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PERIOD ENDING MAY 31, 1982**

F. C. Maienschein, Director

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OVERVIEW

The progress reports of the Engineering Physics Division, formerly the Neutron Physics Division, tend to be somewhat historical since they primarily consist of abstracts or summaries of publications that have appeared during the reporting period. As such, they reflect the continuing changes that have occurred in the past but do not always cover all the on-going research. Neither do they always indicate which areas of research are currently receiving the most emphasis. In this report, which covers the period from November 30, 1980 to May 31, 1982, an attempt is made to bridge this gap with introductions to the various sections that include some discussion of current research and of the directions in which the work is expected to proceed.

A review of the section titles will reveal no obvious change in subject matter since our last report was published. As has been our tradition for many years, we remain heavily involved in the development of data and methods required to calculate the transport of nuclear radiation through various systems, as well as in the application of those data and methods. Thus the first three sections of this report discuss measurements of nuclear data made at the Oak Ridge Electron Linear Accelerator (ORELA), together with analyses and evaluations of these and other data; describe how such data are prepared for use in transport calculations; and report on the development of the transport methods themselves. A fourth section discusses how through sensitivity and uncertainty analyses we can now determine which data are the most important for particular calculations and, moreover, what the uncertainties are on specific calculated results due to uncertainties associated with the input data or the calculational methods.

In addition, however, two of these first sections (Sections 3 and 4) reflect the fact that we are also adapting some of our methods of calculation to other areas of research, and in some cases we are developing new methods. In both sections, for example, will be found papers describing extensions of the methods to thermal-hydraulics problems, following initiatives that were begun during our last reporting period. Other examples are also cited, including a recent application of sensitivity analysis to the "CO₂ problem." And in our last report we introduced the concept of applying sensitivity analysis techniques to energy modeling problems.

One thing that may not be apparent in Sections 2 and 3 is the condition of "maturity" that has been reached by our major radiation transport methods and also by the methods used to prepare their input data (cross-section sets). With this maturity, the emphasis on transport methods development work within the division is subsiding. On the other hand, the extension of the associated sensitivity and uncertainty analysis techniques into new fields is accelerating. Further adaptation of the sensitivity and uncertainty methods to reactor problems is also anticipated. They are already increasingly being used in fast-reactor design, for example.

Section 5, titled "Integral Experiments and Nuclear Analyses," illustrates the variety of nuclear systems to which our current transport methods are applied. Subsections deal with the magnetic fusion energy program, with problems associated with the use of high-energy particle accelerators, and with various types of fission reactor programs. The title of the section reflects the difficulty in separating the analyses for specific systems from supporting integral experiments. A forte of the division through the years has been its confirmation of the applicability of a particular calculational

method to a given system by performing a pertinent integral experiment and using the same calculational method to analyze the experiment. Where differences between calculated and measured results for an experiment have existed, the differences have been used in adjusting the calculated results for the actual systems. We are continuing to perform experiments of this type for the Magnetic Fusion Energy Program (using a 14-MeV neutron source) and for fission reactor programs [at the Tower Shielding Facility (TSF)], although at a significantly reduced level, particularly for the latter. In recent years, integral experiments have also served as a basis for adjusting cross-section data sets, a trend which is apparent in papers included both in Section 5 and in Section 4.

Sections 6 and 7 describe two areas of research in which the division has become engaged relatively recently, although both sections were included in our last progress report. Section 6, "Energy Economics Modeling and Analyses," contains abstracts for a dozen publications corresponding to a high level of effort that peaked in 1980. Most of that work consisted of an energy model evaluation effort for the DOE Energy Information Administration (EIA). Currently we are engaged in a smaller but steadier program of model development for other parts of DOE, the current interests centering on future supplies of petroleum products and alternate liquid and gaseous fuels.

Section 7, "Safety and Reliability Assessments for Nuclear Power Reactors," reports on a program that began in the mid-1970s with studies of the safety and/or reliability of reactor components for the DOE Office of Reactor Research and Technology. This program later expanded into various types of risk assessments and human factors studies related to reactor safety systems.

Finally, in Section 8, "Information Analysis and Distribution," we continue to report on the myriad activities of the Radiation Shielding Information Center that was established within the division in 1963. RSIC operates under the management umbrella of EPIC (Engineering Physics Information Centers), along with the Technical Data Management Center (TDMC) established in 1976 to serve NRC programs. During this reporting period EPIC has also included the Center of Model Evaluation (COMET), which was established to support the ORNL energy model evaluation program (see preceding paragraph), but the work of that center has now been transferred elsewhere.

In addition to the EPIC centers, the division's two reactor data bases CREDO (Centralized Reliability Data Organization) and SACRD (Safety Analysis Computerized Reactor Data) remain in operation. And this past year the division has begun participation in ORNL's new Carbon Dioxide Information Center (CDIC), together with other divisions.

With the preceding overview of this report - and of the division's work - it seems appropriate to mention the pertinence of and incentives for some specific areas of our research. The nuclear-data measurements and evaluation, for example, continue to produce many results which are rather closely oriented towards the perceived needs of the numerous data users as determined through the U.S. Cross Section Evaluation Working Group (CSEWG). Improvements to the Evaluated Nuclear Data File (ENDF) thus continue as they have for a long time. However, some measurements tend to be definitive, the best example in our division being a measurement of the number of neutrons emitted by ^{252}Cf per fission. Values obtained for this universal standard (see Paper 1.11) are the result of six years of work and carry an estimated uncertainty of unprecedented accuracy ($<0.2\%$). The measurement of this standard, which is now being used to extend higher accuracies to other isotopes present in reactors, is unlikely to be repeated in a serious way in our lifetimes. Possibly in another six years we will look at measurements of the energy spectra of neutrons emitted from fission as similarly having been determined definitively by a series of ORELA measurements. Such a determination of data that are broadly applicable to a technology is a highly desirable role of a national laboratory.

Other papers in Section 1 illustrate the somewhat uncommon but nevertheless real phenomenon of significant feedback from applied measurements to basic research. The example in Paper 1.15 impacts the search for "missing" solar neutrinos since the best estimate of spectra from beta-ray decay which produces neutrinos is available from our fission-product measurements, which were made for reactor after-heat studies.

Several papers in Section 1 that describe nuclear data for thorium reflect a time lag. When alternate reactor systems were studied heavily a few years ago, the data for thorium were so ancient and incomplete as to hinder competent scoping studies. This lack now has been largely eliminated, and a center of interest has returned to ^{232}Th . The capture of neutrons in ^{238}U stubbornly resists definition, however. Capture measurements made in integral assemblies having spectra like fast reactors differ from differential cross-section measurements. Something is wrong or left out of the comparisons since differences in some energy ranges amount to about 10%, which should be well beyond the estimated accuracies of the measurements. Since capture in uranium to produce plutonium is important in all power reactors this puzzle must be pursued in an integrated program with other parts of the nuclear-data community.

Future nuclear-data needs have been assessed in a uniquely broad survey report (Paper 1.39) and the entire experimental program at ORELA has been reviewed (Paper 1.24). It is surely worth noting that our long campaign to include uncertainties (and the necessary correlations) in nuclear data and in the derived multigroup sets has substantially succeeded. The last reluctant or doubtful groups have now joined in their use and important habits are being established for the future. The task remains of establishing credible uncertainties for the significant materials in ENDF.

As has already been pointed out above, in the area of sensitivity and uncertainty analyses, we expect continuing extension of the methods developed for reactor physics into other disciplines, and an application to the "CO₂ problem" has been cited as one example. That example (Paper 4.14) reflects development of adjoint techniques as applied to determining uncertainties in the impact upon climate of increasing carbon dioxide in the atmosphere due to energy production. This work is now being extended to more realistic multidimensional global-circulation models of the atmosphere. Possibilities for optimization and intelligent control of distributed systems are considered in another very new effort.

In our work for specific reactor systems, it should be noted that our efforts in the magnetic fusion energy program include ORELA nuclear data measurements tailored to specifically meet the needs of that program. In addition, a small data evaluation effort has been initiated. It is also to be noted that the transport calculational methods we are applying in our nuclear analyses of fusion-related systems mostly are successful adaptations of methods originally developed for application to fission-related systems. In all of our fusion energy work, careful attention is given to the changing emphasis upon various concepts within the U.S. fusion program and upon engineering versus physics aspects of design to ensure that the methods and data developments remain relevant.

In our efforts for fission reactor systems, we continue to perform shielding studies for gas-cooled reactors. Our most recent work has been for the current High-Temperature Gas-Cooled Reactors (HTGRs), and for these we have identified a few areas requiring experiments, which are now being designed. Measurements and analyses performed for the now-abandoned Gas-Cooled Fast Breeder Reactor (GCFR) are well documented in reports cited in Section 5, and many of the results may be used for other reactor systems.

For light-water reactors, our several activities are understandably less design oriented and more directed toward facilitating the use of presently operating reactors. Thus, this work includes extensions of Electric Power Research Institute's analysis methods for fuel management and a long series

of studies with the goal of accurately characterizing the neutron fluence that damages pressure vessels. We are also participating in attempts to define the risk of pressure-vessel failure due to thermal shock. More broadly, we are performing risk studies to consider the impacts upon reactors of actions taken by human operators, with results of simulator studies compared with actual experience insofar as it can be reconstructed. Logical and systematic approaches to training and certification are considered. Other studies, mostly by subcontract, are concerned with potential but extremely unlikely common-mode failures of reactor safety systems due to events such as severe floods. These reliability and safety- or risk-related studies grew out of our operation of the nationally evaluated CREDO and SACRD data bases mentioned above, both of which now operate with sophisticated and smooth computer storage and retrieval. The reliability data are slowly but steadily being expanded in scope and statistical meaning.

The absence in Section 5 of a subheading on nuclear analyses supporting the design of Liquid Metal Fast Breeder Reactors may be noted. We are still engaged in such studies; however, the work is being reported in a second volume to this report (see ORNL-5897, Vol. II). In general, our efforts in the LMFBR program have been reduced.

In addition to the nuclear analyses for reactor systems, we are continuing analyses for various high-energy physics programs and for the Defense Nuclear Agency (DNA). Our support of high-energy physics evolves with the overall national effort. Thus while shielding and detector-efficiency calculations continue in high-energy accelerator studies, similar studies are under way for neutron and neutrino oscillation experiments. At the TSF we have participated in mockup measurements preliminary to a proposed neutron-antineutron oscillation experiment that would use the unique large-volume neutron source at the Oak Ridge Research Reactor. In addition, the use of accelerators as sources of neutrons for breeding fissile material is again under consideration and a small study in that program is anticipated.

For DNA, work continues on maintaining and updating the methods and data bases that were accumulated and made available to the defense community over the past decade. Specific air-transport calculations continue to define more precisely the radiation doses received in Japan during World War II, with new source data allowing a better evaluation. The importance of the observed radiation effects in setting national and world standards for radiation exposure provide impetus for updating the long-used earlier calculations of the exposures.

Many other specific investigations performed by the division will be apparent from the titles of the 193 papers included in the two volumes of this progress report. As might be guessed, we are also embarking on some additional projects that have not yet reached the stage to be introduced here. Those of us who were present when the old Applied Nuclear Division was formed in 1955 to perform reactor physics and shielding studies for the Aircraft Nuclear Propulsion Program could not have anticipated the variety of disciplines its successor, the Engineering Physics Division, is engaged in today, but, looking back, we see a logical development which no doubt will lead us into other challenging areas of research in the future.

Section 1

NUCLEAR CROSS SECTIONS AND RELATED QUANTITIES

Cross Section Measurements

Cross-Section Analyses, Evaluations, and Reviews

1.0. INTRODUCTION

As the time period covered by this progress report was beginning -- in December, 1980 -- the Oak Ridge Electron Linear Accelerator (ORELA) was ending its first decade of operation as a unique experiment facility dedicated to the study of neutron-nucleus interactions by time-of-flight techniques. The research results obtained from ORELA up to now have more than fulfilled the expectations of its founders. The data from ORELA have strongly influenced the Evaluated Nuclear Data File (ENDF/B), particularly insofar as the measurements largely have been tailored to address the needs for specific cross sections as expeditiously as possible. Documentation of the facility's accomplishments in providing needed cross sections for fission and fusion reactor development, and also in providing nuclear data contributing to a basic understanding of the nucleus (from experiments performed primarily by the ORNL Physics Division), is now available as a bibliography of ORELA research published through March, 1982. The bibliography is included as an appendix in a report (ORNL/TM-8225) that reviews past research and suggests areas for future research. The report also discusses possible future facility improvements.

Included in the ORELA bibliography are citations to most of the work abstracted or summarized in this section of the progress report. These papers and reports not only describe the recent ORELA measurements and their analyses, but also present evaluations of some of the cross-section (ENDF) files to which ORELA measurements have contributed. In addition, some papers review the status of and needs for certain cross sections, while others describe research directed toward finding satisfactory theoretical representations of cross sections for fertile and fissile nuclides in the resonance regions. Finally, some work on the development and use of cross-section covariance (uncertainty) data is summarized.

Cross-Section Measurements

During this reporting period, measurements at ORELA primarily have fallen under three classifications: (1) measurements of the resonant structures of the cross sections for the fertile nuclides; (2) measurements of the average number of neutrons emitted per fission of the fissile nuclides; and (3) measurements in the keV neutron energy region of the average cross sections of the reactor fuel nuclides and the higher actinide nuclides. These and other categories are discussed below.

Defining Resonant Structures of the Cross Sections for Fertile Nuclides. -- Except for those fueled with highly enriched uranium, all reactors contain significant quantities of ^{238}U or ^{232}Th in their fuel regions. At neutron energies below approximately 10 keV, these nuclides have boldly defined resonance structures that demand consideration in practical applications of nuclear data. At these energies the "dilute" average cross sections cannot be applied in practice because "self shielding" depresses the flux intensity through absorption in or scattering out of each narrow energy region where there is a strong resonance. To simulate the actual case in reactor calculations, the analyst must be able to reproduce the frequency function of various cross-section values as a function of neutron energy and fuel temperature for nuclides having high number densities in the system under study.

With the high (i.e. narrow) energy resolution available at ORELA, experimenters can observe almost all the fluctuations of the cross sections with energy that occur in an operating reactor at temperature. The experimental approaches used to define this structure through narrow-resolution transmission and capture cross-section measurements have been supplemented during this period with self-indication measurements using a broad range of absorber thicknesses. The self-indication measurements consist of the detection of neutron captures in a thin sample while the sample is in a flux spectrum modulated by penetration through various thicknesses of the same material. In principle, the experimental results can be combined to give shielded cross sections similar in definition to those required by the reactor analyst.

It is anticipated that problems and inconsistencies in the present interpretations of fast reactor integral experiments can be clarified or removed by a combination of these new efforts with the extension of explicit resolved resonance analysis for ^{238}U to energies extending above the present arbitrary upper boundary of 4 keV. Figure 1.0.1 illustrates the strong resonance structure actually observed in the 6-keV energy region of ^{238}U .

Measurements of Average Number of Neutrons Emitted Per Fission. -- Our experimental program to measure the number of prompt neutrons per fission for the ^{252}Cf spontaneous fission standard has been completed and has yielded a value of $3.773 \pm \text{about } 0.2\%$!! This new result is already influencing current data bases because it has been available in preliminary form for some time. This value is about one-half of a percent higher than that used prior to ENDF/B-V and leads to improved consistency in evaluation of the thermal cross sections of fissile nuclides and in the analyses of series of critical experiments with varying dimension. Work at ORELA to measure the energy dependence of this parameter for the major fuel nuclides is also far along, with results for ^{233}U released during this reporting period.

Measurements of Average Cross Sections in keV Neutron Energy Region for Fissile Nuclides and Higher Actinides. -- At energies where the resonances of the fuel nuclides overlap sufficiently that the cross sections are relatively smooth, the measurement of average cross sections is appropriate. Work completed during this reporting period has continued ORELA's strong influence on cross sections for the fuel nuclides and higher actinides in this energy region. For fast reactors, over one-half of the sensitivity of important parameters to neutron capture in fuel materials occurs at energies below 25 keV, which is an energy range for which ORELA techniques are required. Figure 1.0.2 shows the average capture cross-section measurements made for ^{237}Np , a nuclide especially important for the estimation of the concentration of ^{238}Pu in nuclear reactor waste.

Measurements of Fission-Product Yields and Beta-Ray Emissions. -- A few years after completion of a series of definitive experiments performed at ORNL to measure fission-product decay heat in the time range of 1 s to 10^4 s after fission, many spinoff papers from that work are still appearing. Considerable information on fission-product yields based on radiometric determinations using extremely small samples of the various fissile materials has been released. In addition, observations of the energy spectra of beta rays emitted at very short times after fission have been most useful in validating calculations that also permit estimation of reactor neutrino spectra important to various physics efforts across the globe to study neutrino apparent absorption cross sections. Most of the "easy" experiments to determine fission-product yields have now been performed, but more difficult ones to measure fission-product yields as a function of incident neutron energy would be interesting and worthwhile.

Fig. 1.0.1. Measured Neutron Transmissions Through 0.254- and 3.620-cm-thick ^{238}U Samples.

Fig. 1.0.2. Average Capture Cross Section of Neptunium from 100 eV to 200 keV. The solid line is the fit with average resonance parameters and the dashed line is ENDF/B-V. The average resonance parameter fit does not take into account competition from inelastic neutron scattering. The lowest levels of ^{237}Np are indicated.

ORELA Development Work: Improving the Reliability and Intensity per Pulse of the Accelerator. -- The report mentioned above on the ORELA facility and its research (see Paper 1.24 of this report) discusses directions for future development of the facility. The performance of ORELA depends primarily on the amount of microwave energy stored in the waveguide when the electron pulse is injected and on the magnitude and sharpness of that electron pulse. The design values of the corresponding parameters are difficult to meet routinely and therefore provide a challenging focus for machine improvement. During this reporting period a paper was issued documenting the techniques found to be the most reliable for producing the electron guns that are a crucial element of the injector system. More work in this area will be required.

As feasibility is fully proven for various nuclear power devices and attention gradually turns to cost minimization and eventually to more efficient fuel utilization, the need for detailed knowledge of the cross-section data base increases. While many cross sections for those nuclides that are present in fission reactors only in small concentrations may now be satisfactorily known, a great amount of challenging work remains to be accomplished to make available sufficiently accurate data for the nuclides present in greater concentrations. Also, investigations of the nuclides and energy regions of interest to the design of fusion reactors are needed. The ORELA facility represents a valued national resource for obtaining the hard experimental data without which precise cross sections cannot be obtained.

Cross-Section Analyses, Evaluations, and Reviews

Cross-section evaluation remains an important effort of the division, with several staff members participating in the work of CSEWG (Cross Section Evaluation Working Group) to refine and improve the ENDF/B data files. An important facet of the evaluations is, of course, the improvement of evaluation techniques, including the development of theoretical representations of specific cross sections over energy regions. This subsection also includes the development of cross-section covariance data, reviews and critiques of cross-section data, and assessments of future cross-section needs.

ENDF/B Evaluations. -- The ENDF/B-V data for calcium have been reevaluated to include the effects of precompound particle emission; the results are available in ENDF/B format from our Radiation Shielding Information Center. In addition, descriptions of the ENDF/B-V evaluations for carbon, calcium, iron, and copper, and for ENDF/B-V Revision 2 evaluations for calcium and iron have been published. On-going evaluation work presently centers on Mod 2 of the ENDF/B-V file, which is to be completed this year.

Cross-Section Reviews and Critiques. -- A critical review of some of the problems associated with the measurement and representation of neutron cross sections of fertile and fissile nuclei in the resolved and unresolved regions has been published, with recommendations for improvements included. Some of these recommendations are an outgrowth of studies to determine the sensitivity of self-shielded ^{238}U cross sections to the uncertain average resonance parameters used to represent resonance-region cross sections above 4 keV. Other reviews centered on summarizing the status of knowledge concerning particular classes of cross sections. Such summaries are often useful to help focus the attention of scientists at various conferences.

