberg, Oak Ridge National Laboratory (ORNL), the committee reported[1] that "science must undergo a social reorganization to enable it to remain unified even though it continues to grow ... The beginnings and the shape of this reorganization can be discerned in the emergence of scientists who are primarily handlers of information — who sift, retrieve, and analyze the information created by others and who, in so doing, synthesize new information based upon the individual findings of others."

. . . .

ORNL led in the pioneering effort to make the envisioned concept come alive,[2] and its Radiation Shielding Information Center (RSIC) is a viable example of this effort.[3] ... Functioning within the Neutron Physics Division, RSIC has long been engaged in shielding and radiation transport research. It is a technical institute serving, since early 1963, the international scientific community engaged in research and development for the design of shields that provide protection from biological and physical damage due to ionizing radiation from nuclear reactors, radioisotopes, nuclear weapons, accelerators, etc. The Center's personnel collect, organize, evaluate, package, and disseminate radiation protection and shielding information to anyone whose work requires such information. In general, all information concerned with the transport of ionizing radiation is covered, with partial coverage of the areas of radiation instrumentation and neutron thermalization.

News of RSIC material is distributed currently to more than 1500 persons, of whom about 24% are located in countries other than the United States. The RSIC Newsletter is mailed to individuals in 39 different nations of the world. More than 3080 separate letters of request were received in FY 75, requiring more than 6268 separate activities to satisfy the requests. In return, we ask for information and the cooperation of the international shielding scientist in making the information analysis center an efficient technical institute.

RSIC also treats the complex computer codes and associated data libraries required for radiation transport calculations as an inseparable part of shielding information... Computer code and data exchange has had a tremendous impact on shielding technology.[4]

By the nature of its functions and the manner in which it implements them, RSIC contributes substantially to the unification of the community engaged in the various areas of radiation transport and shielding. The dissemination of information to otherwise unrelated users is in itself a unifying activity. RSIC, however, promotes a closer relationship between the Center and user. Communication channels between the Center and the worker in the field are kept open and extensively used. Many RSIC projects have benefitted by the active participation of the Center's users. Seminar-workshops, topical meetings, policy advisory committees, review articles, and data collection and management activities point to an industry-wide cooperative enterprise. The RSIC forum is both an information exchange medium and a focal point for the initiation and organization of new activities and the identification of new leadership in the field.

Early voluntary informal efforts at providing guidelines for good documentation of computer programs, and for programming practices to facilitate exchange has evolved into a formal standards effort under the auspices of the American Nuclear Society (ANS). Two standards have been published;[5,6] a third has become an ANSI standard.[7] An RSIC staff member has been a motivating force behind and a participating member in the effort since the beginning.

RSIC is also involved in work on shielding standards. A staff member is chairman of the ANS Standards Subcommittee on Shielding, ANS-6, whose goals are to establish standards in connection with radiation shields, to provide shielding information to other standards groups, and to prepare recommended sets of shielding data and test problems.

We can claim some credit for the advancement of the state-of-the-art in shielding calculations. Our insistence on documenting code development and our assistance given to the code developer while he is documenting his work is well known. Our encouragement for the use of higher-level programming languages to facilitate exchange, and for the use of standards in programming practices, is also well known. We are quick to refute the claiming of proprietary interests which would withhold programming efforts from free exchange. In shielding, this trend has been kept to a minimum. You will find interesting code development from the private sector in the RSIC collection.

In addition to providing a center with the total information spectrum, RSIC handling of computer codes as an essential part of shielding information has additional benefits. We can be of greater assistance to the shielding research man. We supply the bibliography with abstracts for his prior study; we suggest calculational methods to solve his particular problem which requires the use of the computer; we help him select the computer program best suited to his problem and to his computer environment, and we give him assistance in making it operable at his installation. We spend a great deal of time trouble-shooting while a customer is seeking a solution to his problem. And, when we review the report of his research at a later time, we have more insight as to how he got his results.

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We also make a difference to the code developer. If his work is of interest to a large number of people, we save him time, effort, and expense. He prepares his code material once for us, gives us a workshop, and we then handle the requests. He "freezes" his work at a given level to place it in the public domain and goes full speed ahead exploiting and developing his own work; when he reaches a new plateau, he freezes again and updates the current code package. Many times, contributors have requested that we send to them their code packages of a certain level, because in their own advance code development they took a bad turn and want to begin again from the earlier point.

Maintaining close relations with the American Nuclear Society (ANS) has also been a means of unifying and serving the shielding community. Active participation and leadership in the ANS Shielding and Dosimetry Division by members of the RSIC staff has served to increase the usefulness of both the Center and the Society.

We have previously stated that the Center's cooperative activities have not been limited to the United States. Shielding information travels in all directions across the oceans. Various exchange agreements exist between nations. The United States Energy Research and Development Administration (ERDA) and the Organization for Economic Cooperation Development (OECD) Nuclear Energy Agency (NEA) have exchanged computer programs and cross-section data since 1963. Their agreement has been extended to third parties under defined circumstances. International conferences on reactor shielding provide new impetus to fruitful exchange and to further unite the shielding community across all national boundaries. A fifth such conference is now being planned. We also collaborate with EURATOM's European Shielding Information Service (ESIS) in such international information exchange.

. . . .

RSIC will continue to vigorously pursue its mission within the context of flourishing international cooperation and to value highly the deepening understanding between nations engaged in common research as being beneficial to all society.

. . . .

*RSIC is sponsored by ERDA Division of Reactor Development and Demonstration, ERDA Division of Magnetic Fusion Energy, the Nuclear Regulatory Commission, and the Defense Nuclear Agency.

1. President's Science Advisory Committee, *Science, Government, and Information: The Responsibilities of the Technical Community and the Government in the Transfer of Information,* The White House, 1963.

2. Francois Kertesz, "The Information Center Concept," Nucl. Eng. Design 9, 383-391, (1969).

3. D. K. Trubey, "The Radiation Shielding Information Center — A Technical Information Service for Nuclear Engineers," Nucl. Eng. Design 9, 392-5 (1969).

4. See paper 7.7.

5. *ANS Standard — A Code of Good Practices for the Documentation of Digital Computer Programs,* prepared by Subcommittee 10, ANS Standards Committee, ANS-STD-2-1967 (approved December 5, 1967).

6. *ANS Standard — Recommended Programming Practices to Facilitate the Interchange of Digital Computer Programs,* prepared by Subcommittee 10, ANS Standards Committee, ANS-STD-3-1971 (approved April 1, 1971).

7. *American National Standard — Guidelines for the Documentation of Digital Computer Programs,* prepared by the American Nuclear Society Standards Committee ANS-10, ANSI N413-1974 (approved June 20, 1974).

7.4 THE RADIATION SHIELDING INFORMATION CENTER — AN INTERNATIONAL TECHNOLOGY RESOURCE*

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

(Summary of paper to be presented at American Nuclear Society Annual Meeting, June 12-17, 1977, New York City)

The Radiation Shielding Information Center (RSIC)[1,2] is a technical institute serving the international 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, (2) collecting, checking out, packaging, and distributing large computer codes[3-5] and evaluated and processed data libraries.[6] The data packages include multigroup coupled neutron-gamma-ray cross sections and kerma coefficients, other nuclear data, and radiation transport benchmark problem results.