Development of Techniques and Data Bases for Uncertainty Analyses. -- Two papers represent the considerable effort expended in recent years to determine and adequately represent the covariances of evaluated cross sections. This effort, no longer confined to our laboratory, requires explicit representation of covariances among experimental and theoretical results, improved evaluation techniques to combine such data retaining covariance information, and development of methods to codify evaluated covariances so they can be applied in nuclear analyses to obtain uncertainties in computed results. Many of the important results of this effort are now appearing as part of the documentation of individual experiments or evaluations. The goal of a credible evaluated nuclear data base that includes commensurate covariance data now appears attainable, even though many additional developments will be required.

Defining Cross-Section Needs. -- Finally, one task assumed by the division this past reporting period was to summarize and explain requirements for cross-section data that are currently identified as essential to nuclear energy programs and to provide some insight into how these data might be obtained. The assessment of the status of the current cross sections was based in part on tests of ENDF/B files against benchmark integral experiments.

8

Cross-Section Measurements

1.1. ^{206}Pb LEVEL STRUCTURE FROM $^{206}\text{Pb}(n,n'\gamma)$ MEASUREMENTS*

J. K. Dickens

(Abstract of ORNL/TM-8157, March 1982)

A study of gamma-ray data produced by neutron inelastic scattering from a lead sample enriched in the isotope ^{206}Pb has resulted in placements, or tentative placements, of 146 gamma rays as transitions among 112 known or postulated levels of the ^{206}Pb level structure.

*Research sponsored by U.S. DOE Office of Basic Energy Sciences.

1.2. THE RADIATIVE CAPTURE YIELD OF THORIUM-232 FROM 100 TO 4000 eV*

R. B. Perez G. de Saussure R. L. Macklin[†]
J. Halperin[‡] N. W. Hill[§]

[Abstract of Nucl. Sci. Eng. 80, 189 (1982)]

The neutron capture yield in two ^{232}Th samples (0.0008 and 0.0027 atom/b, respectively) were measured with the ORELA time-of-flight facility over incident neutron energies from 100 eV up to 4 keV. A detailed comparison of the measured capture yields with calculations based on ENDF/B-V resonance parameters suggests that above 500 eV the evaluation needs additional work: in particular the average capture appears systematically underestimated.

*Research sponsored by U.S. Department of Energy.

[†]Physics Division.

[‡]Chemistry Division.

[§]Instrumentation and Controls Division.

1.3. THORIUM S-WAVE NEUTRON WIDTHS FROM 21 TO 2006 eV*

D. K. Olsen R. W. Ingle[†] J. L. Portney[‡]

(Reprint of Trans. Am. Nucl. Soc. 38, 647, 1981)

A ^{232}Th total cross-section measurement in the resolved resonance region has been requested

with a 2% accuracy to obtain resonance parameters with a 5% accuracy.¹ These data are required for both fast and light water breeder reactor studies. Inspection of the prior ^{232}Th differential data base shows several problems: the thermal cross sections seem inconsistent; the measured capture widths have large uncertainties; the two rather complete sets of measured^{2,3} neutron widths are systematically discrepant; and the differential data appear to give cadmium capture ratios and shielded-capture resonance integrals smaller than those required by integral measurements. To improve the differential data base, we have measured neutron transmission spectra through eight samples of ^{232}Th . Resonance parameters have been obtained from these data using the SIOB computer code.^{4,5} Fits to these data up to 440 eV, which concentrated on the capture widths, have been previously reported.⁶ In this paper, we report the results of extending these fits to 2.0 keV and discuss the result in terms of the s-wave strength function and the dilute-capture resonance integral.

Transmission spectra as a function of neutron flight time were measured at 40 m from the Oak Ridge Linear Accelerator (ORELA) tantalum-target-moderator assembly employing an 88.9-mm diam, 1.0-mm-thick lithium glass disk viewed edge on by two RCA-8575 phototubes. The summed fast signals, in coincidence with biased slow signals, produced 23 522 channel flight time spectra stored in a large-capacity, rapid, random-access disk attached to a SEL 810 computer that automatically updated various scalars and cycled up to 20 filters and sample combinations in and out of the beam. Table 1.3.1 lists some of the experimental parameters. More details of the measurement can be found in Ref. 7 along with a description of the data reduction. In particular, the transmission normalizations are believed to be accurate to within $\pm 0.5\%$ and less than a 1% time-dependent background, as determined from black ^{238}U , aluminum, and ^{232}Th resonances, was subtracted from the spectra.

The eight transmission spectra were simultaneously least-squares shape fitted using the

Table 1.3.1. Experimental Parameters

Flight path	40.015
Pulse rate	150 pps
Pulse width	12 ns
Beam power	~3.5 kW
γ -filter	Pb-1.9 cm
Time filter	B-1.1 kg/m ²
Bgd. filters	Al-0/2663 b ⁻¹ ²³⁸ U-0.0188 b ⁻¹
²³² Th samples	0.19308 b ⁻¹
	0.11588 b ⁻¹
	0.03868 b ⁻¹
	0.012566 b ⁻¹
	0.004030 b ⁻¹
	0.0013949 b ⁻¹
	0.0004537 b ⁻¹
	0.0001608 b ⁻¹

SIOB computer code,^{4,5} which employs the multilevel Breit-Wigner approximation. In addition, considerable care was taken to account for distant levels so that the accurately measured transmission normalizations could be exploited in the fits. In particular, the data were divided into the 15 regions listed in Table 1.3.2. The cross section for each region was calculated, with the s-wave contribution from resonances extending ~250 eV beyond the fitted region explicitly included. Even more distant levels were accounted for with "picket fence" terms. These cross sections were Gaussian Doppler broadened, giving eight transmissions that were resolution broadened with an asymmetric resolution function⁵ and least-squares fitted to the data by varying the resonance parameters, effective radius, and the resolution width.

Several iterations of this procedure resulted in the effective radii "a" and equivalent moderator distances "d" of Table 1.3.2, which are satisfying energy independent. These average the appropriate values of 97.2 fm and 25.4 mm, respectively. More importantly, Fig. 1.3.1 summarizes the s-wave neutron widths for the ENDF/B-V population⁸ where local strength functions for 0.5-keV intervals are plotted as percent differences from the ENDF/B-V evaluation. Also shown are the Rahn *et al.*² and Ribon³ measured values and the Derrien⁹ and Los Alamos National Laboratory¹⁰ evaluated values. The ORELA

Table 1.3.2. Least-Squares Fitted Effective Radii and Resolution Full-Width at Half-Maximum

Region (eV)	a(fm)	d(mm)
9 - 89	96.4	26.4
89 - 186	97.4	25.5
186 - 295	97.4	24.2
295 - 440	96.4	23.4
440 - 551	97.4	23.6
551 - 636	98.0	23.7
636 - 759	97.5	23.7
759 - 881	97.2	23.9
881 - 1001	96.0	23.3
1001 - 1170	96.8	23.6
1170 - 1319	96.3	24.1
1319 - 1495	96.9	22.9
1495 - 1690	97.4	23.8
1690 - 1880	96.9	23.9
1880 - 2030	98.2	24.0

Fig. 1.3.1. Comparison of ²³²Th Local s-Wave Strength Functions for 0.5-keV Intervals. The local strength functions, all assuming the ENDF/B-V s-wave population, are plotted as percent differences from the ENDF/B-V values. Above 0.5 keV, the Oak Ridge National Laboratory results give the largest strength function that has been measured.

measurement and analysis give the largest s-wave strength function above 0.5 keV that has been reported. The direct comparison of the ORELA transmission to ENDF/B-V calculated transmissions in Ref. 7 indicate that the trend shown in Fig. 1.3.1 continues to 4 keV.

The neutron and capture widths described in this paper and in Ref. 6 increase the agreement between calculated and integrally measured dilute capture resonance integrals $I\gamma$. Greneche¹¹ reports an evaluated $I\gamma = 85.8 \pm 0.9$ b (experiment) ± 1.6 b (normalization) from a review of the available integral measurements. Both the ENDF/B-V⁸ and Derrien⁷ evaluations give $I\gamma = 83.7$ b, whereas the ORELA widths would give $I\gamma = 86.2$ b.

*Research sponsored by U.S. DOE Reactor Research and Technology.

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1.4. MEASUREMENT AND RESONANCE ANALYSIS OF THE ^{232}Th TOTAL CROSS SECTION*

D. K. Olsen R. W. Ingle[†] J. L. Portney[‡]

(Abstract of *Nucl. Sci. Eng.*, in press)

Using the ORELA pulsed neutron source and a 1-mm-thick Li-glass detector, neutron transmission spectra through ^{232}Th have been measured at 22 m and 40 m. At 22 m, transmission spectra through five sample thicknesses were measured from 7 MeV to 15 eV. At 40 m, spectra through eight sample thicknesses were measured from 15 eV to 4 keV. The resulting total cross section from 0.1 to 20.0 eV is smaller than that given by the ENDF/B-V evaluation. Least-squares shape analysis of the transmissions up to 2.0 keV gives larger neutron widths above 0.5 keV than those previously reported. An average radiation width of 25.2 MeV is obtained for 19 low-energy s-wave resonances.

*Research sponsored by U.S. DOE Office of Reactor Research and Technology.

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1.5. MEASUREMENT OF NEUTRON TRANSMISSION SPECTRA THROUGH ^{232}Th FROM 8 meV TO 4 keV*

D. K. Olsen R. W. Ingle[†]

(Abstract of ORNL/TM-7661 (ENDF-307), April 1981)

Neutron transmission spectra through room-temperature ^{232}Th samples have been measured using the time-of-flight technique, the ORELA pulsed neutron source, and a 1-mm-thick Li-glass detector. The measurement and data reduction are described in detail. The 40-m transmission spectra through eight samples are directly compared from 15 to 4000 eV, with resolution-broadened transmission spectra calculated from the ENDF/B-V total cross section. Two sets of 22-m transmission spectra through five samples are combined into one total cross section from

0.008 to 15.0 eV and compared with the ENDF/B-V evaluation.

*Research sponsored by U.S. DOE Office of Reactor Research and Technology.

[†]Instrumentation & Controls Division.

1.6. K-COMPONENTS FOR THE ~ 1.4 , ~ 1.6 , AND ~ 1.7 -MeV STRUCTURES IN THE FISSION OF $^{232}\text{Th} + n^0$

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G. de Saussure R. B. Perez J. A. Harvey[§]

(Abstract of Los Alamos Report LA-8671, July 1981)

Neutron-induced angle-integrated fission cross sections of ^{232}Th were measured from 0.7 to 9 MeV with a nominal neutron energy resolution of 0.15 ns/m. Data were taken for the angular intervals 0 to 23.4°, 0 to 33.7°, 0 to 51.7°, and 0 to 90°. The structures at ~ 1.4 , ~ 1.6 , and ~ 1.7 MeV were interpreted in terms of rotational bands $K = 1/2$, $3/2$, and $\geq 5/2$. The approximate relative fission strengths for the K-bands are in the proportion 1.7:2.4:1.0, 0.0:2.6:1.0, and 1.0:2.8:0.0 for the three structures, respectively.

*Research sponsored by U.S. DOE Division of Reactor Research and Technology.

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[§]Instrumentation and Controls Division.

[¶]Physics Division.

1.7. MEASUREMENTS OF THE ^{238}U CAPTURE CROSS SECTION AND SELF-INDICATION RATIO FOR INCIDENT NEUTRON ENERGIES FROM 5 eV TO 10 keV*

Jing-Tong Yang[†]

(Abstract of ORNL/TM-8223, March 1982; also Dissertation for Doctor of Philosophy Degree, University of Tennessee)

The Oak Ridge Electron Linear Accelerator (ORELA) was utilized as the pulsed neutron source for measurements of the ^{238}U neutron capture cross section and self-indication ratio by the time-of-flight technique. Capture gamma rays were detected by four non-hydrogenous (C_6F_6) scintillation detectors. The incident neutron flux

was monitored by a $^{10}\text{BF}_3$ ionization chamber. The cross section was normalized by the saturated resonance technique.

Extensive tests of the pulse-height weighting technique for total energy detector (TED) were performed to validate the experimental results.

The experimental results were compared with theoretical computations of the capture yield and self-indication ratio based on the neutron resonance parameters prescribed by the ENDF/B-V evaluation. It was found that the neutron capture yield and self-indication ratio measurements performed in this work validate the results of the ENDF/B-V evaluation within the bounds of the experimental uncertainties.

*Research sponsored by U.S. DOE Office of Reactor Research and Technology.

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1.8. ^{238}U SELF-INDICATION RATIO MEASUREMENTS IN THE RESONANCE REGION*

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G. de Saussure R. B. Perez J. H. Todd

(Summary of paper to be presented at ANS Topical Meeting on Advances in Reactor Physics and Core Thermal Hydraulics, Kiamesha Lake, New York, September 22-24, 1982)

An accurate knowledge of the ^{238}U resonance parameters and of the $^{238}\text{U}(n,\gamma)$ cross section in the resolved resonance region is required for the calculation of several parameters in both thermal¹ and fast² reactors. Block *et al.*² at RPI have demonstrated the usefulness of self-indication ratio measurements to test the adequacy of resonance parameters. The purpose of this paper is to compare a set of self-indication ratio measurements, performed over a wide range of ^{238}U sample thicknesses, with a calculation based on the ENDF/B-V evaluated data over the resolved resonance region.

The Oak Ridge Electron Linear Accelerator (ORELA) was utilized as the pulsed source of neutrons for the time-of-flight measurements. A ^{238}U capture sample (0.0031 atoms/barn) was surrounded by four nonhydrogenous C_6F_6 scintillation detectors, placed at the 40-m flight-path station. Five high purity ^{238}U self-indication

samples (0.0038, 0.0124, 0.0341, and 0.1146 atoms/barn) were alternated in and out of the neutron beam by a computer-controlled sample changer located at the 20-m flight-path station. The energy region between 0.1 keV and 4 keV was divided into 32 equal-lethargy groups, a group structure also utilized for similar measurements at RPI.⁴

The experimental self-indication ratio averaged over a given neutron energy interval was obtained as the ratio of the net counts in the interval, with a ²³⁸U self-indication sample in the beam, to the net counts in the interval for open-beam conditions. The theoretical average self-indication ratio (SIR) over the energy interval ΔE was calculated from the expression

$$\langle (SIR) \rangle_{\Delta E} = \sum_i T_i F_i / \sum_i F_i, \quad (1)$$

where the sum is taken over the channels corresponding to the energy interval ΔE ,

$$T_i = \exp \{-n_1 \sigma_{ii}\}, \quad (2)$$

F_i = capture probability

$$\approx [1 - \exp \{-n_2 \sigma_{ii}\}] \frac{\sigma_{\gamma i}}{\sigma_{ii}}, \quad (3)$$

n_1 = thickness of the self-indication sample,

n_2 = thickness of the capture sample,

and where the Doppler-broadened capture cross section $\sigma_{\gamma i}$ and total cross section σ_{ii} were computed following standard ENDF/B-V procedures.

The effect of assuming a flat incident neutron spectrum and of neglecting the multiple scattering in (1) are evaluated to be small compared to the errors discussed below. The statistical error in the measurements was usually less than 1%. The systematic error was dominated by uncertainties in the background and varied from a few percent to 50% or more for those energy groups and sample thicknesses where the foreground to background ratio was low. In Figs. 1.8.1 and 1.8.2 we show typical results for the low and high energy groups. The figures show the SIR ratios for Group 1, 152 - 168 eV; Group 2, 168 - 186 eV; Group 31, 3058 - 3380 eV; and Group 32, 3380 - 3736 eV. The present measurements are

Fig. 1.8.1. Comparison of the Present Self-Indication Ratio Measurement with the Data of Byoun *et al.* (Ref. 4) and the Calculation Based on ENDF/B-V for the Neutron Energy Group 29 (2503 keV $\leq E \leq$ 2767 eV) and Group 30 (2767 eV $\leq E \leq$ 3058 eV) as a Function of Sample Thickness.

Fig. 1.8.2. Comparison of the Present Self-Indication Ratio Measurement with the Data of Byoun *et al.* (Ref. 4) and the Calculation Based on ENDF/B-V for the Neutron Energy Group 31 (3058 eV $\leq E \leq$ 3380 eV) and Group 32 (3380 eV $\leq E \leq$ 3736 eV) as a Function of Sample Thickness.

compared to the data of Byoun⁴ and to the calculation based on ENDF/B-V. Note that in Group 2 there are no large resonances, so that the signal-to-background ratio is small and the errors very large. The present self-indication ratio measurements appear to be generally consistent with the results of the ENDF/B-V calculation.

It is worthy to note that self-indication ratio measurements provide stringent tests of theoretical models because of the added requirement of consistency within each neutron energy group for all self-indication sample thicknesses. Thus it is hoped that these measurements, together with similar previous self-indication data, will provide a useful integral test for differential cross-section adjustment and validation procedures for the ENDF/B-V files.

*Research sponsored by U.S. DOE Division of Reactor Research and Technology.

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1.9. RESOLVED RESONANCE PARAMETERS FOR ²³⁸U FROM 4 TO 6 keV*

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(Reprint of Trans. Am. Nucl. Soc. 41, 560, 1982)

The ²³⁸U shielded-capture cross sections are among the most important nuclear data for reactor application. Nevertheless, a precise knowledge of these cross sections over a fast-reactor spectrum remains somewhat illusive. In particular, the ratio of ²³⁸U capture to ²³⁹Pu fission, calculated from ENDF/B-V differential

data,¹ tends to be larger than those measured in integral experiments.² The source of this possible discrepancy is not clearly understood. Below 4 keV, the capture and scattering cross sections are calculated from resolved-resonance parameters which are reasonably well established.³ Between 4 and 149 keV, these cross sections are calculated from average parameters based on lower-energy resonances and adjusted to fit the measured, broad-group, infinitely-dilute capture cross section. In this abstract, we report the results from a resonance analysis of the ORELA 150-m ²³⁸U transmission data,^{4,5} allowing an extension of the resolved resonance region to higher energies. The resultant s-wave strength function from 4 to 6 keV is substantially smaller than that from 0 to 4 keV.

The measurement (up to 100 keV) and resonance analysis (up to 4 keV) have been reported previously in detail.^{4,5} Transmission TOF spectra were measured in the 150-m station of flight path 6 of ORELA employing 800 12-ns-wide neutron pulses per second from a water-moderated and water-cooled Ta target. Neutrons were detected with a 12.7-mm thick and 111-mm-diameter Li-glass detector whose face was optically coupled to a RCA 4522 phototube. Filters of ¹⁰B, Pb and Al, Mn, and S were used to prevent time overlap, to attenuate gamma flash and to measure the background, respectively. Transmission spectra, believed to be accurate to $\pm 1.5\%$ of the average potential-scattering transmissions, were measured through 0.076-, 0.254-, 1.080-, and 3.620-cm-thick samples.

The four transmission spectra were simultaneously least-squares shape fitted with the SIOB computer code^{5,6} which contains both Doppler and resolution broadening and employs the multilevel Breit-Wigner formalism. The data were divided into 8 energy regions, each fitted separately with fixed capture widths and variable resonance energies, neutron widths, effective radii, and resolution function parameters. Approximately 200 eV of levels above and below the fitted region were explicitly contained in the cross section with levels beyond accounted for with picket fence terms. At these energies, most of the uncertainty in the extracted neutron widths

is caused by uncertainty in the shape and width of the resolution function from experimental effects. The resolution function was taken to be of the form x^2e^{-x} , expected from an idealized homogeneous moderator, with a variable width. In addition, the transmissions were folded with an exponential tail expected from multiple scattering in the detector, light pipe, phototube, air and beam line. This exponential had a variable decay constant and strength. The resolution parameters and effective radii converged to reasonable values somewhat independent of incident neutron energy.

Figure 1.9.1 shows a section of a simultaneous fit to the four transmission spectra around 5520 eV. The shape fits are reasonably good. The transmission maxima of the large s-wave resonances are largely obscured by the asymmetric resolution function. Altogether, parameters for 151 resonances were obtained. The derived neu-

Fig. 1.9.1. Simultaneous Least-Squares Shape Fit With Multilevel Breit-Wigner Formalism to Transmission Spectra Through 0.076- (Upper Curve), 0.254-, 1.080-, and 3.62-cm-Thick (Lower Curve) Samples of ^{238}U Resulting in Resonance Energies and Neutron Widths. The ordinate tick marks are for oil intervals with the transmissions arbitrarily offset for clarity.

tron widths are believed to have a systematic uncertainty of less than 7%.

The neutron width results are summarized in Table 1.13.1 where local "s-wave" strength functions, S^0 , from this and other measurements⁷⁻⁹ are listed for various energy intervals. The ENDF/B-V evaluation³ and the ORELA data⁴ give 1.17×10^{-4} and 1.20×10^{-4} , respectively, for S^0 over the resolved-resonance region, whereas $S^0 = 1.05 \times 10^{-4}$ is employed in the ENDF/B-V file¹ for the unresolved-resonance region. The ORELA local S^0 from 4 to 6 keV, $\sim 0.70 \times 10^{-4}$ assuming all observed resonances to be s-waves, as well as those from the other measurements⁷⁻⁹ are substantially smaller but are somewhat confused since all observed resonances have been included. Only those resonances with very large neutron widths are unambiguously s-wave. In addition, many smaller resonances have surely been missed; however, their contribution to S^0 should be small. The observed smaller resonances can be either s-wave or p-wave; however, estimates of the p-wave resonance population could only reduce the S^0 values in Table 1.9.1. Average parameters from the ENDF/B-V resolved-resonance region³ indicate that at most $\sim 0.1 \times 10^{-4}$ of the values for S^0 can be from p-waves.

Table 1.9.1. Local s-Wave Strength Function ($\times 10^4$)

Energy Interval (keV)	ENDF/B-V Ref. 3	ORELA Ref. 5	Geel Ref. 8	Rahn Ref. 7	JAERI Ref. 9
0.0 - 4.0	1.17	1.21	1.19	1.07	1.14
4.0 - 4.5	1.05	0.44	0.34	0.31	0.33
4.5 - 5.0	1.05	0.85	0.74		
5.0 - 5.5	1.05	0.58	0.42		
5.5 - 6.0	1.05	0.91			

In conclusion, energies and neutron widths for 151 resonances from 4 keV up to 6 keV have been obtained from the ORELA 150-m ^{238}U transmission data. These parameters would allow an extension of the resolved resonance region up from 4 to 6 keV and would give a substantially smaller S^0 in that region than is presently assumed. This more accurate treatment of the capture and scattering cross sections may contri-

bute to the solution of the fast-reactor, shielded-capture problem.

*Research sponsored by U.S. DOE Office of Reactor Research and Technology.

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1.10. MEASUREMENT OF THE AVERAGE NUMBER OF PROMPT NEUTRONS EMITTED PER FISSION OF ^{233}U RELATIVE TO ^{252}Cf FOR THE ENERGY REGION 500 eV TO 10 MeV AND BELOW 0.3 eV*

R. Gwin R. R. Spencer R. W. Ingle[†]

[Abstract of ORNL/TM-7988 (ENDF-315), November 1981]

The energy dependence of the average number of prompt fission neutrons emitted per fission, $\bar{\nu}_p(E)$, has been measured for ^{233}U relative to $\bar{\nu}_p$ for ^{252}Cf over the neutron energy ranges 500 eV to 10 MeV and below 0.3 eV. A large Gd-loaded liquid scintillator was used to detect neutrons and the samples of ^{233}U and ^{252}Cf were contained in fission chambers. The present results for $\bar{\nu}_p(E)$ for ^{233}U are in accord with the experimental results of Boldeman and the evaluated results of Lemmel in the thermal energy range, but in the neutron energy region between 100 keV and 1 MeV the present data are 1% or more larger than other experimental values.

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[†]Instrumentation and Controls Division.

1.11. A MEASUREMENT OF THE AVERAGE NUMBER OF PROMPT NEUTRONS FROM SPONTANEOUS FISSION OF CALIFORNIUM-252*

R. R. Spencer R. Gwin R. W. Ingle[†]

[Abstract of ORNL/TM-7940,[‡] August 1981; also *Nucl. Sci. Eng.* **80**, 603 (1982)]

The ORNL large liquid scintillator detector was used in a precise determination of $\bar{\nu}_p$, the number of neutrons emitted promptly, from spontaneous fission of ^{252}Cf . Measurements of the detector efficiency over a broad energy region were made by means of a proton-recoil technique employing the ORELA "white" neutron source. Monte Carlo calculations of the detector efficiency for a spectrum representative of ^{252}Cf fission neutrons was calibrated with these elaborate measurements. The unusually flat response of the neutron detector resulted in elimination of several known sources of error. Experimental measurement was coupled with calculational methods to correct for other known errors. These measurements lead to an unusually small estimated uncertainty of 0.2% in the value obtained, $\bar{\nu}_p = 3.773 \pm 0.007$.

*Research sponsored by U.S. DOE Office of Basic Energy Sciences.

[†]Instrumentation and Controls Division.

[‡]Title of ORNL/TM-7940 is *Absolute Measurement of $\bar{\nu}_p$ for ^{252}Cf Using the ORNL Large Liquid Scintillator Neutron Detector*.

1.12. NEUTRON CAPTURE CROSS SECTION OF NEPTUNIUM-237*

L. W. Weston J. H. Todd[†]

[Abstract of *Nucl. Sci. Eng.* **79**, 184 (1981)]

The neutron capture cross section of neptunium-237 was measured from 0.01 eV to 200 keV. The capture cross section was normalized at 0.0253 eV to a value of 180 ± 6 barns derived from previous total cross section measurements in the resonance region of neutron energies and the shape of the present data from 0.0253 eV to the

resonance region. Resonance parameters were derived for the neutron energy region from 0.01 to 100 eV. Agreement with ENDF/B-V is poor in the thermal region (6.4%), excellent in the resonance region ($\sim 2\%$) except for the 0.491-eV resonance, and good ($\sim 5\%$) in the keV neutron energy region. An uncertainty analysis including a correlation matrix for group averaged cross sections is presented. These results are important both for thermal and fast reactor applications and the calculation of Pu-238 production, an intense alpha emitter.

*Research sponsored by U.S. DOE Office of Reactor Research and Technology.

†Instrumentation and Controls Division.

1.13. YIELDS OF FISSION PRODUCTS PRODUCED BY THERMAL-NEUTRON FISSION OF ^{249}Cf *

J. K. Dickens J. W. McConnell

[Abstract of Phys. Rev. 24(1), 192 (1981)]

Absolute yields have been determined for 107 gamma rays emitted in the decay of 97 fission products representing 54 mass chains created during thermal-neutron fission of ^{249}Cf . These results include 14 mass chains for which no prior yield data exist. Using a Ge(Li) detector, spectra were obtained of gamma rays emanating from a 0.4 μg sample of ^{249}Cf between 45 sec and 0.4 yr after very short irradiations of the ^{249}Cf by thermal neutrons. On the basis of measured gamma-ray yields and known nuclear data, total chain mass yields and relative uncertainties were obtained for 51 masses between 89 and 156. The absolute overall normalization uncertainty is $\sim 8\%$. The measured A-chain cumulative yields make up 77% of the total light mass ($A \leq 123$) yield and 79% of the total heavy mass yield.

The results are compared with fission-product yields previously measured, with generally good agreement. Information on the charge distribution along several isobaric mass chains was obtained by determining fractional yields for 11 fission products and combining these results with other measurements. The charge distribution width parameter for the heavy masses, $A = 128$

to 140, is independent of mass to within the uncertainties of the measurements. For the light masses, $A = 89$ to 112, the charge distribution parameter is also independent of mass but is smaller than for the heavy masses. Total chain yields are in fair agreement with the current evaluation for ^{249}Cf .

*Research sponsored by U.S. DOE Office of Basic Energy Sciences.

1.14. FISSION PRODUCT YIELDS FOR THERMAL-NEUTRON FISSION OF CURIUM-243*

D. G. Breederland†

(Abstract of ORNL/TM-8168, January 1982)

Cumulative fission yields for 25 gamma rays emitted during the decay of 23 fission products produced by thermal-neutron fission of ^{243}Cm have been determined. Using Ge(Li) spectroscopy, 33 successive pulse-height spectra of gamma rays emitted from a 77-ng sample of ^{243}Cm over a period of approximately two and one-half months were analyzed. Reduction of these spectra resulted in the identification and matching of gamma-ray energies and half-lives to specific radionuclides. Using these results, 23 cumulative fission-product yields were calculated. Only those radionuclides having half-lives between 6 hours and 65 days were observed. Prior to this experiment, no fission-product yields had been recorded for ^{243}Cm .

*Research sponsored by U.S. DOE Office of Basic Energy Sciences.

†Hope College, Holland, Michigan.

1.15. CALCULATED BETA-RAY SPECTRA FROM DECAY OF FISSION PRODUCTS PRODUCED BY THERMAL-NEUTRON FISSION OF ^{235}U *

J. K. Dickens

(Abstract of Phys. Lett. B, in press)

Beta-ray spectra from decay of fission products created by thermal-neutron fission of ^{235}U have been calculated using summation techniques

and using an independently created beta-ray data file. The calculated beta-ray spectra are in very good agreement for $E_\beta \approx 1$ to 8 MeV with recently measured spectra of beta rays for two very different exposure and counting time conditions: (a) an "instantaneous" spectrum following a 1.5-day exposure, and (b) an 8-s counting-time spectrum obtained 1.7 s following a 1-s exposure of ^{235}U by thermal neutrons.

*Research sponsored by U.S. DOE Office of Energy Research.

1.16. BETA AND GAMMA DECAY-HEAT MEASUREMENTS FOR FAST AND THERMAL REACTORS, PARTICULARLY FOR Pu-239*

J. K. Dickens

[Summary of a paper presented at the 24th meeting of the Nuclear Energy Committee on Reactor Physics (NEACRP) held at the United Kingdom Atomic Energy Authority's Establishment, Winfrith, Dorset, September 14-18, 1981]

At the end of an LWR core life, the decay heat will be about one-half due to fission of plutonium and about one-fifth of that will be due to ^{241}Pu . The importance of plutonium decay heat has been the stimulation for four recent experiments. A detailed comparison of the decay-heat experimental data from these four experiments is given in Nucl. Sci. Eng. 78, 126 (1981). We conclude that the differences among the four data sets (of up to 20%) are not due to differences in absolute normalization, but rather are due to the fundamental characteristics associated with each of the total energy-release measuring systems.

Two recent laboratory reports from M. Akiyama *et al.* of the University of Tokyo (written in Japanese) present data for fast-fission decay heat, and have been translated. Their results, expressed as decay heat following an infinite irradiation by fast neutrons for both ^{235}U and ^{239}Pu , are equal to our results for thermal fission of both of these isotopes to within either set of uncertainties.

The mounting evidence suggests (a) that the ORNL thermal-neutron fission decay heat data are likely correct to within stated uncertainties; (b) present data files [ENDF/B-V and others related to it, e.g., A. Tobias and B. S. J. Davies,

"UKFPDD-2: A Revised Fission Product Decay Data File in ENDF/B-IV Format," RD/B/N4942 (Central Electric Generating Board) November 1980] require more effort, and as improvements are made agreement with ORNL results is likely to become better; and (c) "fast" fission decay heat, induced by reactor-spectrum neutrons, appears to be little, if any, different from thermal-neutron fission-decay heat.

*Research sponsored by U.S. Department of Energy.

1.17. YIELDS OF SHORT-LIVED FISSION PRODUCTS PRODUCED BY THERMAL-NEUTRON FISSION OF THORIUM-229*

J. K. Dickens J. W. McConnell
K. J. Northcutt†

[Abstract of Nucl. Sci. Eng. 80, 455 (1982)]

The absolute yields of 39 fission products representing 30 different mass chains produced by thermal-neutron fission of ^{229}Th and having half-lives between 15 and 4600 s have been determined using Ge(Li) spectroscopy methods. Spectra of gamma rays emitted in the decay of the fission products between 25 and 2400 s after a 15-s irradiation were obtained. Gamma rays were assigned to the responsible fission products by matching gamma-ray energies and half-lives. Fission-product yields were then obtained from the data by first determining the appropriate gamma-ray activity as of the end of the irradiation, correcting for detector efficiency and gamma-ray branching ratio, and, finally, dividing by the number of fissions created in the sample.

The resulting fission-product yields are compared with previous measurements and with recommended yields given in the recent U.S. ENDF/B evaluation. Relative uncertainties assigned to the present results range between 6 and 65%, with an absolute normalization uncertainty of 13%. The present uncertainties are smaller than or comparable to uncertainties assigned to previous experimental or evaluated yields for 16 mass chains.

*Research sponsored by U.S. DOE Office of Energy Research.

†Analytical Chemistry Division.

1.18. YIELDS OF FISSION PRODUCTS PRODUCED BY THERMAL-NEUTRON FISSION OF THORIUM-229*

J. K. Dickens J. W. McConnell

(Abstract of Phys. Rev. C, in press)

Absolute yields have been determined for 47 gamma rays emitted in the decay of 37 fission products representing 25 mass chains created during thermal-neutron fission of ^{229}Th . Using a Ge(Li) detector, spectra were obtained of gamma rays emitted between 15 min and 0.4 y after very short irradiations by thermal neutrons of a 15- μg sample of ^{229}Th . On the basis of measured gamma-ray yields and known nuclear data, yields for cumulative production of 37 fission products were deduced. The absolute overall normalization uncertainty is <8%. The results are compared with fission-product yields previously measured with generally good agreement. On this basis, and using other measurements for masses not observed in the present experiment, a complete mass distribution for A between 76 and 152 was deduced. The measured A-chain cumulative yields from the present program make up 84% of the total light-mass ($A \leq 115$) yield and 77% of the total heavy-mass yield.

The data were analyzed to obtain values of most-probable charge (Z_p) and charge-dispersion (σ) parameters. Based upon insight gained from study of similar data obtained for thermal-neutron fission of ^{235}U , we postulate a simple functional dependence $\sigma = \sigma(Z_p)$, and using this dependence obtain values of $Z_p(A)$ for 15 mass chains created during fission of ^{229}Th . Values of $Z_p(A)$ were estimated for other mass chains based upon results of a recent study of $Z_p(A)$. Charge distributions determined using the deduced mass distribution and the deduced sets of $Z_p(A)$ and $\sigma(Z_p)$ are in very good agreement with recent measurements, exhibiting a pronounced even-odd effect in elemental yields. These results may be used to predict unmeasured yields for ^{229}Th fission.

*Research sponsored by U.S. DOE Office of Basic Energy Sciences

1.19. ENERGIES AND INTENSITIES OF GAMMA RAYS FROM DECAY OF ^{229}Th AND DAUGHTERS IN EQUILIBRIUM*

J. K. Dickens J. W. McConnell

(Abstract of Radiochem. Radioanal. Ltrs. 47(6), 331 (1981))

Energies and intensities of 52 γ -rays having energies between 31 and 1567 keV have been measured and compared with previous measurements for ^{229}Th , ^{225}Ra , ^{225}Ac , ^{221}Fr , ^{217}At , ^{213}Bi , and ^{209}Tl . Two new excited states of ^{213}Po have been postulated.

*Research sponsored by U.S. DOE Office of Basic Energy Sciences.

1.20. TPASS, A GAMMA-RAY SPECTRUM ANALYSIS AND ISOTOPE IDENTIFICATION COMPUTER CODE*

J. K. Dickens

(Abstract of ORNL-5732, March 1981)

The gamma-ray spectral data-reduction and analysis computer code TPASS is described. This computer code is used to analyze complex Ge(Li) gamma-ray spectra to obtain peak areas corrected for detector efficiencies, from which are determined gamma-ray yields. These yields are compared with an isotope gamma-ray data file to determine the contributions to the observed spectrum from decay of specific radionuclides. A complete FORTRAN listing of the code and a complex test case are given.

*Work sponsored by Nuclear Regulatory Commission, Electric Power Research Institute and U.S. DOE Office of Basic Energy Sciences.

1.21. TESTS ON A DIGITAL NEUTRON-GAMMA PULSE SHAPE DISCRIMINATOR WITH NE213*

Z. W. Bell

(Abstract of Nucl. Instrum. Methods 188, 105 (1981))

A fast digital technique using two gated charge sensitive analog to digital converters to do neutron-gamma pulse shape discrimination is

reported. The converters are gated by short (135 ns) pulses so as to reduce pile-up and the timing is such that the slow and total light output from the scintillator are sampled. Preliminary tests indicate that the system performs reasonably well but poorer than analog systems employing gated integrators or cross over techniques.

*Research sponsored by U.S. DOE Office of Basic Energy Sciences.

1.22. ISOLR - AN ISOMETRIC PLOTTING PACKAGE FOR USE WITH ORPLOT*

Z. W. Bell

(Abstract of ORNL/TM-8192, May 1982)

An isometric plotting package is described. It is written entirely in FORTRAN and is designed to work in conjunction with ORPLOT, a general purpose plotting package in use at the ORELA PDP-10. The isometric plotter supports views from any point in space, logarithmic scales, and hidden lines.

*Research sponsored by U.S. DOE Office of Basic Energy Sciences.

1.23. ORELA ELECTRON GUNS*

O. W. Christian T. A. Lewis[†]

(Abstract of ORNL/TM-7886, September 1981)

This report contains the most recent information concerning the production and performance of ORELA electron guns. Included are descriptions of procedures for gun fabrication, cathode conditioning and high-voltage processing. Highlights of the performance characteristics are also included.

*This work sponsored by U.S. DOE Office of Nuclear Sciences.

[†]Instrumentation and Controls Division.

1.24. NEUTRON RESEARCH AND FACILITY DEVELOPMENT AT THE OAK RIDGE ELECTRON LINEAR ACCELERATOR 1970-1995*

**R. W. Peelle J. A. Harvey[†] F. C. Maienschein
L. W. Weston D. K. Olsen D. C. Larson
R. L. Macklin[†]**

(Abstract of ORNL/TM-8225, in press)

This report reviews the accomplishments of the first decade of operation of the Oak Ridge Electron Linear Accelerator (ORELA) and discusses the plans for the facility in the coming decade. Motivations for scientific and applied research during the next decade are included. In addition, ORELA is compared with competing facilities, and prospects for ORELA's improvement and even replacement are reported. Development efforts for the next few years are outlined that are consistent with the anticipated research goals. Recommendations for hardware development include improving the electron injection system to give much larger short-pulse currents on a reliable basis, constructing an Electron Beam Injector Laboratory to help make this improvement possible, continuing a study of possibly replacing the electron accelerator with a proton machine, and replacing or upgrading the facility's data-acquisition and immediate-analysis computer systems. Increased operating time and more involvement of nuclear theorists are recommended, and an effective staff size for optimum use of this unique facility is discussed. A bibliography of all ORELA-related publications is included.

*Research sponsored by U.S. DOE Office of Basic Energy Sciences.

[†]Physics Division.

Cross-Section Analyses, Evaluations, and Reviews

1.25. A CALCULATION OF NEUTRON AND GAMMA-RAY PRODUCTION CROSS SECTIONS FOR CALCIUM FROM 8 TO 20 MeV*

D. M. Hetrick[†] C. Y. Fu

(Abstract of ORNL/TM-7752, June 1981)

Nuclear model calculations used for the previous evaluation for calcium (ENDF/B-V MAT 1320) lacked the effects of precompound particle emission. Experimental information now available suggests that these effects are significant from 8 to 20 MeV. New calculations were made to include such effects in all reaction cross sections from 8 to 20 MeV. The resulting data set was prepared in the ENDF/B-V format and is referred to as Material 4152 Modification V in the DNA cross-section library.