The fundamental rational underlying the code and data activities is the "open code or data package"--open to improvement and to technical scrutiny (see Fig. 7.4.1).

The radiation treated by the majority of the codes is either neutron or gamma radiation or both, but some codes treat charged particles. The types of geometry treated vary widely, with many codes allowing a general three-dimensional geometry.

For convenience, the available codes are grouped into two classes: (1) those treating radiation transport and identified by Computer Code Collection (CCC) numbers, and (2) those performing auxiliary data processing useful for other radiation analysis purposes and identified by Peripheral Shielding Routine (PSR) numbers. The numerical methods applied in the CCC codes are primarily discrete ordinates (one and two dimensions), Monte Carlo (up to three dimensions), and kernel integration (generally three dimensions). The remaining include removal-diffusion (Spinney), moments, spherical harmonics, and invariant embedding transport theory codes, plus miscellaneous codes to calculate fission product buildup, release and resulting dose; stopping power; optimization; sensitivity analysis; and others.

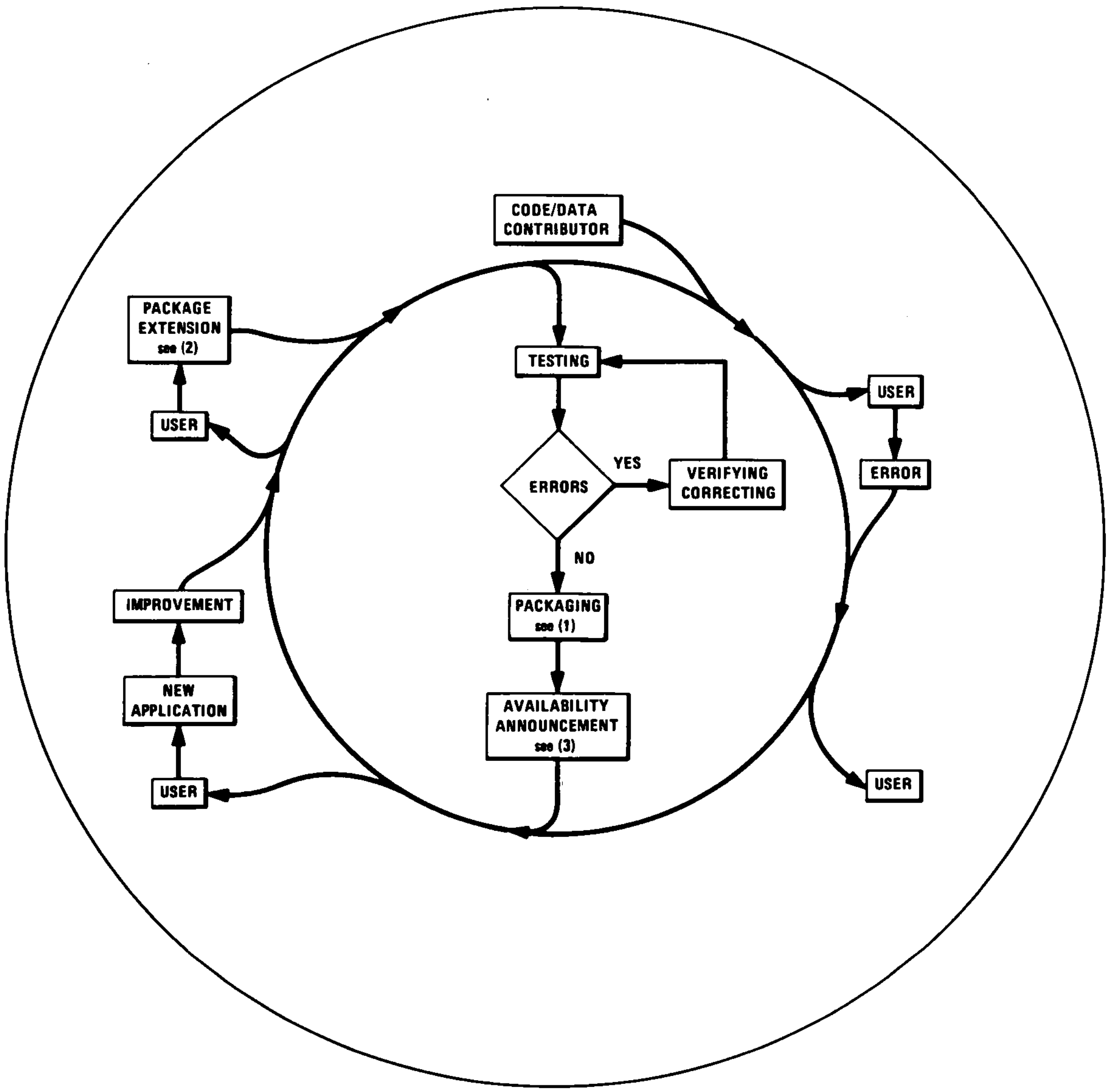
The PSR codes include codes for multigroup cross-section generation and handling, prediction of cross sections with nuclear models, experimental energy spectra unfolding, optimization, plotting, coupling discrete ordinates and Monte Carlo results, random number generation, and other functions.

In addition, RSIC is involved in many data activities, with most emphasis being placed on nuclear cross-section data. Through cooperation with various agencies, RSIC assists in improving the adequacy of basic evaluated cross-section data and packages and distributes various types of data libraries useful in radiation transport analysis. The emphasis of the effort is on the improvement of calculational tools available to the shielding analyst.

The Center assists in the acquisition, checkout, and review of "shielding" cross sections in ENDF format which may ultimately be placed in the ENDF/B file. RSIC also maintains and distributes the Defense Nuclear Agency (DNA) evaluated cross-section library. This is a working library in ENDF format whose content can be modified and revised as often as the evaluator deems such changes to be necessary. A similar function has also been established at RSIC on behalf of ERDA's Division of Magnetic Fusion Energy (DMFE). Evaluated data from both the DNA and DMFE libraries are submitted for inclusion in the latest version of ENDF/B.

For workaday problem solving, it has been found useful to generate and collect multigroup cross-section sets, and to package, document, and distribute them in a format suitable as input to the most-used computer codes. Each data set, packaged as a unit, carries a Data Library Collection (DLC) number. As with the code packages, a particular data package does not remain static but is subject to revision, updating, and expansion as required. Such changes are announced in the RSIC Newsletter. Other data in the collection include gamma-ray interaction cross sections, neutron-induced gamma-ray production spectra, fluence-to-kerma coefficients, radioactive decay spectra and decay schemes, and detailed output from transport calculations.

ORNL-DWG 76-18870



(1) Packaging – Auxiliary routines, data libraries, and sample problem are collected and written onto magnetic tape (the master tape) – the code is run on local computers. Sample problem output is captured and entered as a separate file on the master tape. The tape is scanned, and described on a tape list which accompanies each package (see example attached). Documentation is added to complete the package.