*Research sponsored by the Defense Nuclear Agency.

[†]UCC-ND Computer Sciences Division.

1.26. SUMMARY OF ENDF/B-V EVALUATIONS FOR CARBON, CALCIUM, IRON, COPPER, AND LEAD AND ENDF/B-V REVISION 2 FOR CALCIUM AND IRON*

C. Y. Fu

(Abstract of ORNL/TM-8283 (ENDF-325), in press)

This report, together with documents already published, describes the ENDF/B-V evaluations of the neutron and gamma-ray-production cross sections for carbon, calcium, iron, copper, and lead and the ENDF/B-V Revision 2 evaluations for calcium and iron.

*Research jointly sponsored by Defense Nuclear Agency and U.S. DOE Reactor Research and Technology Division.

1.27. AN EVALUATION OF THE RESOLVED-RESONANCE-REGION CROSS SECTIONS OF ²³²Th*

D. K. Olsen

(Abstract of ORNL/TM-8056 (ENDF-319), March 1982)

An evaluation of the neutron-induced resolved-resonance-region (5 to 4000 eV) cross

sections of ²³²Th is described in terms of explicit resonance parameters and pointwise cross sections. The evaluated average capture width, 24.4 ± 2.0 meV, and the 0-to-4-keV s-wave strength function, $0.826 \pm 0.37 \times 10^{-4}$, are appreciably larger than those of the ENDF/B-V evaluation. The present evaluation, together with the ENDF/B-V thermal and unresolved-resonance region cross sections, gives a dilute capture resonance integral of 86.1 b. The bound levels of ENDF/B-V are retained and pointwise cross sections are given to smoothly connect this evaluation to the ENDF/B-V thermal cross sections.

*Research sponsored by U.S. DOE Office of Reactor Research and Technology.

1.28. PAPIN: A FORTRAN-IV PROGRAM TO CALCULATE CROSS SECTION PROBABILITY TABLES, BONDARENKO AND TRANSMISSION SELF-SHIELDING FACTORS FOR FERTILE ISOTOPES IN THE UNRESOLVED RESONANCE REGION*

J. L. Munoz-Cobos[†]

(Abstract of ORNL/TM-7883, August 1981)

The Fortran-IV code PAPIN has been developed to calculate cross-section probability tables, Bondarenko self-shielding factors and average self-indication ratios for non-fissile isotopes, below the inelastic threshold, on the basis of the ENDF/B prescriptions for the unresolved resonance region. Monte-Carlo methods are utilized to generate ladders of resonance parameters in the unresolved resonance region, from average resonance parameters and their appropriate distribution functions.

The neutron cross-sections are calculated by the single-level Breit-Wigner (SLBW) formalism, with s, p and d-wave contributions. The cross-section probability tables are constructed by sampling the Doppler-broadened cross sections.

The various self-shielded factors are computed numerically as Lebesgue integrals over the cross-section probability tables.

The program PAPIN has been validated through extensive comparisons with several deterministic codes.

*This work sponsored by U.S. DOE Office of Reactor Research and Technology.

[†]O₂ assignment from the University of Valencia, Valencia, Spain.

1.29. SENSITIVITY COEFFICIENTS FOR THE ²³⁸U NEUTRON CAPTURE SHIELDED GROUP CROSS SECTIONS*

J. L. Munoz-Cobos[†] G. de Saussure
R. B. Perez

(*Rep't of Trans. Am. Nucl. Soc.*, 8, 666, 1981)

In the unresolved resonance region, cross sections are represented with statistical resonance parameters. The average values of these parameters are chosen to fit evaluated infinitely dilute group cross sections.¹ This procedure is not unique; several equally reasonable sets of average resonance parameters will fit well the evaluated infinitely dilute group cross sections.² The sensitivity of the shielded group cross sections to the choice of mean resonance data has recently been investigated for the case of ²³⁵U and ²³⁹Pu by Ganesan³ and by Antsipov *et al.*⁴ In this paper, we report on similar sensitivity studies for ²³⁸U. These sensitivities are important in evaluating uncertainties in computed Doppler coefficients of reactivity and central worth due to the parameterization of the unresolved resonance range³ and in adjusting group cross sections with integral experiments.^{5,6}

For this study, the computer program PAPIN was developed. This code constructs cross-section probability tables and shielded group cross sections for several dilutions and for several temperatures. The shielded cross sections are calculated using the Bondarenko formula and averaged by a Lebesgue integral over the probability table.⁷

Several studies were performed. As an example, in Fig. 1.29.1 we show the quantity

$$F_1(T, E, \sigma_0) = \frac{\delta \langle \sigma_{n\gamma} \rangle}{\langle \sigma_{n\gamma} \rangle} \frac{\langle \Gamma'_n \rangle}{\delta \langle \Gamma'_n \rangle},$$

Fig. 1.29.1. The Sensitivity Coefficient (see definition in text) Versus the Dilution Cross Section (b) for ²³⁸U Neutron Capture Shielded Group Cross Sections at 300 K for 4-, 10-, and 20-keV Neutron Energies.

where $\delta \langle \sigma_{n\gamma} \rangle / \langle \sigma_{n\gamma} \rangle$ represents the relative change in the shielded average capture cross section at temperature T , energy E , and dilution σ_0 corresponding to a relative change $\delta \langle \Gamma'_n \rangle / \langle \Gamma'_n \rangle$ in the average p-wave neutron width. For the data shown in Fig. 1.29.1, the capture widths were kept constant and the average s-wave neutron width was readjusted to keep the infinitely dilute capture cross section constant and equal to the ENDF/B-V value. The data of Fig. 1.29.1 are for three typical energies with $T = 300$ K.

If we assume a 5% uncertainty in the value of the average p-wave neutron width at 20 keV, the sensitivity coefficients of Fig. 1.29.1 suggest a corresponding 4% uncertainty in the shielded average capture cross section for a dilution $\sigma_0 = 10$ b. Hence, realistic estimates of the uncertainties in the average resonance parameters and of the covariance between them are required to properly assess the uncertainties in shielded group cross sections and thus in the calculations of the Doppler coefficient of reactivity. As was pointed out by several authors, similar sensitivity studies and uncertainty estimates are also required for any meaningful adjustment study in the unresolved resonance region.^{5,6} Clearly, the effect shown in this work is of relevance to the adjust-

ment process as it affords an extra degree of freedom, in that the group cross sections can be readjusted, within certain limits, without alteration of the differential data.³

In summary, a technique is presented to study the sensitivity of shielded group cross sections to the parameterization of the unresolved resonance region, using the probability table approach. This technique is illustrated with a study of the sensitivity of the shielded ^{238}U capture cross section to the assumed value of the average s-wave and p-wave neutron widths.

*Research sponsored by U.S. DOE Office of Reactor Research and Technology.

†On assignment from the Universidad Politecnica de Valencia, Valencia, Spain.

1. This technique is described in some detail by E. Pennington in File I of the ENDF/B-V evaluation of ^{238}U , MAT 1398.
2. G. de Saussure and R. B. Perez, *NBS Special Publication* 425, p. 371. National Bureau of Standards (1975).
3. S. Ganesan, *Nucl. Sci. Eng.*, 74, 49 (1980).
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6. J. H. Marable, C. R. Weisbin, and G. de Saussure, *Nucl. Sci. Eng.*, 75, 30 (1980).
7. L. B. Levitt, *Nucl. Sci. Eng.*, 49, 450 (1972); also, D. E. Cullen, *Nucl. Sci. Eng.*, 55, 387 (1979).

1.30. SENSITIVITY OF COMPUTED U-238 SELF-SHIELDING FACTORS TO THE CHOICE OF THE UNRESOLVED AVERAGE RESONANCE PARAMETERS*

J. L. Munoz-Cobos† G. de Saussure
R. B. Perez

[Abstract of *Nucl. Sci. Eng.* 81, 55 (1982)]

The influence of different representations of the unresolved resonances of ^{238}U on the computed self-shielding factors is examined. It is shown that the evaluated infinitely dilute average capture cross section does not provide sufficient information to determine a unique set of unresolved resonance parameters; different sets of

unresolved resonance parameters equally consistent with the evaluated average capture cross section yield significantly different computed self-shielding factors. In conclusion it is recommended that the resolved resonance description of the evaluated ^{238}U cross sections be extended to higher energies, and that thick-sample transmission data and self-indication data be used to improve the evaluation of the unresolved resonance region.

*Research sponsored by U.S. DOE Office of Reactor Research and Technology.

†On assignment from the University of Valencia, Valencia, Spain.

1.31. ^{238}U , ISSUES RESOLVED AND UNRESOLVED*

G. de Saussure A. B. Smith†

[Abstract of a paper presented at the Nuclear Data for Science and Technology Meeting, (Antwerp) Geel, Belgium, September 6-10, 1982]

The interaction of 1 eV to 20 MeV neutrons with ^{238}U is discussed with emphasis on recently resolved and remaining issues relevant to both application need and physical understanding.

The apparent inability of older ^{238}U evaluations to predict the measured ^{238}U capture rate in thermal critical lattices has stimulated several recent precise measurements of the ^{238}U cross sections, reanalysis of older data, and improved evaluations. The uncertainty in the evaluated value of the parameters of the first few large resonances is now estimated to be smaller than 2% for the neutron widths and smaller than 3% for the capture widths. The recent evaluations predict satisfactorily the ^{238}U capture rate in thermal critical lattices.

In the region from 1.5 to 4 keV there are differences of the order of 15%, sometimes larger, between the values of the neutron widths of the main resonances reported by several experimenters or obtained by different evaluators. These systematic discrepancies have been examined by several authors but are not yet fully understood and deserve further research.

Above 4 keV there are only sparse results of resonance analysis and most evaluations adopt a

statistical treatment of the resonance structure. However, an inspection of good resolution data suggests that the energy and neutron width of the most prominent resonances could be determined with reasonable accuracy at least up to 10 keV. Since these large levels control the resonance self-protection effect, it appears that it would be desirable to include them explicitly in evaluations. Recent work indicates that the statistical treatment adopted in many evaluations may not account properly for resonance self-shielding. Some factors affecting the determination of the average properties of the resonance parameters will be discussed.

Above the inelastic-scattering threshold, energy-averaged neutron total, scattering, capture and fission cross sections are reviewed in a unified manner integrating measurement, calculation, and evaluation. (n;n') and (n;2n') energy-transfer mechanisms are addressed including pre-equilibrium contributions. Particular attention is given to neutron capture, stressing precisions consistent with applied need. Fission properties are discussed including: prompt and delayed fission-neutron spectra and nubar, and fragment yields and energetics. Physical understanding is assayed with attention to compound-nucleus and direct-reaction mechanisms, and applications impact is illustrated in the context of fast-breeder-reactor performance.

*Research sponsored by U.S. DOE Office of Reactor Research and Technology.

[†]Argonne National Laboratory, Argonne, Illinois.

1.32. ON THE ENDF/B UNRESOLVED RESONANCE REGION FORMALISM REPRESENTATION*

R. B. Perez G. de Saussure
J. L. Munoz-Cobos[†] J. Barhen
R. Q. Wright[†]

(Reprint of *Trans. Am. Nucl. Soc.* 39, 883, 1981)

A considerable amount of effort has gone into the development of neutron cross-section representations in the unresolved resonance region. The central question about the ENDF/B representation¹ of the unresolved resonance region

is whether or not that representation leads to a correct estimate of resonance self-shielding. Ultimately, this question should be answered by direct comparison of computed self-shielding factors, as a function of neutron energy, temperature, and dilution, with values derived from self-indication measurements.² However, to our knowledge, these tests have not been done with ENDF/B-V.

The purpose of this work is to present a test of the validity of the ENDF/B treatment of the unresolved resonance region, taking the ENDF/B-V ²³⁸U, MAT 1398 (Ref. 3) as an example. In this test, the capture self-shielding factor, f_γ , was computed in the resolved resonance region in the energy interval between 1 and 4 keV by the resolved resonance methodology⁴

$$f_\gamma = \frac{1}{\langle \sigma_{n\gamma} \rangle} \frac{\int \frac{\sigma_{n\gamma}(E)dE}{\sigma_t(E) + \sigma_0}}{\int \frac{dE}{\sigma_t(E) + \sigma_0}} \quad (1)$$

For the purpose of the test, we consider the self-shielding factors, f_γ , as correct. For the second part of the test, the energy range between 1 and 4 keV was divided into 13 intervals of ~250 eV each. The average capture cross section at infinite dilution $\langle \sigma_{n\gamma} \rangle$ for each interval was obtained by the relation

$$\langle \sigma_{n\gamma} \rangle = \frac{1}{\Delta E} \int \sigma_{n\gamma}(E)dE \quad (2)$$

where the integration is over the width ΔE of the interval and where $\sigma_{n\gamma}(E)$ is computed from the resolved resonance parameters with the addition of the smooth file contribution (MF3). Using the values of the average resonance parameters given in Table 1.32.1, appropriate values of the average p-wave reduced neutrons widths were obtained by fitting $\langle \sigma_{n\gamma} \rangle$ defined in Eq. (2), while keeping the remaining resonance parameters at their prescribed values. This procedure was followed to adhere to the ENDF/B prescription for MAT 1398 in the unresolved region. The capture self-shielding factors, f_γ , were then calculated from

Table 1.32.1. ENDF/B-V ^{238}U Unresolved Resonance Parameters in the Interval from 4 to 45 keV

l	J	$\langle D \rangle$ (eV)	$\langle \Gamma_n^0 \rangle$ (eV)	$\langle \Gamma_\gamma \rangle$ (eV)
0	1/2	20	0.0021	0.0235
1	1/2	20	GNI	0.0235
	3/2	10	1/2 GNI	0.0235
2	3/2	10	0.0025	0.0235
	5/2	6.67	0.00167	0.0235

l = neutron angular momentum.

J = spin of compound nucleus level $J = |l \pm 1|$

$\langle D(l, J) \rangle$ = average level spacing

$\langle \Gamma_n^0(l, J) \rangle$ = average reduced neutron width

$\langle \Gamma_\gamma(l, J) \rangle$ = average capture widths.

Eq. (3) below,⁵ and then compared with the corresponding values of f_γ^* :

$$f_\gamma = \frac{1}{\langle \sigma_{n\gamma} \rangle} \left(\frac{\sigma_{n\gamma}}{\sigma_i + \sigma_0} \right) \quad (3)$$

Some results of this comparison are presented at 300 K in Table 1.32.2. The first column of the table gives the neutron energies for which the unresolved resonance parameters were defined.

The second column gives the limits of the energy interval ΔE over which the capture cross section was averaged [see Eq. (2)]. The widths of these intervals were selected to correspond approximately to the corresponding width of MAT 1398 in the range from 4 to 5 keV.

The third column of Table 1.32.2 gives the average capture cross section, computed from Eq. (2), and the fourth column gives the reduced p-wave neutron width for $J = 1/2$, GNI. The next three columns give the value of $\langle \sigma_{n\gamma} \rangle$, f_γ , $\langle \sigma_{n\gamma} \rangle f_\gamma^*$, and their ratio, respectively, for a dilution $\sigma_0 = 10$ b. The last six

Table 1.32.2. Uranium-238 Self-Shielded Capture Cross Sections from 1 to 4 keV (b)

E eV	EL-EH eV	$\langle \sigma_{n\gamma} \rangle$ b	GNI 10^{-3} eV	$\sigma_0 = 10$ b ^a			$\sigma_0 = 50$ b ^a			$\sigma_0 = 100$ b ^a		
				$\sigma_{n\gamma} f$	$\sigma_{n\gamma} f^*$	Ratio	$\sigma_{n\gamma} f$	$\sigma_{n\gamma} f^*$	Ratio	$\sigma_{n\gamma} f$	$\sigma_{n\gamma} f^*$	Ratio
1000	900-1100	3.044	9.47	1.075	.852	1.26	1.535	1.257	1.22	1.818	1.525	1.19
1250	1100-1400	2.113	1.00	.651	.676	.96	1.001	1.000	1.00	1.222	1.209	1.01
1500	1400-1600	1.912	2.14	.708	.771	.92	1.026	1.059	.97	1.216	1.231	.99
1700	1600-1800	2.087	6.82	.971	.737	1.32	1.303	1.068	1.22	1.486	1.281	1.16
1950	1800-2100	1.442	5.52	.558	.742	.75	.818	.941	.87	.965	1.045	.92
2250	2100-2400	1.610	4.33	.824	.797	1.03	1.081	1.042	1.04	1.215	1.178	1.03
2500	2400-2600	1.640	5.93	.915	.671	1.36	1.169	.919	1.27	1.295	1.070	1.21
2750	2600-2900	1.256	2.24	.660	.802	.82	.865	.955	.91	.970	1.032	.94
3000	2900-3100	.985	2.54	.464	.712	.65	.646	.827	.78	.738	.878	.84
3250	3100-3400	1.455	6.07	.911	.932	.98	1.123	1.133	.99	1.220	1.231	.99
3500	3400-3600	1.216	3.71	.755	.646	1.16	.933	.823	1.13	1.016	.921	1.10
3750	3600-3800	1.099	2.80	.681	.659	1.03	.843	.795	1.06	.919	.872	1.05
3900	3800-4000	1.094	2.08	.699	.699	1.00	.856	.836	1.02	.927	.909	1.02
900-2100				3.963	3.779	1.05	5.683	5.325	1.07	6.707	6.291	1.07
2100-3100				2.863	2.983	.96	3.761	3.743	1.00	4.218	4.158	1.01
3100-4000				3.046	2.936	1.04	3.755	3.587	1.05	4.082	3.933	1.04
900-4000				9.872	9.698	1.02	13.199	12.655	1.04	15.007	14.425	1.04

^a All calculations shown correspond to a temperature of 300 K.

columns give the corresponding information for dilutions $\sigma_0 = 50$ b and $\sigma_0 = 100$ b. At the bottom of the table, the self-shielded cross sections have been added over intervals of 1 and 3 keV, respectively.

We see from Table 1.32.2 that over 250-eV intervals the values of f_T , obtained from the unresolved resonance statistical model, may differ appreciably from the values of f_T^* : for a dilution of 10 b, the difference between f_T^* and f_T exceeds 30% in 25% of the intervals. For a dilution of 50 b, it exceeds 20% again in 25% of the intervals. Of course, when the self-shielded group cross sections are summed over wider intervals, of 1 or 3 keV, the differences between the resolved and unresolved calculations become relatively less important; nevertheless, even for intervals of 1 keV, differences on the order of 5% remain.

In conclusion, it is particularly noteworthy that over 1-keV intervals, the unresolved resonance model appears to overestimate systematically the self-shielded capture cross section. If it can be assumed that in the interval from 4 to 10 keV, the discrepancies are of the same magnitude and same sign as in the interval from 1 to 4 keV, then this effect could explain, at least in part, the longstanding discrepancy between measured and computed values of the capture rate in ^{238}U -containing assemblies.

Our test was limited to MAT 1398, but it should be clear that validation is also needed for the ^{235}U and ^{239}Pu unresolved representations.^{6,7} In fact, the fissile isotopes unresolved range starts at a much lower energy (82 eV for ^{235}U and 300 eV for ^{239}Pu), where self-shielding effects might be more important.

*Research sponsored by U.S. DOE Office of Reactor Research and Technology.

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²UCC-NU Computer Sciences Division.

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1.33. REPRESENTATION OF THE NEUTRON CROSS SECTIONS OF SEVERAL FERTILE AND FISSILE NUCLEI IN THE RESONANCE REGIONS*

G. de Saussure R. B. Perez

[Reprint of *Trans. Am. Nucl. Soc.* 38, 662, 1981; work also described in ORNL/TM-7945 (ENDF-312), September 1981 and in *Ann. Nucl. Energy* 9, 79 (1982)]

In this critical review, we discuss some problems associated with the measurement and representation of neutron cross sections of fertile and fissile nuclei in the resolved and unresolved resonance regions. We restrict our attention to the heavy nuclei, which are most important for nuclear reactor applications; the resonance structure of the light- and medium-weight nuclei, used for moderators and structural materials, has different characteristics and requires a different approach. In fact, we concentrate on five specific nuclei -- fertile ^{232}Th and ^{238}U and fissile ^{233}U , ^{235}U , and ^{239}Pu -- that play a major role in practical applications and have resonance structures somewhat typical of the actinides.

We discuss some of the experimental problems with neutron cross-section measurements and some of the ambiguities in the resonance analysis of such data resulting from the use of different resonance formalisms and different treatments of "far-away" levels.

The evaluator must review several data sets and the resonance parameters obtained from their analysis and determine an appropriate set of

parameters for inclusion in the ENDF/B evaluation. Since the cross sections are often analyzed with different formalisms and almost always with procedures different from those of ENDF/B, this evaluation process has often led to unfortunate results, such as the ENDF/B specification of "negative cross sections" or of a "smooth" File-III contribution with considerable structure. Much progress has been accomplished recently toward understanding and resolving such problems, but the representation of the resolved region in ENDF/B-V is still inadequate.

In the unresolved resonance region, ENDF/B does not specify cross sections, but defines statistical distributions of fictitious unresolved resonance parameters. These distributions can be used to obtain most probable values and associated probability distributions for average dilute or self-shielded group cross sections as a function of temperature. The methods used by evaluators to generate the average value and distribution of the unresolved resonance parameters are largely arbitrary and ambiguous. Several authors have recently discussed the problems and inconsistencies of such a probabilistic representation of the cross sections and suggested other approaches. We believe that what is most needed are precise self-indication and Doppler effect measurements to test the several possible representations of the cross sections in the unresolved resonance region.

In recent years, much effort has been expended generating covariance files to specify the uncertainties in the ENDF/B cross sections. These covariance files are needed to adjust these cross sections, so that both differential and integral measurements can be taken into account in generating a cross-section library. At the present time, no satisfactory method to specify a covariance file for the resolved and unresolved resonance parameters has been found, but several recent developments appear promising.

In summary, much progress in the accuracy of cross-section measurements and their analysis has been made; nevertheless, many problems remain with the ENDF/B-V representation of the cross

sections of fertile and fissile nuclei in the resolved and unresolved resonance regions. Some of the important tasks remaining are as follows:

1. A better evaluation of the resolved resonance parameters using the best data presently available, particularly for ^{238}U .
2. A better definition of the logical and theoretical basis for the unresolved resonance region representation.
3. The collection and use of good and simple self-indication measurements and Doppler effect benchmarks to test resonance representations.
4. The development of reliable uncertainty covariance matrices for the resolved and unresolved resonance parameters.

*Research sponsored by U.S. DOE Office of Reactor Research and Technology.

1.34. COMPARISON OF THE ENDF/B-V AND SOKRATOR EVALUATIONS OF ^{235}U , ^{239}Pu , ^{240}Pu , AND ^{241}Pu AT LOW NEUTRON ENERGIES*

G. de Saussure R. Q. Wright[†]

(Abstract of a paper presented at the Joint IAEA/NEA Consultants Meeting on Uranium and Plutonium Resonance Parameters, Vienna, September 28 - October 2, 1981).

The U.S. and U.S.S.R.'s most recent evaluations of ^{235}U , ^{239}Pu , ^{240}Pu , and ^{241}Pu are compared over the thermal region and over the first few resonances. The two evaluations rest on essentially the same experimental data base and the differences reflect different approaches to the representation of the cross sections or different weightings of the experimental results. It is found that over the thermal and resolved ranges the two evaluations are very similar. Some differences in approaches are briefly discussed.

*Research sponsored by U.S. DOE Office of Reactor Research and Technology.

[†]UCC-ND Computer Sciences Division.

1.35. REVIEW OF (n,n',γ) CROSS-SECTION DATA ON ACTINIDE NUCLEI*

D. K. Olsen

(Abstract of a paper presented at the Specialists Meeting on Fast Neutron Scattering on Actinide Nuclei, Paris, France, November 23-25, 1981)

The available high-resolution (n,n',γ) cross section data on actinide nuclei, essentially ^{238}U and ^{232}Th , are reviewed by dividing the measurement process into two steps. First, the measurement of discrete photon production cross sections is discussed by comparing white and monoenergetic neutron source techniques. Resultant cross sections are compared and their accuracies assessed. Second, the step of inferring inelastic scattering cross sections from photon production cross sections is discussed with particular emphasis on the problems connected with monopole transitions, low-energy transitions, prompt fission photons, and simple rotational models. The uncertainties introduced in this step are probably larger than those of the first.

*Research sponsored by US DOE Office of Basic Energy Sciences.

1.36. REVIEW OF FAST-NEUTRON CAPTURE CROSS SECTIONS OF THE HIGHER PLUTONIUM ISOTOPES AND Am-241*

L. W. Weston

(Abstract of a paper presented at the NEANDC/NEACRP Specialists Meeting on Fast-Neutron Capture Cross Sections, Argonne National Laboratory, Argonne, IL, April 20-23, 1982)

The fast-neutron capture cross sections of Pu-240, 241, 242, and Am-241 are reviewed. These nuclides are important to core physics of reactors that contain Pu-239. There have been several significant measurements of these cross sections in recent years. These measurements were instigated by the need for these cross sections for reactor calculations involving high burn-up and build-up of the higher actinides. These recent measurements have satisfied the urgent need for these cross sections in the context of the accuracy

needed relative to those of the major fissile isotopes. Problems that exist in the experimental measurements and their evaluation are discussed.

*Research sponsored by U.S. DOE Office of Reactor Research and Technology.

1.37. EXPERIENCE IN USING THE COVARIANCES OF SOME ENDF/B-V DOSIMETRY CROSS SECTIONS : PROPOSED IMPROVEMENTS AND ADDITION OF CROSS-REACTION COVARIANCES*

C. Y. Fu D. M. Hetrick†

(Summary of a paper presented at the Fourth ASTM-EURATOM Symposium on Reactor Dosimetry, Washington, DC, March 27-28, 1982)

Recent ratio data, with carefully evaluated covariances, were combined with eleven of the ENDF/B-V dosimetry cross sections using the generalized least-squares method. The purpose was to improve these evaluated cross sections and covariances, as well as to generate values for the cross-material covariances. Certain problems were encountered, but not unexpectedly. These problems were traced to three areas of deficiencies in the covariances of the ENDF/B-V evaluations. First, some of the ratios calculated from the preliminary output evaluation lie outside the interval defined by the ratio data and the ratios calculated from the ENDF/B-V evaluations. Some examples are: ratios of $^{46}\text{Ti}(n,p)$, $^{54}\text{Fe}(n,p)$, and $^{58}\text{Ni}(n,p)$ to $^{238}\text{U}(n,f)$ near 10 MeV. These results, though theoretically possible, suggest unrealistically large correlation in some of the input evaluations. Second, an unduly large energy range of full correlation enhances propagation of discrepancies. For example, both $^{46}\text{Ti}(n,p)$ and $^{238}\text{U}(n,f)$ have a fully correlated range from 4 to 10 MeV, so that a local discrepancy between ratio data and evaluations near 4 MeV propagates all the way up to 10 MeV. Third, changes in evaluated cross-section values larger than one standard deviation are present in most energy ranges. This implies that uncertainty estimates were generally too small in the ENDF/B-V evaluations and possibly in the ratio data too. For example, the output

$^{63}\text{Cu}(n,\alpha)$ cross sections from 8 to 11 MeV changed from the input by three standard deviations. Such large changes, if due to an oversight in one evaluation, would adversely affect other evaluations through the ratio data and/or through the cross-material correlations. Remedies to these three deficiencies are being sought and tested. Success of this effort would lead to substantially improved cross sections as well as realistic and usable covariances.

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†UCC-ND Computer Sciences Division.

1.38. MAJOR QUESTIONS ABOUT DERIVATION OF VARIANCE-COVARIANCE INFORMATION FOR NUCLEAR DATA EVALUATIONS*

R. W. Peelle

[Abstract of a paper to be presented at Nuclear Data for Science and Technology, (Antwerp), Geel, Belgium, September 6-10, 1982]

The uncertainties in and correlations among evaluated nuclear data must also be evaluated to permit estimation of data-related uncertainties in design quantities and to focus future studies to improve the data. This paper summarizes the major questions that now arise in trying to obtain numerical files of this uncertainty information ("covariance data"), and points toward likely answers. Covariance data must be sufficient to construct the variance-covariance matrix of the evaluated cross sections.

Up to now the most important question has usually been how to combine differential data that are discrepant or for which experiment uncertainties are incompletely reported. Evaluators must be thoughtfully bold in discarding or reweighting data sets; therefore authors may wish to report covariance data on some prior experiments to assure their continued use.

On other questions, (1) covariance data may be required in more detail for use in future evaluations than for use in engineering applications, (2) it is yet unclear whether few-group covariance representations for structural materials will be adequate for both core physics and deep

penetration shielding analysis, (3) data evaluations performed with the aid of nuclear theory must also include output covariance data based at least on the covariance matrix of the model parameters, and (4) a consensus is still lacking on the role of integral data in the evaluation of nuclear cross sections and their uncertainties.

*Research sponsored by U.S. DOE Office of Reactor Research and Technology.

1.39. MEETING CROSS SECTION REQUIREMENTS FOR NUCLEAR ENERGY DESIGN*

C. R. Weisbin D. Gilai[†]
G. deSaussure R. T. Santoro

[Abstract of ORNL/TM-8220 (ENDF-327), in press]

The purpose of this report is to summarize and explain current requirements in cross section data that are essential to nuclear energy programs and to provide some insight into how these data might be obtained. The paper is divided into six sections that describe:

1. Design parameters and target accuracies,
2. Data collection, evaluation, and analysis,
3. Determination of high accuracy differential nuclear data for technological applications,
4. Status of selected evaluated nuclear data,
5. Analysis of benchmark testing, and
6. Identification of important cross sections and inferred needs.

It is generally accepted that the uncertainties in the prediction of reactor properties can result in excessive design margins and, as a result, costs. There are, however, only a few good published studies that demonstrate the consequences of uncertainties in nuclear data. Significant complexity is introduced in such studies when factors such as feedback of operating data, design flexibility in accommodating errors, calibration of prototypic experiments, methods and modeling uncertainties, etc., are included in a systematic way. The cross-section data, and the uncertainties in these data, have changed dramatically in recent years and the necessary updating must be completed and made available. High accuracy

measurements often require several years of planning, data collecting, and analyses. Also, the needs of nuclear energy programs change and as data requirements are met, new problems identify different and new data needs. An attempt is made in this report to update and identify nuclear data requirements and to summarize the state of the data as of 1981.

*Research sponsored by U.S. Department of Energy.

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1.40. CROSS SECTION NEEDS: DETERMINATION AND STATUS*

C. R. Weisbin D. Gilai†

G. de Saussure R. T. Santoro

(Summary of paper to be presented at ANS Topical Meeting on Advances in Reactor Physics and Core Thermal Hydraulics, Kiamesha Lake, New York, September 22-24, 1982)

It is widely recognized that uncertainties in the prediction of reactor properties results in excessive design margins and ultimately cost.^{1,2,3,4} However, there are only a few good published studies which demonstrate the consequences of uncertainties in nuclear data. Significant complexity is introduced in such studies when factors such as feedback of operating data, designer flexibility in accommodating error, calibration to prototypic experiments, methods and modeling uncertainties, etc. are included in a systematic way. For fast reactors, the fundamental work of Grebler¹ is now more than ten years old and an update is sorely needed. For thermal reactors, the recent investigation of Becker and Harris³ shows substantial implication of nuclear data uncertainties on the light water reactor fuel cycle. Critical review of such important conclusions by vendor and utility organizations would be most useful.

It appears that target accuracies for fast reactor core physics are being approached by existing estimated uncertainties.^{2,5} Note, however, that the fast reactor program in the United States still awaits construction of its first demonstration reactor, and commercialization has not really begun. In some sense, assessment of the situation for light water reactors is more difficult since

plant specific information of this type is considered proprietary. Crowther⁶ notes increased accuracy requirements as the stimulus for continued LWR physics evolution and improvement; apparently the enormous monetary incentive to reduce design conservatism still warrants continued progress in the data area. The situation is much more clear with respect to fusion. Because fusion needs for test facilities emphasize an energy range (a few MeV to ~50 MeV) which has not received priority in the past, there is a need for total cross section data for ⁷Li, C, Ni, Ca, O, Si, Cu, ¹¹B, Fe, Cr and Pb to accuracies of the order of 10-15%. Gas production cross sections of Fe, Ni, Cr, V, Ti, Nb and Cu as well as information on secondary energy and angular distributions would be useful if attained to ~20-50% accuracy. However, the nuclear data needs for fusion engineering prototype and demonstration power plant designs are not immediate and would likely not be characterized as top priority.

Because of the interrelationship of different types of differential cross section data and of experimental data and nuclear models, it is often desirable to improve a broad data base rather than address measurements to narrow specific requirements. Indeed, since differential cross section measurements of high accuracy can take from 5-7 years, there are those who suggest that specific reactor problems should be addressed through the use of appropriate integral measurements while differential cross sections are developed to treat unforeseen problems and those problems which cannot currently be satisfactorily addressed with integral measurements. This view is balanced by those who believe that the Department of Energy's nuclear data program should be to support the needs of reactor design and to minimize the role of integral experiments to a confirmatory mode. Both of these views manifest themselves in the latest U.S. compilation of requests for nuclear data,⁷ which is now under review by the Cross Section Evaluation Working Group. It seems obvious that a better definition of need would be facilitated by the existence of high quality uncertainty files for nuclear data. Current information on uncertainties in the

ENDF/B file is very incomplete and often unreliable; improvement of the quantity and quality of this data should be a major goal for ENDF/B-VI.

The adequacy of the Evaluated Nuclear Data File is tested, for several applications, against results of benchmark integral experiments. These benchmarks currently test only a limited domain of important nuclear data (e.g. there is a need for fast reactor burnup benchmarks, high temperature experiments, etc.) and the quality of the experimental uncertainties of these benchmarks is very uneven. The discrepancies between calculated and measured integral quantities may point to data problems, but could also be due to methods or modeling approximations or to errors in the integral benchmark measurement. Only for the light water reactor TRX series was there any attempt by CSEWG to formally estimate methods and data uncertainties separately.⁸ Much more effort in this area is sorely needed in order to derive specific recommendations from the data testing process.

Important differences encountered in data testing^{8,9} still persist in several areas. For example

(A) In fast assemblies, the ^{238}U capture to ^{239}Pu fission ratio is overpredicted by $\sim 5\%$; the ^{239}Pu and ^{235}U central worths are also overpredicted by $\sim 10\text{--}20\%$. Spectral indices in small fast assemblies range from 8-10% underprediction in JEZEBEL to 2-4% overprediction in GODIVA.

New information which would permit computation of the ^{238}U group cross sections to $\sim 3\%$ in the range 1-30 keV for a variety of self-shielding conditions would clearly be helpful. Reassessment of the ^{238}U and ^{239}Pu neutron emission spectra (neutron energy distribution from fission and inelastic), including estimated uncertainties, would contribute greatly toward the JEZEBEL and GODIVA analysis.¹⁰

(B) In thermal spectra, the ^{235}U capture to fission resonance integral ratio is overpredicted, as is the effective criticality constant in $\text{Pu-H}_2\text{O}$ assemblies.

Measurement of the resonance parameters of ^{235}U over the range 0.1-1.0 keV to obtain fission

and capture resonance integrals and to resolve discrepancies in recent fission measurements seems appropriate.

(C) For shielding applications, problems are perceived¹¹ with the total cross section of C, Ni, and Cr, and with the capture gamma ray production in Cl and Cu below 6 MeV. In addition, many materials have no evaluation of photon production data.

Additional needs for fusion, dosimetry, fission products and actinides are described in the full text.

*Research sponsored by U.S. Department of Energy.

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1.41. NUCLEAR FISSION AND NEUTRON-INDUCED FISSION CROSS-SECTIONS*

G. D. James[†] A. Michaudon[‡]
J. E. Lynn[†] J. Rowlands[§]
G. de Saussure

(Vol 1 of *Neutron Physics and Nuclear Data in Science and Technology*, A. Michaudon, Ed., Pergamon Press, Oxford, 1982; the following is an excerpt from the introduction to Chapter IV, "Measurements of Fission Cross-Sections," by G. D. James and G. de Saussure)

Accurate measurements of neutron-induced fission cross-sections for nuclei over a wide range of atomic mass numbers are required for the calculation of nuclear reactor reactivity and behaviour (see Chapter III). The analysis and interpretation of fission cross sections is carried out in terms of basic nuclear physics considerations which entail an understanding of nuclei in highly-deformed states. The neutron energy range of interest is dominated by the energy range of neutrons in thermal and fast nuclear reactors and extends from below 0.025 eV over the thermal neutron spectrum, through the resonance region and up to about 20 MeV. This energy range is likely to be further extended as the data needs of fusion reactors, fission-fusion hybrid reactors and for the accelerator breeding of fissile material become established. For the last, a knowledge of neutron-induced fission reactions may be required up to several hundred MeV to understand the processes which occur in the so-called intra-nuclear cascade produced by the collisions of 1 to 2 GeV protons with heavy nuclei. The data needed include measurements over neutron spectra of known energy distribution, such as the thermal and fast reactor spectra, and also measurements made with good neutron energy resolution. The latter are carried out either with monoenergetic sources or by the powerful neutron time-of-flight technique which employs pulses of short duration containing neutrons over a wide energy range - the so-called white source spectrum. The time-of-flight technique has produced a vast amount of detailed information on neutron cross-sections over a wide energy range. It is effectively complemented by

the other, monoenergetic and spectrum average techniques, which are subject to entirely different experimental and background conditions.

In this chapter we give a resume of the steps involved in the measurement of fission cross-sections.

*ORNL participation sponsored by DOE Division of Reactor Research and Technology.

[†]Atomic Energy Research Establishment, Harwell, United Kingdom.

[‡]Commissariat à l'Energie Atomique, France.

[§]United Kingdom Atomic Energy Authority.

1.42. REPORT TO THE DOE NUCLEAR DATA COMMITTEE*

J. C. Lundy F. G. Perey

(Abstract of ORNL/TM-7786, April 1981; also BNL-NCS-29426, p. 120 (May 1981))

This report was prepared for the DOE Nuclear Data Committee and covers work performed at ORNL since April 1980 in areas of nuclear data of relevance to the U.S. applied nuclear energy program. The report was 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 presentation at technical conferences. As much as possible the complete abstract of the original publication has been reproduced with only minor editing. In a few cases progress reports were written specifically for this publication. The editors have selected the material to be included in this report on the basis of perceived interests of DOE Nuclear Data Committee members.

*Research sponsored by U.S. DOE Office of Basic Energy Sciences.

1.43. STATUS REPORT TO THE DOE NUCLEAR DATA COMMITTEE*

F. G. Perey

(Abstract of data submitted for inclusion in Brookhaven National Laboratory report)

This report was prepared for the DOE Nuclear Data Committee and covers work per-

formed at ORNL since December 1980 in areas of nuclear data of relevance to the U.S. applied nuclear energy program. The report was 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 presentation at technical conferences. As much as possible we have repro-

duced the complete abstract of the original publication with only minor editing. In a few cases progress reports were written specifically for this publication. The editors have selected the materials to be included in this report on the basis of perceived interests of DOE Nuclear Data Committee members and cannot claim completeness.

*Research sponsored by U.S. DOE Office of Basic Energy Sciences.