(2) A package is extended to include versions which have been converted to run on hardware different from that for which the code was designed – or is converted to a language different than that which the code was developed.

(3) Availability is announced through the RSIC Newsletter, which is currently being mailed to more than 1600 persons, 24% of whom are in countries outside the U.S. (39 different nations).

Fig. 7.4.1. RSIC code/data package.

A continuing project, in cooperation with the American Nuclear Society, is to collect, edit, and publish reference data in the form of "benchmark problems."[7] The objective 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 analysis.

Lists of selected codes and data packages are given in Table 7.4.1. These are typical of the most used but not necessarily the best for any particular application. Further information on these or any other codes or data are available from RSIC upon request.

*RSIC is sponsored by ERDA Division of Reactor Development and Demonstration, ERDA Division of Magnetic Fusion Energy, the Nuclear Regulatory Commission, and the Defense Nuclear Agency.

Table 7.4.1 Typical code and data packages

1. Neutron/Gamma-ray Transport Discrete Ordinates Multigroup	1 Dimension	CCC-42/DTF-IV CCC-82/ANISN CCC-126/ASOP (optimization) CCC-130/DTF-69 (X-ray) CCC-204/SWANLAKE (sensitivity analysis) CCC-235/INAP (activation)
	2 Dimensions	CCC-222/TWOTRAN II CCC-230/TRIPLET (triangular mesh) CCC-276/DOT 3.5
Monte Carlo	3 Dimensions	CCC-203/MORSE-CG (multigroup) CCC-187/SAM-CE
2. Multigroup Cross Section Processors	Neutron	PSR-13/SUPERTOG PSR-52/MACK (kerma factors)
	Gamma Ray	PSR-51/SMUG
	Coupled	PSR-63/AMPX
3. Kernel Integration	3 Dimensions	CCC-48/QAD CCC-94/KAP VI CCC-213/ACRA (radioactive cloud)
4. Spectra Unfolding		PSR-17/FERDOR-COOLC PSR-41/MAZE CCC-112/SAND II CCC-233/CRYSTAL BALL
5. Fission Product Inventory		CCC-217/ORIGEN CCC-225/REST CCC-237/BURP 2
6. Multigroup Data Libraries	Neutron	DLC-2/100G (100-group from ENDF/B) DLC-33/MONTAGE (100 group activities)
	Coupled	DLC-27/AMPX01 (104-n,22-γ) DLC-31/FEWG1 (37-n,21-γ) DLC-37/EPR (100-n,21-γ)

1. D. K. Trubey, "The Radiation Shielding Information Center--A Technical Information Service for Nuclear Engineers," Nucl. Eng. Design 9, 392-5 (1969).

2. B. F. Maskewitz, D. K. Trubey, R. W. Roussin, F. H. Clark, "The Radiation Shielding Information Center: A Unifying Force in the International Shielding Community," Fourth International Conference on Reactor Shielding, Paris, France, October, 1972, CONF-721018, Vol. 1, pp. 215-225.

3. D. K. Trubey and Betty F. Maskewitz, "Computer Codes for Shielding Calculations--1969," Nucl. Eng. Design 10, 505-17 (1969).

4. Betty F. Maskewitz, and D. K. Trubey, "Computer Codes for Sheidling Calculations--1970," Nucl. Eng. Design 13, 448-9 (1970).

5. Betty F. Maskewitz, Francis H. Clark, and D. K. Trubey, "Computer Codes for Shielding and Related Calculations--1972," Nucl. Eng. Design 22, 334-341 (1972).

6. R. W. Roussin, D. K. Trubey, and Betty F. Maskewitz, "Data Activities of the Radiation Shielding Information Center," Fourth International Conference on Reactor Shielding, Paris, France, October, 1972, CONF-721018, Vol. 4, pp. 1191-1201.

7. A. E. Profio, Editor, "Shielding Benchmark Problems," ORNL-RSIC-25 (ANS-SD-9) (Sup. 1, 1970, Sup. 2, 1975).

7.5 RADIATION TRANSPORT AND SHIELDING INFORMATION, COMPUTER CODES, AND NUCLEAR DATA FOR USE IN CTR NEUTRONICS RESEARCH AND DEVELOPMENT*

R. T. Santoro   B. F. Maskewitz   R. W. Roussin   D. K. Trubey

(Abstract of paper presented at 9th Symposium on Fusion Technology, June 14-18, 1976, Garmisch-Partonkirchen, Federal Republic of Germany)

The activities of the Radiation Shielding Information Center (RSIC) of the Oak Ridge National Laboratory are being utilized in support of fusion reactor technology. The major activities of RSIC include the operation of a computer-based information storage and retrieval system, the collection, packaging, and distribution of large computer codes, and the compilation and dissemination of processed and evaluated data libraries, with particular emphasis on neutron and gamma-ray cross-section data. The Center has acquired 13 years of experience in serving fission reactor, weapons, and accelerator shielding research communities, and the extension of its technical base to fusion reactor research represents a logical progression. RSIC is currently working with fusion reactor researchers and contractors in computer code development to provide tested radiation transport and shielding codes and data library packages. Of significant interest to the CTR community are the 100 energy group neutron and 21 energy group gamma-ray coupled cross-section data package (DLC-37) for neutronics studies, a comprehensive 171 energy group neutron and 36 energy group gamma-ray coupled cross-section data base with retrieval programs, including resonance self-shielding, that are tailored to CTR application, and a data base for the generation of energy-dependent atomic displacement and gas production cross sections and heavy-particle-recoil spectra for estimating radiation damage to CTR structural components. Since 1964, the Center has been involved in the international exchange of information, encouraged and supported by both government and interagency agreements; and to achieve an equally viable and successful program in fusion research, the reciprocal exchange of CTR data and computing technology is encouraged and welcomed.

*Research sponsored by ERDA Division of Magnetic Fusion Energy.

7.6 RSIC AFTER 14 YEARS--WHERE DO WE GO FROM HERE?*

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

(Abstract of paper to be presented at Fifth International Conference on Reactor Shielding, April 18-22, 1977, Knoxville)

During the fourteen years since the Radiation Shielding Information Center (RSIC) started operations, many changes have taken place in the nuclear industry, both government and commercial, and in shielding technology. Military and space applications of radiation transport technology have waxed and waned, but computing applications and the need for elaborate computer codes and nuclear data for the commercial nuclear power industry have increased manyfold.

The rationale for RSIC operations has been the recognition of the need for government to promote the exchange and transfer of shielding technology to advance the state of the art. With the present need to meet energy demands through nuclear power, and concurrently to minimize radiation exposure to operating personnel and the public, there is great need to increase radiation analysis capabilities among the utilities and other segments of the industry. Yet, because of government cost-recovery restrictions and Electric Power Research Institute (EPRI) precedents, there is a widening gap between what is being done and what is needed.