Section 2

**METHODS FOR GENERATING AND VALIDATING MULTIGROUP
CROSS-SECTION LIBRARIES**

2.0. INTRODUCTION

The Engineering Physics Division is continuing its long-term program of processing and testing multigroup cross-section libraries for use in computer codes that calculate radiation transport. It also continues to package the libraries as DLCs (Data Library Collections) for distribution to users through the Radiation Shielding Information Center (RSIC), which remains a part of the division (see Section 8). While the development of our major processing systems was largely accomplished several years ago, the maintenance of these large systems includes periodic improvements. In addition, some staff members are actively involved in improving other cross-section processing systems and/or cross-section sets for specific requestors, particularly for the utilities, usually through the Electric Power Research Institute. Finally, in recent years the division has become involved in the processing and distribution of cross-section covariance (uncertainty) files as the data become available.

Processing Cross Sections for Studies in Fast Reactor Physics, Fusion Neutronics, Shielding, and Weapons Radiation Transport. -- The MINX-AMPX processing systems described in our earlier progress reports have now been tested through a relatively long production period. The systems have produced multigroup cross-section libraries for use in radiation transport calculations in fast reactor physics and shielding, fusion neutronics and shielding, light-water reactor shielding, and weapons radiation transport. As noted in our last report, the MINX system as used at ORNL contains improvements in the area of resolved resonance reconstruction, linearization, Doppler broadening, unresolved resonance treatment, shielding of scattering transfer matrices, and development of elastic removal shielding factors. Output from MINX is linked to AMPX through the use of the library format of the latter and through the subsequent generation, with AMPX modules LAPAN-GAS and SMUG, of gamma-ray production and interaction cross sections.

Processing Cross Sections for Criticality Studies. -- The NPTXS, ROLAIDS, and XLACS-II modules of AMPX are used for generating neutron multigroup libraries for criticality studies, the libraries having thermal-neutron region upscattering. For some thermal applications, the NJOY system is used.

Processing SAFE Files. -- Processing some of the Special Application Files (SAFE) of ENDF/B-V has required some changes to the processing systems because the SAFE files are "incomplete." For example, reaction cross sections are given, but no angular or energy distributions are provided. Modifications have been incorporated into XLACS to accommodate each file.

Covariance File Processing. -- The PUFF-II code has been upgraded to accommodate new formats and to reduce the running time needed to generate multigroup covariance data.

Improving Cross-Section Processing Techniques for the Utility Industry. -- During the period covered by this report one project in the division has consisted in upgrading EPRI-CELL, a lattice physics code that is employed by the utilities to generate the cell-averaged cross sections for diffusion theory core calculations. The work has included adapting EPRI-CELL to use ENDF-V cross sections and also improving techniques used within the code. Among other improvements, the code now can account for the overlap of resonances of different absorbers and it utilizes a new Dancoff factor expression that accounts for clad interactions. Benchmarking calculations have also been performed for EPRI-CELL using point cross-section Monte Carlo and discrete ordinates methods. The new version of EPRI-CELL will be distributed to the utilities this summer.

Recent Cross-Section Libraries. -- Major production efforts in cross-section processing have resulted in the development of two fine-group libraries based on ENDF/B-V data. The first, VITAMIN-E, contains 174 neutron groups and 38 gamma-ray groups and was generated with the MINX-AMPX system for fast reactor core and shielding analyses, fusion neutronics calculations, weapons radiation transport, and light-water reactor shielding analysis. Available in both AMPX and CCCC formats, VITAMIN-E will serve as the basis for deriving problem-dependent broad-group libraries for particular applications.

The second fine-group library, CSRL-V, was generated with AMPX for criticality safety applications and contains 227 neutron groups. Both VITAMIN-E and CSRL-V are sequels to earlier libraries based on ENDF/B-IV data (VITAMIN-C and CSRL).

A library identified as FLUNG (for Fusion Library for Unshielded Neutron-Gamma-Ray Transport) was also derived from VITAMIN-C. In addition, three libraries, not developed within the division but being distributed by RSIC, have been produced for light-water reactors (84 neutron groups), CANDU reactors (27 groups), and LMFBR reactors (126 groups).

Baffle/Reflector Library for Utilities. -- While the utilities can generate their own cross sections for diffusion theory core calculations, they cannot easily generate cross sections for components outside the core. ORNL, in conjunction with EPRI, used a method to produce cross sections for the baffle and reflector surrounding PWR cores, which when used in diffusion theory calculations will give the same leakage results as other transport calculations. The cross sections have been distributed to the utilities.

Recent Covariance Libraries. -- Multigroup covariance libraries for fast reactor and LWR pressure vessel dosimetry applications have been produced with PUFF-II from ENDF/B-V covariance files. These are DLC-77/COVERV (with 52 neutron groups) and DLC-90/DOSCOV (with 24 neutron groups).

ENDF/B-V Data Testing. -- Finally, the division has participated in a program coordinated by the Cross Section Evaluation Working Group (CSEWG) to test the adequacy of ENDF/B-V cross sections for fast and thermal reactor design applications. Data from VITAMIN-E and from FORSS-V (an undocumented ORNL 100-group library) processed by the MINX code were used to calculate eigenvalue results for 12 fast reactor benchmarks. The results were compared with results obtained at Argonne and Los Alamos National Laboratories.

Several thermal reactor benchmarks have been analyzed and submitted to a cooperative, CSEWG-sponsored effort to determine the accuracy of ENDF/B-V data for predicting eigenvalues and reaction rate ratios.

Future Directions. -- The development of VITAMIN-E will continue through feedback from the user community, with additional materials added as needs arise. Broad-group libraries for Defense Nuclear Agency radiation transport and for LMFBR shielding applications will be developed, and modifications will be made to ORNL processing systems to accommodate evaluated data in new formats. Data in new formats are expected in the fusion program and will apply for neutron interactions with incident energies up to 50 MeV.

It is anticipated that the EPRI-sponsored lattice physics work will end during CY-82, and that the ORNL effort will be directed toward methods and data development for LWR safety analysis required by utilities in performing reload design calculations.

2.1. FLUNG: COUPLED 35-GROUP NEUTRON AND 21-GROUP GAMMA RAY, P_3 CROSS SECTIONS*

R. T. Santoro R. W. Roussin J. M. Barnes[†]

(Abstract of ORNL/TM-7828, June 1981)

A 35-neutron, 21-gamma ray, P_3 cross section library for 63 nuclides is described. The library, called FLUNG (Fusion Library for Unshielded Neutron-Gamma-Ray Transport), was derived from the VITAMIN-C general purpose fine-group library and is useful for the neutronic analysis of fusion reactors and other applications.

*Work sponsored by U.S. DOE Office of Fusion Energy.

[†]UCC-ND Computer Sciences Division.

2.2. EVALUATION OF ENDF/B-V-BASED THREE-FAST-GROUP REFLECTOR CONSTANTS*

W. J. Eich[†] M. L. Williams C. Peng[†]

(Reprint of Trans. Am. Nucl. Soc. 38, 666, 1981)

In the course of cross-section benchmarking, analysis of critical configurations generally requires the use of few-fast-group diffusion reflector constants that provide accurate leakage over a wide range. These constants should be derived from a higher multigroup calculation in a manner that ensures their validity over a range of leakage (core size) and geometry. This paper summarizes the development, testing, application to experiment, and slight empirical modification of such constants derived ultimately from the ENDF/B-V data set.

A systematic procedure, used previously,^{1,2} has been now adapted to codes currently available to develop a three-fast-group generic representation. This set, intended for the analysis of critical configurations at standard temperature and pressure (STP) and at higher³ temperatures, has been generated from the analysis of a single high-leakage infinite slab fuel-water configuration, one for each water density, using as a reference the XSDRN multigroup discrete ordinates code.⁴ The reduction of the XSDRN output to the final three-fast-group constants also requires a modified version of the FOG few-group diffusion

code,⁵ and REKON, a simple iterative routine used previously.² Subsequent comparisons between assorted FOG runs with this explicit spatial reflector representation and geometrically equivalent XSDRN runs (which throughout utilized 106 fast groups, one thermal group, and an S8 order) have validated the three-fast-group constants for two slab thicknesses (leakages), three cylindrical configurations of widely varying leakage, and for two different water densities. In addition, the effort includes a borated slab case with STP water. The explicit validation against higher order calculation for cylindrical geometry had not been possible in the previous application^{1,2} of the general approach.

The procedure utilizes a one-dimensional slab FOG run with input fuel constants derived from the four-group edit of a B-1 point XSDRN calculation, in turn based on a unit cell homogenization. The reflector fast-group representation is developed in two steps. In the first, the reflector is represented by $\Delta\phi/\phi$ boundary conditions at the fuel/reflector interface. A single four-group calculational series of runs is made with version of FOG modified so that the boundary conditions are iteratively modified until the four-group currents at the interface match those edited from the reference XSDRN calculation. The FOG-converged flux values (which differ by 2.5% at most from the XSDRN values), the common value of the currents, the XSDRN reflector-side group one of four removal reaction rate and three-fast-group absorption rates, and the reflector thermal group constants form the bases for the input to the REKON calculation. The latter solves for the appropriate removal and diffusion coefficients.⁶

Table 2.2.1 summarizes the eigenvalue comparisons between the various XSDRN and FOG runs. For Case 3, the slight modification of the fast-group constants required in accounting for the presence of soluble boron utilizes a consistent adjustment of absorption and removal in groups two and three. The FOG values calculated for the cylindrical configurations are extremely consistent, even though no adjustment of the few-group diffusion coefficients as a function of radius has been made.

Table 2.2.1. Eigenvalue (λ) Comparisons*

Case Description	Equivalent Buckling (m^{-2})	STP Water % $\Delta\lambda/\lambda$	81% of STP $\Delta\lambda/\lambda$
1. 10-cm slab	75	+ .10 ^a	+ .06 ^a
2. 15-cm slab	50	-.03	-.05
3. 15-cm slab 800 ppm B in reflector	50	+ .04	--
4. 18-cm cylinder	90	-.02	+ .06
5. 20-cm cylinder	75	-.08	-.01
6. 27-cm cylinder	50	-.10	-.08

$$\frac{\Delta\lambda}{\lambda} = \frac{\lambda_x - \lambda_f}{\lambda_x}; \text{ x denotes XSDRN; f denotes FOG.}$$

^aBase Cases.

Finally, the absolute accuracy of this reflector representation was evaluated by calculating eigenvalues for actual borated critical "endpoint" configurations. Previous studies⁷ have shown that the eigenvalues calculated for these endpoints are typical of the intermediate points (which show little scatter in λ). Eigenvalues were calculated for three sets of borated critical endpoint configurations^{8,9} utilizing the EPRI-CELL code¹⁰ with ENDF/B-V cross sections. These three sets showed a very consistent difference in reactivity as a function of leakage, corresponding to an error of <1% in leakage.

Comparison between this set of reflector constants and those derived previously¹ shows remarkable agreement except in the group one diffusion coefficient. This suggests significant similarity (stability over the years) in the hydrogen and oxygen cross sections corresponding to the two basic multigroup sets. Substitution of the first group D from the older set substantially eliminates the slight leakage discrepancy noted for the endpoints. Efforts are continuing to explain the one difference and many similarities between the two reflector sets.

At the present time, however, fast-few-group cross-section sets required for assorted ENDF/B-V benchmarking are now available based on, as well as verified against, consistent multigroup high-order transport calculations, with only very modest empirical modification localized in one group and one constant.

*Research sponsored by Electric Power Research Institute.

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23. ENDF/B-V-BASED FEW-GROUP PWR BAFFLE/REFLECTOR CONSTANTS*

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M. L. Williams

(Reprint of Trans. Am. Nucl. 39, 907, 1981)

Recently, generic fast few-group light water reflector constants intended for critical experiment analysis¹ have been systematically developed from and verified against multi-fast-group XSDRN transport calculations.² The latter utilized ENDF/B-V cross-section data and fission spectra. This paper discusses the extension of this calculational approach to the generation of baffle and reflector constants for application to the typical configuration at the perimeter of pressurized water reactor (PWR) cores. This extension is substantially complicated by the requirements, not encountered in critical analyses, of generating single as well as three fast-group and coarse-mesh as well as fine-mesh diffusion representations. The single fast-group and coarse-mesh requirements arise from the group and mesh (a unit cell pitch) structure practically required in XY radial plan PDQ reactor core representations.³

Figure 2.3.1 illustrates the range of average assembly powers computed for Comanche Peak-2, upon using three baffle/reflector representations (A, B, and C), derived on the basis of reasonable engineering approximations made in lieu of the availability of a multigroup transport calculation. These three representations depend on edits from unit fuel supercell calculations, edits from infinite baffle and/or water regions, and the like. The results labeled D correspond to application of the representation whose development is discussed below. The close agreement between C and D is fortuitous, since representation C is derived from an infinite region homogenization of a 7/8-in. baffle with 1/8-in. of water, whereas D is based ultimately on the multigroup transport calculation explicitly edited for baffle and reflector.

The unusual combination of loading pattern and enrichment specified for this reactor results

in peak assembly powers at the core periphery, thus exacerbating, in terms of average assembly power, differences among the various baffle representations relative to most other PWR configurations. The motivation for this effort was derived not only from the prospect of a definitive baffle/reflector representation *per se*, but also for its importance relevant to a general upgrading of radial plane PDQ/HARMONY calculations⁴ under the designation of "Pseudax." This refers to normalizing such calculations on an average assembly basis (i.e., resonance temperature, water density) to information edited from an axial plane of a three-dimensional nodal solution incorporating all necessary thermal and neutronic feedback. Thus, this approach is to be strongly distinguished from crude attempts made to account for axial feedback effects on radial plane solutions that have involved arbitrary adjustments such as *ad hoc* modifications to baffle and/or reflector constants.

The basic slab geometry used in the derivation consisted of 15 cm of homogenized fuel, baffles of various thicknesses, and 25 or more cm of 70% standard temperature and pressure water borated to 500 ppm. A modified version of the FOG code⁵ was used to match few-group currents edited from XSDRN at the fuel/baffle and then baffle/reflector interfaces. These searches are performed because few-group diffusion fluxes differ from the corresponding XSDRN few-group edits upon matching FOG currents to the transport code edits. Between performing these interface searches, an iterative Gaussian elimination routine was used to obtain good estimates for the baffle diffusion coefficients, quantities not available from the XSDRN edits. Following the baffle/reflector interface current matching, the REKON iterative routine⁶ solved for the three fast-group reflector constants. Thereupon, FOG runs with explicit representation of baffle and reflector produced eigenvalue agreement ranging between 0.04 and 0.07% Δ , with the XSDRN run. These results correspond to four FOG runs: three and one fast-group, fine-, and coarse-mesh. This work is being extended to the thicker baffles encountered in most PWRs and has also been

.66	.79	.76	1.32	.84	1.12	.95	1.11
.64	.77	.75	1.31	.87	1.11	.95	1.12
.63	.76	.74	.99	.86	1.11	.95	1.14
.62	.74	.72	.97	.85	1.11	.95	1.15
	.72	.96	.84	1.04	.91	1.13	1.08
	.70	.94	.83	1.07	.90	1.13	1.09
	.69	.93	.82	1.07	.90	1.14	1.11
	.68	.91	.81	1.05	.90	1.14	1.12
		.82	1.06	.90	1.15	.94	1.03
		.81	1.04	.89	1.15	.95	1.05
		.80	1.03	.88	1.14	.95	1.06
		.79	1.02	.88	1.14	.95	1.07
			.90	1.09	.97	1.18	.75
			.89	1.09	.97	1.19	.77
			.88	1.08	.97	1.20	.78
			.88	1.08	.97	1.21	.80
				1.12	1.11	1.17	
				1.12	1.12	1.17	
				1.11	1.12	1.20	
				1.12	1.13	1.22	
					1.10	.85	
					1.12	.88	
					1.12	.89	
					1.14	.91	

A	see text
B	
C	
D	

Fig. 2.3.1. Core Octant Assembly Average Powers.

successfully extended to a hybrid baffle/reflector set of constants, i.e., a single set reproducing the combined effects of both regions. Such a hybrid set has possible application in coarse-mesh reactor calculations.

*Research sponsored by Electric Power Research Institute.

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2.4. THE CSRL-V ENDF/B-V 227-GROUP NEUTRON CROSS-SECTION LIBRARY AND ITS APPLICATION TO THERMAL REACTOR AND CRITICALITY SAFETY BENCHMARKS*

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(Abstract of a paper presented at a seminar/workshop on Thermal Reactor Benchmarks, Calculations, Techniques, Results, and Applications, Brookhaven National Laboratory, May 17-18, 1982)

Characteristics and contents of the CSRL-V
(Criticality Safety Reference Library based on

ENDF/B-V data) 227-neutron-group AMPX master and pointwise cross-section libraries are described. Results obtained in using CSRL-V to calculate performance parameters of selected thermal reactor and criticality safety benchmarks are discussed.

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2.5. CORRECTION OF MULTIGROUP CROSS SECTIONS FOR RESOLVED RESONANCE INTERFERENCE IN MIXED ABSORBERS*

M. L. Williams

(Abstract of ORNL/TM-8354, in press)

The effect that interference between resolved resonances has on averaging multigroup cross sections is examined for thermal reactor-type problems. A simple and efficient numerical scheme is presented to correct a preprocessed multigroup library for interference effects. The procedure is implemented in a "design oriented" lattice physics computer code and compared with rigorous numerical calculations. The approximate method for computing resonance interference correction factors is applied to obtaining fine-group cross sections for a homogeneous uranium-plutonium mixture and a uranium oxide lattice. It was found that some fine group cross sections are changed by more than 40% due to resonance interference. The change in resonance interference correction factors due to burnup of a PWR fuel pin is examined and found to be small. The effect of resolved resonance interference on collapsed broad-group cross sections for thermal reactor calculations is discussed. It is found that the ^{238}U and ^{235}U epithermal cross sections are fairly insensitive to interference effects, but the ^{239}Pu value increases about 3.5% and the ^{240}Pu value decreases by more than 7% for a PWR pin.

*Research sponsored by Electric Power Research Institute.

2.6. IMPROVEMENTS IN EPRI-CELL METHODS AND BENCHMARKING OF THE ENDF/B-V CROSS SECTION LIBRARY*

M. L. Williams R. Q. Wright[†] J. Barlow

(Abstract of EPRI Report NP-2416, in press)

This report documents in-house modifications made by Oak Ridge National Laboratory (ORNL) to the lattice physics code EPRI-CELL, and subsequent benchmarking of the new ENDF/B-V EPRI-CELL cross section library for UO_2 lattices. The code modifications were made mainly in the THERMOS and GAM portions of the program, and resulted in improved reactor physics methods and more efficient running times. Benchmark calculations done on a variety of UO_2 criticals show that the ENDF/B-V library used in conjunction with the modified version of the code gives excellent results for the lattice eigenvalues. The average value of k_{eff} for the asymptotic critical lattices is about 0.4% higher with the ENDF/B-V library than the average eigenvalue obtained using the originally released library. Monte Carlo results obtained by ORNL and Brookhaven National Laboratory (BNL) are presented for three "numerical benchmark" problems to provide further verification of the EPRI-CELL methods. It was found that EPRI-CELL agrees within 0.3% - 0.4% of the average Monte Carlo values for k_{∞} , and within 0.6% for the ^{238}U epithermal-to-thermal capture ratio.

The ORNL EPRI-CELL code in its present form represents an interim version of the program which was developed to (a) examine possible improvements to the EPRI-CELL methods and (b) be used in benchmarking the new ENDF/B-V library. As such, it will not be officially released by EPRI. However, the modifications made by ORNL have been provided to EPRI to be included, along with other improvements, in a new version of EPRI-CELL which will be officially released to utilities in its finalized form at a later date.

*Research sponsored by Electric Power Research Institute.

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2.7. BENCHMARKING OF EPITHERMAL METHODS IN THE LATTICE PHYSICS CODE EPRI-CELL*

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(Abstract of a paper presented at a seminar/workshop on Thermal Reactor Benchmarks, Calculations, Techniques, Results, and Applications, Brookhaven National Laboratory, May 17-18, 1982)

Most of the LWR analysis codes used by industry rely on relatively simple computational methods to obtain cell-averaged broad-group cross sections for subsequent input to diffusion theory calculations of the core. The computer programs employed in the lattice physics calculations must be economical to use, but still give accurate results for the range of problems encountered in design and fuel management decisions. The Advanced Recycle Methodology Program (ARMP)¹ is a system of computer codes distributed by the Electric Power Research Institute (EPRI) to utilities for performing core-follow and reload calculations. The PWR lattice physics module in this system is called EPRI-CELL (E-C).² The function of this program is to read fine-group, problem-independent libraries for the fast and thermal energy ranges, to perform resonance shielding and cell averaging for a pin cell, and then to collapse the multigroup cross sections to a coarse group structure for diffusion theory calculations of an assembly or some fraction of the reactor core. The change in the pin composition due to burnup is also taken into account, so that the final output of E-C is a set of lattice-dependent broad-group cross sections tabulated as a function of burnup (and possibly other parameters such as soluble boron concentration).

Recently EPRI has embarked on a program to upgrade the thermal, fast, and depletion E-C libraries to use ENDF/B-V data.³ In conjunction with this effort, a review was made of the lattice physics approximations used in E-C, which resulted in several code modifications, and the new version of E-C was benchmarked for various criticals and "numerical benchmarks" that had been computed with continuous-energy Monte Carlo codes.⁴

While the critical analysis provides useful comparisons of calculational and experimental eigenvalues and in some cases integral ratios, it does not totally eliminate the possibility of a fortuitous cancellation of data and methods errors in E-C. Benchmarking against Monte Carlo results based on the same basic cross-section data helps in exposing methods deficiencies in E-C; however, the comparisons must be made on relatively coarse parameters such as broad-group (e.g., four-group) cross sections to obtain meaningful Monte Carlo statistics. The numerical benchmark calculations have shown that the most common source of disagreement between E-C and Monte Carlo results is in the range from approximately 1 eV - 5 keV, which contains ²³⁸U resolved resonances.

The purpose of the present study is to benchmark E-C resonance-shielding and cell-averaging methods against a rigorous deterministic solution on a fine-group level (~30 groups between 1 eV and 5.5 keV). The benchmarking code used is OZMA,⁵ which solves the space-dependent slowing-down equations using continuous-energy discrete ordinates or integral transport theory to produce fine-group cross sections. Results are given for three water-moderated lattices - a mixed oxide, a uranium metal, and a tight-pitch high-conversion uranium oxide configuration. The latter two lattices were chosen because of the strong self shielding of the ²³⁸U resonances. Particular emphasis is placed on examination of three E-C approximations: (a) neglect of resonance interference effects between ²³⁵U and ²³⁸U, (b) use of the intermediate resonance approximation for resonance shielding, and (c) use of equivalence theory for treating heterogeneous effects. An attempt is made to isolate and quantify differences between the OZMA and E-C fine-group cross sections due to the above approximations.

*Research sponsored by Electric Power and Research Institute.

[†]UCC-ND Computer Sciences Division.

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2.8. DEVELOPMENT OF IMPROVED METHODS FOR THE LWR LATTICE PHYSICS CODE EPRI-CELL*

M. L. Williams R. Q. Wright[†] J. Barben

(Abstract of ORNL/TM-8411, in press)

A number of improvements have been made by ORNL to the lattice physics code EPRI-CELL (E-C) which is widely used by utilities for analysis of power reactors. The code modifications were made mainly in the thermal and epithermal routines and resulted in improved reactor physics approximations and more efficient running times. The improvements in the thermal flux calculation included implementation of a group-dependent rebalance procedure to accelerate the iterative process and a more rigorous calculation of interval-to-interval collision probabilities.

The epithermal resonance shielding methods used in the code have been extensively studied to determine its major approximations and to examine the sensitivity of computed results to these approximations. The study has resulted in several improvements in the original methodology. These include: treatment of the external moderator source with intermediate resonance (IR) theory, development of a new Dancoff factor expression to account for clad interactions, development of a new method for treating resonance interference, and application of a non-linear optimization method to compute "best-estimate" values for the Bell factor and group-dependent IR parameters. The modified E-C code with its new ENDF/B-V

cross section library is tested for several "numerical benchmark" problems. Integral parameters computed by E-C are compared with those obtained with point-cross section Monte Carlo calculations, and E-C fine group cross sections are benchmarked against point-cross section discrete ordinates calculations. It is found that the code modifications improve agreement between E-C and the more sophisticated methods. E-C shows excellent agreement on the integral parameters and usually agrees within a few percent on fine-group, shielded cross sections.

*Research sponsored by Electric Power Research Institute.

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2.9. INCORPORATION OF CLAD EFFECTS INTO SAUER'S METHOD FOR COMPUTING DANCOFF FACTORS*

M. L. Williams D. Gilai[†]

(Abstract of *Ann. Nucl. Energy*, Vol. 9, 137 (1982))

A simple method is presented to modify Sauer's approximation for the lattice Dancoff factor to account for clad interactions. It is shown that for some tight pitch lattices the difference in Dancoff factor with and without clad present may cause up to 1/2% difference in k_{eff} . The present technique has the advantages of (a) being easy to implement into existing codes, (b) having negligible impact on execution time, and (c) showing good agreement with sophisticated methods.

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2.10. FAST REACTOR DATA TESTING OF ENDF/B-V AT ORNL*

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J. H. Marable

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The Cross Section Evaluation Working Group (CSEWG) is coordinating a program to assess the adequacy of ENDF/B-V cross sections for

both fast- and thermal-reactor design applications. A secondary goal is to evaluate cross-section processing codes, cross-section libraries, and radiation-transport codes. Fast reactor data testing (FRDT) goals are accomplished, in part, by comparison of calculated results with documented performance parameters of CSEWG fast reactor benchmarks¹⁻³ and with results obtained by other data testers. The purpose of this paper is to describe the results of FRDT at Oak Ridge National Laboratory (ORNL).

Data from two ENDF/B-V processed multigroup cross-section libraries, VITAMIN-E and FORSS-V, were used in the benchmark calculations. VITAMIN-E is an evolving 174-neutron, 38-gamma group library applicable to fusion reactor neutronics, fast reactor core and shield analysis, weapons analyses, and LWR shielding and dosimetry problems.^{4,5} FORSS-V is an undocumented ORNL 100-neutron-group library for FRDT *per se*. Neutron data in the libraries were processed with the MINX code.⁶ Resonance self-shielding calculations for the benchmarks were made with BONAMI-2;^{7,8} one-dimensional transport calculations were made with either XSDRNPM^{7,8} or ANISN.⁹ Bias factors accounting for modeling approximations were obtained from Refs. 1-3.

ORNL eigenvalue results for 12 benchmarks are given in Table 2.10.1 along with a set of average values which were obtained by averaging results reported by ANL,¹⁰ LANL,¹¹ and ORNL.^b The ORNL C/E (calculated/experimental) values for selected central reaction-rate ratios^c are given in Table 2.10.2. Central reactivity worths have also been obtained for ZPR-6/7, GODIVA, and JEZEBEL.

Results in Tables 2.10.1 and 2.10.2 lead to the following observations, conclusions, and recommendations:

1. The average ORNL eigenvalue for the Pu-fueled benchmarks is 1.0051 and for the U-fueled benchmarks is 1.0037; the average for the 12 benchmarks is 1.0042. Thus, there is essentially no bias between the Pu- and U-fueled benchmarks when all 12 bench-

marks are used as the reference set. When we consider only the two pairs, JEZEBEL:GODIVA and ZPR-6/7:ZPR-6/6A there is a bias of 0.8% between the Pu-fueled and U-fueled benchmarks.

2. The ZPR-6/6A eigenvalue is quite low relative to the other benchmarks; this may indicate a need to re-examine the benchmark specifications and correction factors.
3. k_{eff} for FLATTOP-25 is 0.8% higher than the GODIVA value. The core composition of these two benchmarks is essentially the same; the differences are the core radius and the fact that GODIVA is unreflected while FLATTOP-25 has a reflector. This may indicate that the leakage from GODIVA is overpredicted.
4. k_{eff} of ZPR-9/31 is 1% higher than k_{eff} of ZPR-6/7; this difference may be due to spectral differences and should be investigated.
5. The JEZEBEL F28/F25 C/E is ~8% low and the calculated leakage spectrum is too soft relative to the measured spectrum. These two results may justify a harder fission spectrum.
6. For GODIVA, FLATTOP-25, and the four intermediate spectrum benchmarks, the F28/F25 C/E values are well above unity; the average is 1.06 for the six benchmarks. Since the F28/F25 ratio is very sensitive to the spectrum, the extent to which this bias is due to spectral determining factors (χ , σ_{int}), rather than the fission cross sections, should be investigated.
7. For the U-fueled benchmarks there is a trend for k_{eff} and F28/F25 C/E to increase with increasing ²³⁸U number density.
8. There is a trend for C28/F25 C/E to increase for the softer spectrum assemblies.

Table 2.10.1. FRDT Eigenvalues Based on ENDF/B-V

Benchmark ^a	k_{eff}			
	Measured F28/F25 ^b	Three-Lab Average ^c	ORNL ^d	$\Delta\%$ ^e
Hard Spectrum:				
JEZEBEL	0.2137	1.0053	1.0043	-0.10
GODIVA	0.1647	0.9970	0.9967	-0.03
FLATTOP-25	0.1488	1.0056	1.0046	-0.10
Intermediate Spectrum:				
ZPR-3/6F	0.078	1.0081	1.0061	-0.20
ZPR-3/12	0.047	1.0052	1.0044	-0.08
ZPR-3/11	0.038	1.0121	1.0121	0.0
BIG TEN	0.0373	1.0102	1.0102	0.0
Soft Spectrum:				
ZPR-3/48	0.0321	1.0078	1.0089	0.11
ZPR-3/56B	0.0308	1.0054	1.0056	0.02
ZPR-9/31	0.02896	1.0068	1.0066	-0.02
ZPR-6/6A	0.02388	0.9891	0.9915	0.24
ZPR-6/7	0.02261	0.9974	1.0000	0.26

^aPu-fueled benchmarks include JEZEBEL, ZPR-3/48, ZPR-3/56B, ZPR-9/31, and ZPR-6/7; the others are U-fueled (benchmarks are ordered according to spectral hardness).

^bThe measured ^{238}U fission to ^{235}U fission central reaction-rate ratio is an indication of spectral hardness; the breakpoint between hard and intermediate is 0.1, and between intermediate and soft 0.035.

^cThe "average" is defined as $(\text{ANL} + \text{LANL} + \text{ORNL})/3.0$; for GODIVA and JEZEBEL the ANL-VIM values were also included in the average.

^dORNL results are from VITAMIN-E data with the exception of ZPR-3/6F, ZPR-3/12, ZPR-3/11, ZPR-3/48, and ZPR-3/56B which are from FORSS-V.

^e $\Delta\% = \frac{\text{ORNL} - (3 - \text{Lab Avg.})}{(3 - \text{Lab Avg.})} \times 100.$

9. k_{eff} 's for ZPR-3/11 and BIG TEN are $\sim 1\%$ high. Also the F28/F25 C/E's are overpredicted by about $\sim 7-9\%$. These benchmarks have large ^{238}U concentrations and as a result $\sim 25\%$ of the fissions are from ^{238}U . The high values of k_{eff} and F28/F25 relative to the experimental values are apparently the result of an overprediction of the ^{238}U fission rate. This overprediction appears to be caused by the calculated flux spectrum rather than by the ^{238}U fission cross sections. The BIG TEN calculated flux shape

is not consistent with the measured spectrum. In particular, the calculated flux is high relative to the experiment in the energy range 0.82-3.68 MeV. A relatively small increase (~ 5 to 10%) in the ^{238}U inelastic cross section in this energy range would result in significant improvement in k_{eff} and F28/F25 for the benchmarks.

10. Using ENDF/B-V, the ZPR-6/7 C28/F49 C/E, 1.05, is reduced slightly relative to the value obtained using ENDF/B-IV, 1.07.¹³ The above results summarize the conclusions based on fast reactor data testing of

**Table 2.10.2. ORNL C/E Values for Central Reaction Rate Ratios
Based on ENDF/B-V^{a,b}**

Benchmark ^c	F28/F25 C/E	F49/F25 C/E	C28/F25 C/E	C28/F49 C/E
Hard Spectrum:				
JEZEBEL	0.917	0.973	NC ^d	
GODIVA	1.039	0.996	NC	
FLATTOP-25	1.039	1.002	NC	
Intermediate Spectrum:				
ZPR-3/6F	1.035	1.038	0.930	
ZPR-3/12	1.093	1.013	0.970	
ZPR-3/11	1.089	1.003	0.954	
BIG TEN	1.070	1.005	0.972	
Soft Spectrum:				
ZPR-3/48	1.064	NC	0.976	
ZPR-3/56B	0.989	0.961	NC	
ZPR-9/31 ^e				1.036
ZPR-6/6A	0.987	NC	1.031	
ZPR-6/7	0.961	0.992	1.041	1.050

^a Results are from VITAMIN-E data with the exception of ZPR-3/6F, ZPR-3/12, ZPR-3/11, ZPR-3/48, and ZPR-3/56B which are from FORSS-V. Symbolically, the ^{238}U fission to ^{235}U fission reaction-rate ratio is denoted herein as F28/F25. Likewise, F49 denotes ^{239}Pu fission and C28 denotes ^{238}U capture.

^b The uncertainties in the experimental values (1σ) range from 1 to 5%. The C/E values must be interpreted relative to the uncertainty in the measured value.

^c Pu-fueled benchmarks include JEZEBEL, ZPR-3/48, ZPR-3/56B, ZPR-9/31, and ZPR-6/7; the others are U-fueled (benchmarks are ordered according to spectral hardness).

^d NC, means not calculated because experimental value was not available.

^e The ZPR-9/31 F49/F25 experimental value is presently not considered to be reliable; the C/E values (other than C28/F49) should receive no consideration and have been omitted.

ENDF/B-V at ORNL. Generally, the ORNL results are in good agreement with those of other data testers. One of the highest priorities for future work is a review of the existing CSEWG benchmarks, with specific emphasis on the flux spectra.

^aVITAMIN-E is sponsored by the Department of Energy (DOE) Division of Reactor Research and Technology (DRRT), the Defense Nuclear Agency (DNA), and the Electric Power Research Institute (EPRI).

^bFast reactor data testing results as calculated by the various CSEWG data testers have been collected and reported in Ref. 12.

^cSymbolically, the ^{238}U fission to ^{235}U fission reaction-rate ratio is denoted herein as F28/F25. Likewise, F49 denotes ^{239}Pu fission and C28 denotes ^{238}U capture.

*Research sponsored by U.S. DOE Office of Reactor Research and Technology.

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2.11. NEUTRON CROSS-SECTION LIBRARIES IN THE AMPX MASTER INTERFACE FORMAT FOR THERMAL AND FAST REACTORS*

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[Abstract of ORNL/CSD/TM-164 (ENDF-317), December 1981]

Neutron cross-section libraries in the AMPX master interface format have been created for three reactor types. Included are an 84-group library for use with light-water reactors, a 27-group library for use with heavy-water CANDU reactors, and a 126-group library for use with liquid metal fast breeder reactors. In general, ENDF/B data was used in the creation of these libraries, and the nuclides included in each library should be sufficient for most neutronic analyses of reactors of that type. Each library has been used successfully in fuel depletion calculations.

*Work sponsored by U.S. DOE Waste Management Analysis for Nuclear Fuel Cycles through the ORNL Chemical Technology Division.

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Section 3

METHODS FOR REACTOR AND SHIELD ANALYSES

3.0. INTRODUCTION

The development of methods for computer-based analyses of reactor cores and shields has continued to comprise an important part of the division effort. The work has concentrated less on the development of major codes, however, and more on enhancing current capability by improving the physical modeling in the codes, extending their range of applicability, and introducing other improvements. The division recognizes that with tight coupling of the methods, extensive proof testing, and the establishment of expertise comes reliability in analysis. It also recognizes that the computer codes we develop are widely used by others.

Discrete Ordinates Codes. -- Perhaps the most widely used computer code developed within the division is the two-dimensional discrete ordinates code DOT. A new version of this code has recently been released as DOT 4.3. In DOT 4.3 improved techniques are included for accelerating the rate of convergence of problems, and careful documentation has been made available to support its application and to make the contents and calculational techniques known to users. The code is operational on the local IBM computers and on CRAY, CDC, UNIVAC, AMDAHL, and IBM computers at other installations.

DOT 4.3, the one-dimensional discrete ordinates code ANISN, and related codes are used locally in the Discrete Ordinates System (DOS), which also has recently been fully documented. DOS includes a number of auxiliary codes to facilitate graphic displays, cross-section preparation and mixing, data reformatting, etc.

Another Oak Ridge discrete ordinates code, the TDA code (Time-Dependent ANISN) has recently been used in producing benchmark problems that will aid in the testing and evaluation of methods and codes that employ the time-dependent one-dimensional multigroup Boltzmann transport equation with explicit representation of delayed neutrons.

Monte Carlo Codes. -- During this period the three-dimensional Monte Carlo code MORSE-CG was modified so that fusion devices in toroidal geometry could be calculated. The high-energy nucleon-meson transport code HETC was also modified to treat neutrons from fission. As a result, the modified code, called HETFIS, can be used for problems such as studying the effects of using protons from an accelerator to generate fissile material from fertile material or designing facilities that are to produce an intense source of neutrons by using medium-energy protons.

Diffusion Theory Codes. -- The diffusion theory approximation to neutron transport for treating up to three-dimensional problems is applied with the local BOLD VENTURE Computation System. Capability is available for calculating exposure and modeling the reactor history in several ways. These code modules may be used repeatedly in selective order without new input instructions. The system contains the VENTURE and the new VALE neutronics codes released in 1981. Whereas the VENTURE code applies the mesh centered finite-difference formulation, the mesh points in VALE are located on the corners of triangles on a uniform grid for treating hexagonal fuel assemblies while planes of points are located between material interfaces, mesh centered. Increasing the number of leakage surfaces about a point from five in VENTURE to eight in VALE can improve

the modeling of the neutron flux in three dimensions. The new code alternatively admits application of the linear finite-element approximation on planes. Information has been published about techniques that have been developed for error analysis. The calculational methods have been extensively tested and are used to generate results of benchmark quality.

A report was published that condenses the information needed to apply the codes in the BOLD VENTURE system. The PEBBLE code has been added to the system for two-dimensional thermal hydraulic calculations of the Pebble Bed Reactor, a concept under development in Germany. Another addition is the code module FUELMANG for treating refueling, materials accounting, and fuel costing. This capability extends that of the system for assessing the performance of a reactor core over its life, and especially to make comparative evaluations of alternative designs and variation in the mode of operation. The thermal hydraulic and fuel management capability was supported by specific DOE projects (that is, not methods development funding).

Methodology Development for Reactor Surveillance and Diagnostics. -- Under the auspices of the Electric Power Research Institute, the division is working on a methodology (LEPRICON) which will allow reliable estimates to be made of the neutron fluences within a PWR pressure vessel wall on the basis of dosimetry measurements made at various surveillance locations (not within the wall). This methodology is supported by the analysis of integral experiments described in Section 5 of this report.

A problem analyzed for NRC was the inverse conduction problem that arises when experimental measurements are taken in the interior of a body and it is desired to calculate temperature and heat flux values on the surface.

In other work, the division has participated briefly in studies related to the utilization of stochastic signals (reactor "noise") for power reactor diagnostics and surveillance, in particular with the problem of handling efficiently the substantial amount of information involved in the application of methods.