During 1975, RSIC had a contract with EPRI (Electric Power Research Institute) to integrate shielding and technology information exchange of the nuclear power industry with that of the government and its contractors. Included in the work performed by RSIC on behalf of EPRI was a survey of the radiation protection, radiation transport, and shielding information needs of the nuclear power industry. Such needs include computing technology, nuclear data, special information needs, such as fission product and "crud" transport, and expansion more generally into radiation protection areas such as environmental radiological analysis.

*RSIC is sponsored by ERDA Division of Reactor Development and Demonstration, ERDA Division of Magnetic Fusion Energy, the Nuclear Regulatory Commission, and the Defense Nuclear Agency.

7.7 ABSTRACTS OF DIGITAL COMPUTER CODE PACKAGES ASSEMBLED BY THE RADIATION SHIELDING INFORMATION CENTER (VOL. IV)*

Betty McGill Betty F. Maskewitz C. Marie Anthony
Hemma E. Comolander Henrietta R. Hendrickson

(Abstract of ORNL-RSIC-13, Vol. IV, January, 1976)

ORNL/RSIC-13, Volumes I-IV, is a loose-leaf report series containing the abstracts of digital computer code packages assembled by the Radiation Shielding Information Center. Volume IV includes abstracts for code packages CCC-169 through CCC-263. The purpose of the abstracts is to give to a potential user several criteria for deciding whether or not he wishes to request the code package. In general, each abstract describes the nature of the problem solved and the method of solution, cites any restrictions or limitations, describes computer hardware and software required for local implementation, and cites references and gives credit for the code development and contribution to the center.

The report is indexed by code package name and by code package number for the abstracts published in Volume IV only, and by keywords and key phrases for the entire computer code collection, CCC-1 through CCC-269.

*Research sponsored by ERDA Division of Reactor Development and Demonstration, ERDA Division of Magnetic Fusion Energy, the Nuclear Regulatory Commission, and the Defense Nuclear Agency.

7.8 SURVEY OF RADIATION PROTECTION, RADIATION TRANSPORT, AND SHIELDING INFORMATION NEEDS OF THE NUCLEAR POWER INDUSTRY*

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

(Abstract of EPRI-NP-155, April, 1976)

The report documents the results of a survey made by the Radiation Shielding Information Center (RSIC) of the radiation protection, radiation transport and shielding information needs of the nuclear power industry. Eighty-three installations were covered, including public and private utilities, architectural-engineering and other consulting firms and individuals who serve them, and vendors. A general summary provides the information and technology environment of each responding installation for a better understanding of the needs expressed. This is followed by separate sections which delineate, respectively, the computing technology needed, nuclear data needs, suggestions for RSIC operations, projects, and services to fill expressed needs, and references for the technology cited in the report.

The analysis revealed several distinct trends:

1. There is a need for well-defined benchmark calculations relevant to the utility industry.

2. New and improved data libraries are needed. Included are evaluated and processed cross sections, radioactive decay nuclide data, irradiated fuel data, albedo data, etc.

3. New and improved code development. Many want an accurate and fast code to perform neutron streaming studies. Almost everyone wants <u>fast</u> codes, sometimes accurate! Several want a fast, accurate labyrinth code. All those using RSIC-packaged codes want better documentation and more sample problems.

4. General trends emerged. Several respondents want radiation effects information and ask for the revival of the Radiation Effects Information Center (REIC), now defunct. Many are comfortable in their working relationships with RSIC and some suggest that RSIC organize industry-wide working groups to see to their own needs cooperatively. Several feel that RSIC has no bounds and suggest a myriad of new projects.

*Research sponsored by Electric Power Research Institute.

7.9 A REVIEW OF RADIATION ENERGY SPECTRA UNFOLDING: PROCEEDINGS OF A SEMINAR-WORKSHOP, APRIL 12-13, 1976*

D. K. Trubey, Compiler

(Abstract of ORNL/RSIC-40, October, 1976)

On April 12-13, 1976, the Radiation Shielding Information Center of Oak Ridge National Laboratory convened a seminar-workshop on the subject of unfolding radiation energy spectra. More than 20 papers were presented. They describe theoretical approaches and practical experience in determining neutron and gamma-ray spectra from responses of detectors, such as NE-213 and NaI scintillators and neutron activation detectors. Taken as a whole, they represent a description of the state of the art.

*Seminar-workshop jointly sponsored by ERDA's Division of Reactor Development and Demon stration and by the Defense Nuclear Agency.

7.10 RADIATION PROTECTION AND SHIELDING STANDARDS*

E. J. Vallario† D. K. Trubey

(Abstract of paper to be presented at Fifth International Conference on Reactor Shielding, April 18-22, 1977, Knoxville, Tennessee)

This presentation discusses radiation protection and shielding standards from the overall management viewpoint, considering the development mode and some of the problems and issues.

How Are Standards Developed? Hundreds of organizations are developing nuclear and non-nuclear standards in the United States, of which only a small percentage are concerned with the development of radiation protection and shielding standards. These include the American Nuclear Society, Health Physics Society, Industrial Hygiene Association and others. The radiation protection and shielding standards development system is described, including professional society involvement with the management consensus system of the American National Standards Institute. In the latter case, the most recent organizational changes will be noted.

Enhancing the Standards Management System. How may the standards management process be enhanced? In this context the key areas to be discussed are:

1. Defining the role of the professional societies to ensure maximum utilization of expertise.
2. Encouraging greater society involvement.
3. Upgrading the "need" system.
4. Time reduction of the standards process.
5. Interface with regulatory agencies.
6. Publication of Society vs ANSI standards.

Some Unresolved Questions. Several questions continue to arise which will be discussed. These are:

1. Should fundamental data be the subject of a standard?
2. Is a tutorial rendition appropriate in a standard?
3. How do you treat standard reference data?
4. Should a standard be in the short form?

Priority Needs. A priority need is discussed — radiation protection standards applicable to the design of nuclear facilities.

*Research sponsored by Energy Research and Development Administration.

†Energy Research and Development Administration.

7.11 BCTIC — THE BIOMEDICAL COMPUTING TECHNOLOGY INFORMATION CENTER*

Betty F. Maskewitz R. L. Henne† W. J. McClain†

(Note: The following summary is typical of papers presented at the Sixth Symposium on the Sharing of Computer Programs and Technology in Nuclear Medicine, January 26, 1976, Atlanta; the Southeast Region MUMPS Users' Group Meeting, May 28, 1976, Atlanta; the Society of Nuclear Medicine 23rd Annual Meeting, June 8-11, 1976, Dallas; the Association for Computing Machinery Annual Conference, October 20-22, 1976, Houston; the International Symposium on Medical Radionuclide Imaging, October 25-29, 1976, Los Angeles; the 29th Annual Conference on Engineering in Medicine and Biology, November 6-10, 1976, Boston; the Sixth Annual Conference of the Society for Computer Medicine, November 11-13, 1976, Boston; the Symposium on Computer Assisted Data Processing in Nuclear Medicine, Society of Nuclear Medicine Computer Council, January 16-17, 1977, Atlanta; and the ACM Computer Science Conference, January 31-February 2, 1977, Atlanta)

Computer applications in biomedicine are growing in number at an increasingly rapid rate. Part of this growth is attributable to enterprising researchers and clinicians who purchase small general-purpose computers and develop the special hardware and software needed to fulfill their specific needs. Another part is accounted for by commercial, turnkey systems purchased on a ready-to-use basis. Even these commercial systems frequently provide for the development of user programs, since change is the only constant in the biomedical sciences. Like organisms, those systems incapable of adaption to a changing environment soon die away.

When a new computer application is described in a major journal (often many months after its development), researchers elsewhere are likely to desire to test the application themselves. Since software is seldom available from the authors due to time and economic constraints, a bottom-up development of the software may occur at several secondary installations.

Those installations which succeed in the effort may produce improvements in the various aspects of the software, which may (or more likely may not) find their way back to the journals, starting the process over again. Eventually, the software may be incorporated into a commercial system.

The serious inefficiencies in this system became acutely obvious to the nuclear medicine community, the first clincal discipline to extensively implement computers due to the inherently digital information it encounters. With the advent of minicomputers and the resultant drop in hardware prices, development costs for software and special interfaces became the major concerns. Persons involved in this effort quickly concluded that independent research development projects could be clinically viable only if a concerted effort were made to share developments among the various installations, effecting cost savings and avoiding duplication of effort.

As a result of this conclusion, Dr. Henry N. Wagner, Jr., then president of the Society of Nuclear Medicine, formed the SNM Computer Committee to further the idea of sharing.

Through the efforts of this committee and individuals at ORNL, the Division of Biomedical and Environmental Research of the Energy Research and Development Administration (ERDA) established the Biomedical Computing Technology Information Center (BCTIC) at the Oak Ridge National Laboratory in 1975 as a focal point for the sharing of computer technology in biomedicine in general and nuclear medicine in particular. ERDA shares support for BCTIC with the Society of Nuclear Medicine, the Food and Drug Administration's Bureau of Radiological Health, and the Society for Computer Medicine, organizations which have supported the concept since its inception.

BCTIC's mission is "to collect, organize, evaluate, and disseminate information in computing technology pertinent to biomedicine in general, and Nuclear Medicine in particular."

Specifically, BCTIC provides:

1. A clearinghouse for algorithms, computer programs, data, and interface designs pertinent to clinical and/or research biomedicine. (This service permits users to transfer technological advances to others at minimal cost and, in return, to acquire new technology from other sources.)

2. A bimonthly BCTIC Newsletter highlighting important developments in the field, surveying existing biomedical computer installations, noting upcoming events of interest, and providing a bibliography of recent literature.

3. Seminars, workshops, and topical meetings on biomedical computing.

4. A computerized information storage and retrieval system containing bibliographic citations and abstracts.

5. Promotion of hardware and software standards as they affect biomedical computing.

6. An index of equipment and clinical procedures performed at the various installations. (The aim of this Workbook of Clinical Resources is to enable users or prospective users of biomedical computing technology to locate other clinicians and researchers performing similar procedures and/or using similar equipment.)

BCTIC's other functions are rounded in scope, making it an international technology resource as opposed to a program library. Over 850 requests for information and services have been received from all parts of the world, and the Newsletter is distributed bimonthly to over 1200 persons and agencies worldwide. BCTIC has leveled a concerted and organized attack on the problems of biomedical computing technology transfer and is striving hard to realize its ultimate goal — improving patient care.

*BCTIC is sponsored by ERDA Division of Biomedical and Environmental Research, the HEW/FDA Bureau of Radiological Health, and the Society of Nuclear Medicine.

†Computer Sciences Division.

7.12 DIRECTORY OF COMPUTER USERS IN NUCLEAR MEDICINE*

R. L. Henne† J. J. Erickson‡ W. J. McClain† D. L. Kirch§

(Excerpt from ORNL/BCTIC-1, January, 1977)

The *Directory of Computer Users in Nuclear Medicine* is physically organized into two major divisions: a Users' Section and a Vendors' Section. The Users' Section consists primarily of detailed installation descriptions and Indexes to these descriptions. A typical Installation Description contains the name, address, type, and size of the institution, plus the names of persons to contact within the responding department. If the department has access to a central computer facility for data analysis or timesharing, that fact is stated along with the type of equipment available. Complete descriptions are then given for each dedicated data processing unit operated by the responding department, whether the unit is commercially available or locally developed. Included are the attached peripherals, languages used, modes of data collection, and other pertinent information.

Following the hardware descriptions are listed the type of studies for which the computers are utilized, including the language(s) used, the method of output, and an estimate of how often the study is performed.

Critical to the use of the indexes is the four-digit number appearing in angle brackets in the upper right-hand corner of each installation description. This is the Key Number, which will be used in the Indexes to locate individual installations. The installation descriptions are arranged physically in alphabetical order by state, within states by city, etc. (All non-U.S. installations appear following the U.S. installations, arranged alphabetically by nation.) This arrangement permits a user to easily locate any installations in his proximity, or within a given state.

The Installation Index, which appears immediately following this text, lists the name of each responding institution in alphabetical order, regardless of its location. With each name is given the key number by which the respective installation description may be located.

Following the descriptions are the three hardware indexes. The Central Computer Index lists those processors referred to by the respondees as "Central Computing Facilities." The Commercial System Index lists the commercially available nuclear medicine data acquisition and processing systems cited, and the User-System Index lists general-purpose minicomputers around which users have developed their own nuclear medicine systems.

In each case, a given computer or system is followed by the list of key numbers for those installations using that equipment.

Following the hardware indexes is the Studies Index. This index lists the different types of studies performed with the computer by the responding departments, again with lists of the key numbers for each type of study. The studies are arranged alphabetically within the index. In general, they appear as an organ followed by a letter abbreviation. The letter "S" indicates "Static." "F" indicates "Function" or "Flow." "P" is "Perfusion," and "V" is "Ventilation." Also used with heart studies are "E" for "Effusion," "O" for "Output," and "G" for "Gated," and for thyroid studies, "I" is used for "Image" and "U" for "Uptake."

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After the Users' Section is the Vendors' Section, which consists of short descriptions of current commercially available nuclear medicine systems as supplied by the vendors themselves. The descriptions are organized identically in order to facilitate comparisons, with comments appearing at the end. This section is intended to be used as a preliminary source of information regarding the various systems. As such, it is not intended to be (nor could it be) a fully complete and definitive description of the systems.

All the information included in this directory was obtained for the users and manufacturers by means of questionaires which were distributed by BCTIC. The very thing which makes a directory of this sort desirable, i.e., the rapid growth of the discipline, also makes it difficult to compile such a collection of information without including some obsolete data. In an attempt to reduce the amount of obsolete data and to include new institutions in future updates of the directory, a user questionaire is included. New and old computer users are invited and urged to submit this form to BCTIC any time they feel that the data from their institution should be updated.

[*]Research sponsored by ERDA Division of Biomedical and Environmental Research, the HEW/FDA Bureau of Radiological Health, and the Society of Nuclear Medicine.

[†]Computer Sciences Division.

[‡]Division of Nuclear Medicine, Vanderbilt University Hospital, Nashville, Tennessee.

[§]Nuclear Medicine Service, Veterans Administration Hospital, Denver, Colorado.

7.13 THE DIRECTORY OF COMPUTER USERS IN NUCLEAR MEDICINE: WHAT IS IT AND WHAT CAN IT DO FOR YOU?[*]

R. L. Henne[†] W. J. McClain[†] J. J. Erickson[‡]

(Abstract of paper presented at Symposium on Computer Assisted Data Processing in Nuclear Medicine, Society of Nuclear Medicine Computer Council, January 16-17, 1977, Atlanta, Georgia)

The recently published *Directory of Computer Users in Nuclear Medicine,* a joint project of the Biomedical Computing Technology Information Center (BCTIC) and the Workbook Task Group of the SNM Computer Council (formerly Committee), provides the present and/or prospective computer user with a source of valuable information. Using this document, one can locate other installations doing similar work with similar equipment, perhaps resulting in direct sharing of technology and less duplication of effort. This paper will detail the contents, arrangement, and use of the Directory, and, hopefully, will stimulate discussion on how future issues can be made more valuable.

[*]Research sponsored by ERDA Division of Biomedical and Environmental Research, the HEW/FDA Bureau of Radiological Health, and the Society of Nuclear Medicine.

[†]Computer Sciences Division.

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7.14 COMPUTER APPLICATIONS IN MEDICINE: A REVIEW*

R. L. Henne†

(Abstract of paper to be presented at Association for Computing Machinery Southeastern Regional Meeting, April 18-20, 1977, Biloxi, Mississippi)

One result of the growing interest in the application of computers to health care and biomedical research was the creation in 1975 of the Biomedical Computing Technology Information Center (BCTIC) at Oak Ridge National Laboratory. As a part of its mission to "collect, organize, evaluate, and disseminate information on computing technology pertinent to biomedicine in general and nuclear medicine in particular," BCTIC is accumulating a unique computerized data base of bibliographic citations and abstracts in the field. This paper presents a general review of current activity in the biomedical computing sciences as a result of initial examination of this data base and also considers areas of current research which may lead to future applications. The paper also presents a guide to further information on biomedical computing.

*Research sponsored by ERDA Division of Biomedical and Environmental Research, the HEW/FDA Bureau of Radiological Health, and the Society of Nuclear Medicine.

†Computer Sciences Division.

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MARCH 1, 1977

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C. A. WATSON[16,c]

QUALITY ASSURANCE COORDINATOR

W. R. CASTO[f]

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[2]LMFBR COORDINATOR
[3]GCBR COORDINATOR
[4]MFE COORDINATOR

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[16]DIVISION PROPERTY REPRESENTATIVE
[17]SEMINAR COORDINATORS

[a]ADJUNCT RESEARCH PARTICIPANT
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[c]ON ASSIGNMENT FROM FINANCE AND MATERIALS DIVISION
[d]ON ASSIGNMENT FROM INSTRUMENTATION AND CONTROLS DIVISION
[e]ON ASSIGNMENT FROM COMPUTER SCIENCES
[f]OPERATIONS DIVISION
[g]ON LOAN
[h]PART-TIME
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[j]CONSULTANT

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CAMBRIDGE, MASSACHUSETTS

ROBERT E. UHRIG, VICE PRESIDENT
OF NUCLEAR AFFAIRS
FLORIDA POWER & LIGHT CO.
MIAMI, FLORIDA

SCIENTIFIC AND PROFESSIONAL ACTIVITIES

L. S. Abbott	Chairman, Exhibits and Technical Tours, 5th International Conference on Reactor Shielding, Knoxville, April 1977
R. G. Alsmiller, Jr.	Member, Admissions Committee, American Nuclear Society Member, ERDA Advisory Panel on Accelerator Radiation Safety Member, ERDA Evaluation Panel on Electronuclear Fuel Production Chairman, ERDA Advisory Panel on Accelerator Radiation Safety for Review of the Positron Electron Storage Ring Project (PEP) at SLAC
D. E. Bartine	Chairman, Program Committee, Radiation Protection and Shielding Division, American Nuclear Society Member, Finance Committee, Oak Ridge Section, American Nuclear Society Member, Publications Committee, 5th International Conference on Reactor Shielding, Knoxville, April 1977 Invited Paper, American Nuclear Society 1975 Winter Meeting, "Shielding Calculations for a 300-MW(e) Gas-Cooled Fast Breeder Reactor" Seminar, University of Tennessee, March 7, 1977, "The Non-Proliferation Issue: General Aspects of the ORNL Program"
G. T. Chapman	Member, NASA Committee to review and make recommendations on proposals on gamma-ray-astronomy experiments on board the space shuttle during orbital flight test and/or Skylab-II flights, January 1977 Completed a four-day intensive training course on the use of the DECSYSTEM-10 Computer System, U. of Tennessee, December 1976
G. de Saussure	Member, National Program Committee (Division Representative), American Nuclear Society Member, Executive Committee, Reactor Physics Division, American Nuclear Society Best Paper Award of Reactor Physics Division, American Nuclear Society, November 1975, in collaboration with R. B. Perez, E. G. Silver, R. W. Ingle, and H. Weaver
J. K. Dickens	Member, American Nuclear Society Standard Subcommittee 5.1: Decay Heat Standard Invited Colloquim Speaker, Dept. of Physics, U. of Kentucky "Delayed Fission Product Energy Release and Consequences for a Loss-of-Coolant Accident," April 1976 Referee, *Nuclear Science and Engineering*
W. W. Engle, Jr.	Invited Paper, American Nuclear Society 1975 Winter Meeting, with M. B. Emmett and M. L. Williams, "Analysis of the Complex Reactor Cavity Shield in the FTR"
G. F. Flanagan	Assistant General Chairman, Conference on Advances in Reactor Physics, Gatlinburg (to be held in 1978) Secretary, Reactor Safety Division Program Committee, American Nuclear Society Member, Youth Steering Committee, American Nuclear Society Member, Technical Program Committee, Fast Reactor Safety Conference, Seattle (to be held in 1979) Member, American Nuclear Society Standards Subcommittee on Reactor Physics, 19.3 Chairman, Publicity Committee, Local Section, American Nuclear Society Chairman, Program Committee, Local Section, American Nuclear Society

Coordinator, ORNL Bimonthly Colloquim, 1976-77
Member, Ph.D. Recruiting Program, ORNL, 1976
Coordinator and teacher, Short Course on Reactor Safety, U. of Tennessee, September 1976
Assistant Professor, U. of Tennessee, Fast Reactor Safety Graduate Course, Spring Quarter, 1977

C. Y. Fu

Member, Cross Section Evaluation Working Group (CSEWG) Nuclear Model Codes Subcommittee
Referee, *Nuclear Science and Engineering*
Oral Presentations (4): "Guidelines for the Evaluations of Carbon as a Neutron Scattering Standard," Task Force on Standards for ENDF/B-V, Brookhaven National Laboratory, March 1976; "Calculations of Gas Production Cross Sections for ^{56}Fe and ^{63}Cu," Special Meeting of the CSEWG Standards Sbucommittee on Neutron-Induced Gas Production, Activation, and Dosimetry Cross Sections for ENDF/B-V, Los Alamos Scientific Laboratory, August 1976; "TNG Calculations of 11 Reaction Cross Sections for ^{59}Co up to 35 MeV," CSEWG Nuclear Model Codes Subcommittee, Hanford Engineering Development Laboratory, October 1976; "^{23}Na(n,γ), ^{54}Fe(n,p), ^{56}Fe(n,p) and ^{58}Fe(n,γ) Cross Sections for Reactor Dosimetry," Task Force on Reactor Dosimetry for ENDF/B-V, National Bureau of Standards, March 1977.

T. A. Gabriel

Awards Coordinator, Neutron Physics Division
Advisor to SLAC for PEP Shielding
Advisory to SLAC for Calorimeter Design

J. Gurney

Chairman, Program Committee, IAC Forum, 1977
Member, Public Relations Committee, IAC Forum, 1975-76
Member, Hospitality Committee, 5th International Conference on Radiation Shielding
Coordinator, IAC Forum Exhibit

P. M. Haas

Member, Publicity Committee, Local Section, American Nuclear Society
Member, Newsletter Staff, Local Section, American Nuclear Society
Member, Ad Hoc Committee for Review of Brookhaven National Laboratory Medical Reactor Operations
Participant, IEEE Reliability Seminar, February 1977

R. E. Maerker

Member, Shielding Subcommittee, CSEWG
Member Data Testing Subcommittee, CSEWG
Invited Paper, "An Example of Calculations with Coupled Sets and Simultaneous Comparisons with Experiment," CSEWG Meeting, Brookhaven National Laboratory

F. C. Maienschein

Member, ERDA Advisory Committee on Reactor Physics (ACRP)
Member, Nuclear Energy Agency Committee on Reactor Physics (NEACRP)
Member, University of Illinois, Department of Nuclear Engineering Advisory Committee
Keynote Address, 5th International Conference on Radiation Shielding, April 1977
Organized Special Session, New York Meeting of the American Nuclear Society on "Definitions of Breeding Ratio (BR) and Doubling Time (DT)," June 1977
General Chairman, American Nuclear Society Conference on Advances in Reactor Physics, Gatlinburg (to be held in 1978)

B. F. Maskewitz

Member, Honors and Awards Committee, American Nuclear Society (1976-1979)
Member, ANS-10 Subcommittee of the American Nuclear Society Standards Committee
Member, Executive Committee, Radiation Protection and Shielding Division, American Nuclear Society (1976-1979)

	Secretary-Treasurer, Society of Nuclear Medicine Council on Computers (1976-77); Nominee (1977-78) Member, Organizing Committee for the Annual Scientific Symposium of the Society of Nuclear Medicine Council on Computers (1971-1977) Member, Standards Committee, Society for Computer Medicine (1977-1978) Member, Program Committee, 1977 National Computer Conference, June 1977, Dallas, American Federation of Information Processing Societies (AFIPS) Deputy Chairman, 5th International Conference on Reactor Shielding, Knoxville, April 1977 Chairman, Publications, 5th International Conference on Information Processing in Medical Imaging, Vanderbilt University, Nashville, June 1977 Member, Advisory Committee, National Science Foundation sponsored Ultrasound Tissue Signature Project, Carnegie-Mellon Institute of Research, Pittsburgh (1977-1978) Member, Committee for Review of ERDA-NEA Information Exchange Agreement (1977) Member, Executive Committee, ORNL IAC Forum (1977) UCND Representative to the Argonne Code Center Archivist, MUMPS Development Committee
F. R. Mynatt	Chairman, Radiation Protection and Shielding Division, American Nuclear Society Chairman, Technical Program of the WATT-ec Conference, Knoxville, 1977 Member, Program Committee, 5th International Conference on Reactor Shielding, Knoxville, April 1977 Chairman, ORNL Computing Committee
R. W. Peelle	Working Group Chairman, "Specialists Meeting on Fast Neutron Fission Cross Sections of ^{233}U, ^{235}U, ^{238}U, and ^{239}Pu," June 1976, Argonne National Laboratory Member, Normalization and Standards Subcommittee of CSEWG Member, Internal Sabbatical Selection Panel Chairman, ORNL Proposal Review Committee Member, Program Committee, International Specialists Symposium on Neutron Standards and Applications, March 1977 Invited Lecture, "ORELA Usage for Condensed Matter Research," National Academy of Science, March 1977 Invited Paper, "Thermal and Epithermal ^{235}U Measurements," International Specialists Symposium on Neutron Standards and Applications, March 1977
F. G. Perey	Member, ERDA Nuclear Data Committee Member, Cross Section Evaluation Working Group (CSEWG) Chairman, Data Covariance Subcommittee, CSEWG Invited Paper, "Use of ORELA for Scattering Measurements in the MeV Region," New York Meeting of the American Physical Society, February 1976 Member, ANS 6 Subcommittee on SI Units
R. B. Perez	Member, Program Committee, American Nuclear Society
W. A. Rhoades	Cochairman, Neutron Physics Division Seminars
R. W. Roussin	Chairman, Shielding Subcommittee, CSEWG Chairman, Membership Committee, Radiation Protection and Shielding Division, American Nuclear Society Chairman, Proceedings, 5th International Conference on Reactor Shielding

R. T. Santoro	Member, Honors and Awards Committee, Local Section, American Nuclear Society Member, Publications Committee, Division of Controlled Nuclear Fusion, American Nuclear Society
D. L. Selby	Member, Membership Committee, Local Section, American Nuclear Society Member, Publications Committee, Local Section, American Nuclear Society Member, Editorial Staff, ANS Newsletter, Local Section, American Nuclear Society
E. G. Silver	Member, Education Committee, East Tennessee Section, American Nuclear Society Member, Program Committee, American Nuclear Society Lecture Series on "The Physics of Energy," ORAU-EEO Short Courses on Energy (four sessions)
D. K. Trubey	Chairman, 5th International Conference on Reactor Shielding, Knoxville, April 1977 Chairman, Nominating Committee, Radiation Protection and Shielding Division, American Nuclear Society Member, Public Information Education Committee, Local Section, American Nuclear Society Chairman, ANS-6 Radiation Protection and Shielding Subcommittee, Standards Committee, American Nuclear Society Member, ANS-6.5 Shielding Glossary, American Nuclear Society Member, Working Group on Code Interfaces and Formats, Safety Analysis Data Coordinating Group
D. R. Vondy	Member, M&C Benchmark Committee, American Nuclear Society Member, ANS-10 Standards Committee, American Nuclear Society Alternate Member, ANSI Standards Committee, X-3, American Nuclear Society Alternate Member, X-3 Fortran Committee, X3J3, American Nuclear Society Member, ERDA, Division of Reactor Development and Demonstration, Physics Branch, Large Core Methods Evaluation Committee Member, ERDA, Division of Reactor Development and Demonstration, Physics Branch, Cooperative Computer Coding Committee Coordinator, Two-Day VENTURE Users Workshop (December 1976)
C. R. Weisbin	Member, ANS 6.1.2 Standards Committee on Shielding, American Nuclear Society Member, Reactor Physics Division Program Committee, American Nuclear Society Member, ANS 5.1 Decay Heat Standard Working Group, American Nuclear Society Member, Code Evaluation Working Group, Committee on Computer Code Coordination Member, Cross Section Evaluation Working Group (CSEWG) Member, Program Committee, 1978 Reactor Physics Topical Meeting, "Advances in Reactor Physics," Gatlinburg
L. W. Weston	Evaluator, ^{240}Pu and ^{241}Pu Neutron Cross Sections for ENDF/B-V Representative, Neutron Physics Division, Continuing Education Program Committee
M. L. Williams	Invited Paper, "Analysis of the Complex Reactor Cavity Shield in the FTR," with W. W. Engle, Jr. and M. B. Emmett, American Nuclear Society Winter Meeting, November 1975

NEUTRON PHYSICS DIVISION SEMINARS AT ORNL

D. K. Olsen and W. A. Rhoades succeeded D. C. Larson and C. R. Weisbin as Seminar Cochairmen during the year 1976. The following seminars were held during the period covered by this report:

R. G. Alsmiller, Jr., Neutron Physics Division, "Uncertainties in Calculated Heating and Radiation Damage in the Toroidal Field Coil of a Tokamak Experimental Power Reactor Due to Neutron Cross-Section Errors"

R. G. Alsmiller, R. W. Peelle, Neutron Physics Division, and T. A. Lewis, Instrumentation and Controls Division, "ORELA Improvement Program - The Prebuncher"

D. E. Bartine, Neutron Physics Division, "Design Support Analysis for the GCFR Radial Shield"

C. M. Bartle, Australian National University, Canberra, Australia "^{6}Li(n,γ) Reactions"

J. Blons, Saclay, France, "Evidence for Rotational Bands in the ^{232}Th Fission Cross Section"

J. D. Callen, Fusion Energy Division, "Neutron Beam Heating Calculations for Present and Future Tokamaks"

J. K. Dickens, Neutron Physics Division, "Fission Product Decay Heat Measurements"

G. F. Flanagan, Neutron Physics Division, and N. M. Greene, Computer Sciences Division, "LMFBR Reactor Safety Code Data Base (SACRD)"

T. B. Fowler, Neutron Physics Division, "What Bold Venture Can Do for You"

T. A. Gabriel, Neutron Physics Division, "Radiation Damage Calculations: PKA Spectra and DPA Cross Sections"

D. V. Gopinath, Reactor Research Centre, Safety Research Laboratory, Kalpakkana, India, "Recent Development of the Anisotropic Source -- Flux Iteration Technique for Radiation Transport"

G. Hale, Los Alamos Scientific Laboratory, "R-Matrix Analysis of ^{6}Li and ^{10}B"

J. A. Harvey, J. Halperin, N. W. Hill, and S. Raman, Physics Division, ORELA, "(n,p) and (n,γ) on ^{59}Ni"

J. M. Kallfelz, Georgia Institute of Technology, "The European Approach to the Heterogeneous LMFBR"

R. A. Karam, Georgia Institute of Technology, "LMFBR Core Design"

F. R. Mynatt, Neutron Physics Division, "The ORNL Long-Range Computing Plan"

F. R. Mynatt, Neutron Physics Division, "The Accelerator Breeder Concept"

G. H. Neilson, Fusion Energy Division, "Tokamaks for the Layman"

E. M. Oblow, Neutron Physics Division, "Energy Channel Theory"

S. Plattard, Bruyères-le-Châtel, Saclay and Los Alamos Scientific Laboratory, "Neutron Physics at Los Alamos and Bruyères-le-Châtel"

J. B. Roberto, Solid State Division, "Energy Dependence of Neutron Damage in Metals: Theory and Experiment Using Be(d,n) Neutrons"

M. T. Robinson, Solid State Division, "Radiation Damage Analysis: A Critical Assessment"

R. W. Roussin, Neutron Physics Division, "Experience with the DCTR/DRRD Multigroup Libraries"

M. Saltmarsh, Physics Division, "INGRID: An Intense Neutron Generator for Radiation-Induced Damage"

R. T. Schneider, Nuclear Engineering Sciences Department, University of Florida, "Nuclear Pumped Lasers"

R. Sinclair, Atomic Energy Research Establishment, "Recent Neutron Experiments and Upgrading of the Linac at Harwell"

C. O. Slater, Neutron Physics Division, Analysis of GCFR Pin-Streaming Experiment at the Tower Shielding Facility"

R. G. Stokstad, Physics Division, "The Future of Nuclear Science: Progress Report from an Ad-Hoc Panel Established by the Committee on Nuclear Science, National Academy of Sciences"

A. Todd-Pokropek, University College Hospital, London, England, "IAEA Coordinated Research Program on the Intercomparison of Computer-Assisted Scintigraphic Techniques"

D. K. Trubey, Neutron Physics Division, "The Role of RSIC in the NPD Program"

D. R. Vondy, Neutron Physics Division, "Application of New Depletion Capability"

C. R. Weisbin, Neutron Physics Division, "Application of the FORSS Sensitivity Code System to Fast Reactor Analysis"

C. R. Weisbin, Neutron Physics Division, "Moving from Sensitivity Analysis to Uncertainty Analysis"

D. R. Winkler, Computer Sciences Division, "The New PDP-10 Monitor - What Can It Do?"

PUBLICATIONS AND PAPERS PRESENTED AT SCIENTIFIC MEETINGS

PUBLICATIONS*

ALSMILLER, R. G., JR.

"Nucleon-Meson Transport Calculations," p. 139 in "Spallation Nuclear Reactions and Their Applications," Shen/Merker (Eds.), D. Reidel Pub. Co., Dordrecht-Holland, 1976. (1975)

ALSMILLER, R. G., JR., F. S. ALSMILLER,** AND J. BARISH**

"Calculations Pertaining to the Design of a Prebuncher for a 150-MeV Electron Accelerator," ORNL/TM-5419 (in press). (1.29)

ALSMILLER, R. G., JR., AND J. BARISH**

"The Spatial Variation of the Damage Energy and Gas Production in the Experimental Volume of a Li(D,n) Neutron Radiation Damage Facility," ORNL/TM-5554 (October 1976). (6.9)

"Neutron Kerma Factors for H, C, N, O, and Tissue in the Energy Range of 20 to 70 MeV," ORNL/TM-5702 (December 1976). (6.17)

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ALSMILLER, R. G., JR., J. BARISH,** AND C. R. WEISBIN

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Sixth Symposium on the Sharing of Computer Programs and Technology in Nuclear Medicine, Atlanta, Georgia, January 26, 1976

MASKEWITZ, BETTY F., R. L. HENNE,** AND W. J. McCLAIN,** "Early Progress of the Biomedical Computing Technology Information Center (BCTIC)" (7.11)†

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