Theoretical Studies. -- Several theoretical studies have been carried out to improve our understanding of systems governed by the Boltzmann transport equation. In one, the concept of a leakage importance function has been developed; this function gives a measure of the relative importance of each source particle in phase space contributing to the leakage. The usefulness of this concept has been illustrated by considering several idealized nuclear systems.

In another study, the finiteness of the number of discrete eigenvalues occurring in neutron transport has been examined and shown to be intimately related to the Legendre moments of the scattering function. Convergence criterion for these moments has been derived, and this criterion is both necessary and sufficient for the discrete eigenvalues to be finite in number.

Another work has analyzed spectral effects that arise from solving the k -, α -, γ -, and δ -eigenvalue formulations of the neutron transport equation. Several general recommendations could thus be offered regarding the use of a particular eigenvalue equation for specific applications.

Finally, theoretical foundations of a new methodology for accurate treatment of resonance absorption in thermal reactor lattice analysis have been developed.

3.1. AN UPDATED VERSION OF THE DOT 4 ONE- AND TWO-DIMENSIONAL NEUTRON/PHOTON TRANSPORT CODE*

W. A. Rhoades R. L. Childs[†]

(Abstract of ORNL-5851, in press)

DOT 4 is designed to allow very large transport problems to be solved on a wide range of computers and memory arrangements. Unusual flexibility in both space-mesh and directional-quadrature specification is allowed. For example, the radial mesh in an R-Z problem can vary with axial position. The directional quadrature can vary with both space and energy group. Several features improve performance on both deep penetration and criticality problems. The program has been checked and used extensively.

*Work sponsored by Defense Nuclear Agency and U.S. DOE Division of Reactor Research and Technology.

[†]UCC-ND Computer Sciences Division.

3.2. DOS: THE DISCRETE ORDINATES SYSTEM*

W. A. Rhoades M. B. Emmett[†]

(Abstract of ORNL/TM-8362, in press)

The Discrete Ordinates System (DOS) determines the flux of neutrons or photons due either to fixed sources specified by the user or to sources generated by particle interaction with the problem materials. It also determines numerous secondary results which depend upon flux and provides various file manipulation techniques. DOS contains a control driver, several subsystems including DOT, ANISN, DOPES, and DOGS, and several file manipulation programs including BNDRYS, RTFLUM, and GIP.

*Research sponsored by Defense Nuclear Agency and U.S. DOE Office of Energy Technology, Division of Reactor Research and Technology.

[†]UCC-ND Computer Sciences Division.

3.3. IMPROVEMENTS IN DISCRETE-ORDINATES ACCELERATION*

W. A. Rhoades

(Reprint of Trans. Am. Nucl. 39, 753, 1981)

Many discrete-ordinates transport codes depend upon variations of the synthesis method¹

to achieve convergence within acceptable cost limits. An approximate process is used between iterations of the high-accuracy process in order to estimate the results of further high-accuracy iterations. Partial-current rebalance (PCR),² diffusion synthetic acceleration,³ and consistent diffusion acceleration (CDA)⁴ are examples of this process.

Previous acceleration routines have concentrated on the iterations which resolve implicitness caused by scattering.⁵ In fact, similar procedures can accelerate the resolution of implicitness due to reflective boundary conditions. In addition, a modest change to the PCR method considerably eases the problem of choosing the proper damping parameters.

The acceleration of boundary flux can be illustrated in terms of the 1-d PCR process, although its application is more general. Where the solution estimate ϕ^{K-1} is input as a guess to the k'th iteration cycle and ϕ^K is the result of the k'th high-accuracy iteration, the PCR method produces ϕ^K , an improved estimate of ϕ^∞ , to be used as a guess for the k+1'th cycle. The intermediate result, $\bar{\phi}^K$, obeys the balance condition:

$$\begin{aligned}\bar{F}_{i+1,j} &= \bar{F}_{i,j+1} + \bar{F}_{i-1,j} - \bar{F}_{i,j-1} \\ &+ V(\sigma^R + \sigma^S)\phi^K \\ &= VS + V\sigma^S\phi^{K-1},\end{aligned}\quad (1)$$

where $\bar{F}_{i+1,j}^K$, for example, is the flow from mesh cell i to the mesh cell i+1 corresponding to flux ϕ^K . This would be a satisfactory solution only if ϕ^{K-1} were very close to ϕ^K . By assuming $F_{i\pm 1,j}^K/\bar{F}_{i\pm 1,j}^K$ proportional to ϕ^K/ϕ^{K-1} , the improved solution can be found by solving:

$$\begin{aligned}&-(\bar{F}_{i,j+1}^K\phi_{i+1}^K/\bar{\phi}_{i+1}^K + \bar{F}_{i,j-1}^K\phi_{i-1}^K/\bar{\phi}_{i-1}^K) \\ &+ [(\bar{F}_{i+1,j}^K + \bar{F}_{i-1,j}^K)/\bar{\phi}^K + V\sigma^R]\phi^K \\ &= -(\bar{F}_{i,j+1}^K + \bar{F}_{i,j-1}^K) + (\bar{F}_{i+1,j}^K + \bar{F}_{i-1,j}^K) \\ &+ V\sigma^R\bar{\phi}^K + V\sigma^S(\bar{\phi}^K - \phi^{K-1}).\end{aligned}\quad (2)$$

Since this does not involve the direction variable, it can be iterated to convergence at relatively modest cost. At convergence, the implicitness caused by the σ^S term in Eq. 1 has been resolved, at least within the accuracy of Eq. 2. It has been traditional to apply the boundary conditions of the high-accuracy process to Eq. 2, i.e.:

$$F_{1,j+1}^K = \alpha F_{1+1,j}^K; \bar{F}_{1,j+1}^K = \alpha \bar{F}_{1+1,j}^K \quad (3)$$

where α is the boundary albedo, and where we use I to denote the boundary value of i . This has the effect of deleting terms in $\bar{F}_{I+1,J}^K$ in Eq. 2 and multiplying terms in $\bar{F}_{I+1,J}^K$ by $(1-\alpha)$.

A closer look, however, reveals that the appropriate boundary condition for Eq. 2 is really

$$\bar{F}_{I+1,J}^K = \alpha \bar{F}_{I+1,J}^{K-1} \neq \alpha \bar{F}_{I+1,J}^K \quad (4)$$

while 3 applies to the improved flow, $\bar{F}_{I+1,J}^K$. Thus, at the I boundary, Eq. 2 becomes

$$\begin{aligned} & -(\bar{F}_{I,J-1}^K - \bar{F}_{I,J-1}^{K-1}) \\ & + \{[(1-\alpha)\bar{F}_{I+1,J}^K + \bar{F}_{I-1,J}^K]/\bar{\phi}^K \\ & + V\sigma^R\} \phi^K \\ & = -(\bar{F}_{I,J-1}^K) + \{[(1-\alpha)\bar{F}_{I+1,J}^K + \bar{F}_{I-1,J}^K \\ & + V\sigma^R\bar{\phi}^K] \\ & + V\sigma^S(\bar{\phi}^K - \bar{\phi}^{K-1}) + (\alpha\bar{F}_{I+1,J}^K - \bar{F}_{I,J+1}^K) \} \quad (5) \end{aligned}$$

Thus, the boundary flow correction plays a role comparable to that of the scattering correction. It will be seen that this correction improves both PCR and CDA results when boundary flow is important to the overall problem.

Table 3.3.1 shows a series of 2-D problems solved by both methods. Savings range from a low of 3% for a large concrete cylinder to more

than 200% for a small steel tube. Surprisingly, a k calculation for a relatively large cylinder of homogenized FBR core material is improved by 5 to 18%, depending upon the boundary condition and rebalance method. CDA is improved more than PCR, although it is still less efficient for these problems, contrary to the usual case. This type of problem is susceptible to roundoff effects, especially on IBM computers, and that may explain the anomaly.

An important limitation of the PCR method is that the user adjust an arbitrary multiplier, X , for maximum efficiency on difficult problems. X traditionally starts at 1.0 for the first iteration and increases by ΔX after each iteration. We have long wished for a sophisticated way of updating X based upon the nature of the problem itself. Lacking that, we now use an unsophisticated way which works very well. Defining

$$e^K = (\bar{\phi}^K - \bar{\phi}^{K-1})/\bar{\phi}^K \quad (6)$$

$$f^K = (\phi^K - \bar{\phi}^K)/\bar{\phi}^K \quad (7)$$

$$\hat{e}^K = \text{value of } e^K \text{ where } |e^K| \text{ is maximum,} \quad (8)$$

$$\hat{f}^K = \text{value of } f^K \text{ where } |f^K| \text{ is maximum,} \quad (9)$$

then the increment ΔX is applied if and only if:

$$\hat{f}^K \hat{e}^K < 0, \quad (10)$$

Table 3.3.1. Effects of Boundary Flux Acceleration on Various Test Problems

Description	Source	Right ^a Bndy. Cond.	Accel. ^b Type	CPU Min W/O Bndy. Accel.	CPU Min With Bndy. Accel.	Speed ^c Improv.	Comments
1.1 14-gp steel rod, 0.3 cm rad x 40 cm hgt 1 x 40 mesh	Fixed, Top Boundary	Cyl.	PCR	.614	.485	27%	Right boundary flow error was also markedly reduced.
1.2 14-gp steel tube, 0.3 cm rad x 40 cm hgt 2 x 40 mesh; 0.15 cm rad bore along centerline	"	"	"	1.216	.653	86%	Self-adjusting damping was very helpful to PCR here.
	"	"	CDA	>3.000	.937	220%	
1.3 14-gp concrete cylinder, 30 cm rad x 100 cm high, 6 x 20 mesh	"	"	PCR	.391	.376	4%	These problems converged
	"	"	CDA	.457	.439	4%	quickly with or without the
	"	$\alpha=.6$	PCR	.367	.357	3%	boundary acceleration.
1.4 3-gp cylinder of FBR core material; K-calculation, 30 cm rad x 100 cm hgt, 6 x 20 mesh	Fission	"	"	.174	.166	5%	K = .829
	"	Refl.	"	.150	.136	10%	K = 1.247
	"	Cyl.	"	.147	.134	10%	K = 1.246
	"	"	CDA	.173	.147	18%	K = 1.246

^a"Cyl." B.C. means reflective in η , but white in μ . ^a $\alpha=.6$ means an albedo of 60%.

^bAll PCR runs used self-adjusting damping.

^c(New Speed - Old Speed)/Old Speed.

or

$$j^k j^{k-1} < 0, \quad (11)$$

or

$$|j^k| > |j^{k-1}| \quad (12)$$

This allows a value of ΔX sufficiently large to solve difficult problems to be built into the code with little impairment of efficiency in problems which need a small value of X for best efficiency.

In summary, the boundary acceleration provides a "no cost" improvement to problems where the boundary flux is slowly convergent. The self-adjusting damping provides new convenience and reliability for users who continue to use PCR acceleration rather than CDA.

*Research sponsored by Defense Nuclear Agency.

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3.4. DOMINO-II, A GENERAL PURPOSE CODE FOR COUPLING DOT-IV DISCRETE ORDINATES AND MONTE CARLO RADIATION TRANSPORT CALCULATIONS*

M. B. Emmett[†]

(Abstract of ORNL/TM-7771, May 1981)

The DOMINO computer program has been rewritten to allow processing of a DOT-IV multiple boundary source tape. The DOT-IV tape can contain either I-boundary sources, J-boundary sources, or both; but in the latter case DOMINO can process only one of the types in a given run.

As with the previous version, there are four types of DOMINO tapes that may be written for use by a Monte Carlo code.

A detailed discussion of the differences in this code and the previous one is included.

*Research sponsored by Defense Nuclear Agency.

[†]UCC-ND Computer Sciences Division.

3.5. TIME-DEPENDENT, ONE-DIMENSIONAL, MULTIGROUP TRANSPORT THEORY BENCHMARK PROBLEMS*

H. L. Dodds[†] W. W. Engle, Jr. T. R. Hill[†]

(Reprint of *Trans. Am. Nucl. Soc.* 39, 477, 1981)

The primary objective in developing the computational benchmark problems described in this paper is to aid in the testing and evaluation of methods and codes that employ the time-dependent, one-dimensional, multigroup Boltzmann transport equation with explicit representation of delayed neutrons. The equations for this mathematical model, in one-dimensional slab geometry with azimuthal symmetry and isotropic scattering, are

$$\begin{aligned} & \frac{1}{v_g} \frac{\partial}{\partial t} \phi_g(x, \mu, t) + \mu \frac{\partial}{\partial x} \phi_g(x, \mu, t) \\ & + \Sigma_f(x, t) \phi_g(x, \mu, t) \\ & = \frac{1}{2} \sum_{g'=1}^{NOG} [\chi_g^p(1 - \beta) \nu \sigma_f^{g'}(x, t) \\ & + \Sigma_f^{g' \rightarrow g}(x, t)] \phi_{g'}(x, t) \\ & + \frac{1}{2} \sum_{i=1}^{NOP} \lambda_i C_i(x, t) \chi_g^i + Q_g(x, \mu, t); \\ & g = 1, 2, \dots, NOG, \end{aligned} \quad (1)$$

$$\begin{aligned} & \frac{\partial}{\partial t} C_i(x, t) + \lambda_i C_i(x, t) \\ & = \beta_i \sum_{g=1}^{NOG} \nu \Sigma_f^g \phi_g(x, t) ; \\ & i = 1, 2, \dots, NOP, \end{aligned} \quad (2)$$

where NOG is the number of energy groups, NOP is the number of delayed precursor families, $\phi_g(x, \mu, t)$ is the angular group flux, $C_i(x, t)$ is the concentration of type i precursor, and the remaining quantities are defined in detail in Ref. 1 or 2.

Nine different problems have been defined for this mathematical model, of which the first three

have been solved by both TIMEX (Ref. 3) and TDA (Ref. 4) and are presented herein. The main emphasis in these problems is the accuracy of the treatment of the time variable; hence, a fixed spatial mesh and a fixed angular quadrature are recommended in the problem definitions. The geometry and material designations are shown in the insert of Fig. 3.5.1. The materials were selected to be representative of a simple liquid-metal fast breeder reactor,⁵ including blanket

Fig. 3.5.1. Total Power Versus Time Calculated by Both TIMEX and TDA for Benchmark Problems 1, 2, and 3 (Problem Geometry Displayed in the Insert).

(zones 1 and 7), core (zones 2, 4, and 6), and a mixture of sodium and control rod materials (zones 3 and 5). The inward-directed angular flux at each external boundary is zero, and the extraneous source, Q , is zero everywhere. Each problem employs two energy groups, six precursor families, and begins with the same steady-state critical condition.

The perturbations for each of the problems were selected to emphasize space-time effects. For the first problem, at time = 0.0 s, the density of the material in zone 2 is increased by 5%, and the density of the material in zone 6 is decreased by 5%, resulting in a step reactivity insertion and, subsequently, a delayed supercritical transient. For the second problem, the pertur-

bations are identical to those of problem 1, except for the percent change in densities, which is 10%, resulting in a prompt supercritical transient. The perturbation for the third problem is somewhat more complicated. Specifically, at time = 0.0 s, the zone 5 material is changed from a mixture of sodium and control rod material to 100% sodium (i.e., control rod ejection); then at time = 10^{-4} s, the zone 3 material is changed from the initial mixture of sodium and control rod material to 100% control rod material (i.e., full insertion of control rod). This perturbation results in a transient that is initially prompt supercritical, then subcritical with very strong spatial effects.

Solutions of these three problems have been obtained using TIMEX and TDA with the recommended spatial mesh (114 intervals) and quadrature (S_4) and with variable time-step sizes. TDA employs a weighted differencing scheme⁶ for both the spatial and time variables (i.e., partially implicit⁷ time differencing) and conventional discrete-ordinates for the angular variable. TIMEX employs a semi-implicit time differencing scheme,⁸ conventional discrete-ordinates for the angular variable, and the spatial variable is treated with a linear discontinuous finite element approach.³

The total reactor power versus time, for each problem as calculated by both codes, is presented in Fig. 3.5.1. The data shown are for the finest time mesh used by each code. The results obtained with TIMEX and TDA are essentially identical for each of the three problems. Thus, we conclude that, based on the preliminary results obtained thus far, these solutions are reasonably accurate solutions of the three problems described herein for the spatial and angular mesh specified in the problem definitions. More detailed results including a computing cost comparison of the two codes, sensitivity of results to time-step size, and flux tilt information for each of the problems will be presented in the full paper and in Supplement No. 3 of the Benchmark Problem Book.⁹

⁵Research sponsored by U.S. Department of Energy.

⁷University of Tennessee.

⁸Los Alamos National Laboratory.

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3.6. TOROIDAL GEOMETRY SUBROUTINES FOR MORSE-CG*

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H. Iida[§] Y. Seki[§] H. Narita[¶]

(Abstract of ORNL/TM-8244, April 1981)

The equations, coding, and procedures that are required to include a torus in the Combinatorial Geometry subroutines of the MORSE-CG code are described. The derivation and solutions of the quartic equation that describes a torus along with additional subroutines and the modifications to existing subroutines required to carry out the transport of neutrons and gamma rays in toroidal geometry are presented. The input requirements and a sample problem are included.

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3.7. HETFIS - HIGH-ENERGY NUCLEON-MESON TRANSPORT CODE WITH FISSION*

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F. S. Alsmiller R. G. Alsmiller, Jr.

(Abstract of ORNL/TM-7882, July 1981)

A model that includes fission for predicting particle production spectra from medium-energy

nucleon and pion collisions with nuclei ($Z \geq 91$) has been incorporated into the nucleon-meson transport code, HETC. This report is primarily concerned with the programming aspects of HETFIS (High-Energy Nucleon-Meson Transport Code with Fission). A description of the program data and instructions for operating the code are given. HETFIS is written in FORTRAN IV for the IBM computers and is readily adaptable to other systems.

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[†]UCC-ND Computer Sciences Division.

3.8. CARDIOGRAMA: A STOCHASTIC, SEMI-EMPIRICAL METHODOLOGY FOR POWER REACTOR SURVEILLANCE AND DIAGNOSTICS*

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R. B. Perez

(Reprint of *Trans. Am. Nucl.* 39, 951, 1981)

The utilization of stochastic methods (reactor "noise") for power reactor diagnostics and surveillance applications is by now a relatively well-established technique.¹ In this technique, the power spectral density (PSD) of the fluctuations of a specified state variable is often used to define the reactor's "signature" at a given configuration. Typically, this PSD possesses considerable structure as a function of frequency (i.e., peaks and valleys), and these features have been related with various degrees of success to specific causative mechanisms such as fuel element vibrations, core barrel motions, and thermal-hydraulic resonances.² These features also form the basis for surveillance methods whereby changes in the structure of the PSD are detected and then correlated with altered plant conditions.

The purpose of the present work is to address the problem of handling efficiently the substantial amount of information involved in the application of reactor surveillance and diagnostics methods. Specifically, a methodology is described for (a) representing the PSDs parametrically, and (b) detecting changes from the reactor's baseline PSD (normal signature).

To this end, we model the resonances observed in the reactor PSD as arising from pairs of complex conjugate poles in the detector response

function $G(f)$. These poles appear at frequencies $f\lambda = \nu\lambda + i\mu$, where $\nu\lambda$ is the location of the resonance and $\mu\lambda$ is its width. The detector response can thus be modeled as

$$G(f) = \sum_{\lambda=1}^N \left[\frac{R_{\lambda}(f)}{i(f - \nu_{\lambda}) + \mu_{\lambda}} + CC \right], \quad (1)$$

where $R_{\lambda}(f)$ are the residues in the partial fraction expansion of $G(f)$, N is the number of poles necessary to describe the structure of the PSD, and CC stands for complex conjugate. The PSD, $\Phi G(f)$, is then defined as

$$\Phi G(f) = G(f)G^*(f) \quad (2)$$

Insertion of Eq. (1) into Eq. (2) yields

$$\begin{aligned} \Phi G(f) = & \sum_{\lambda=1}^N \left[\frac{A_{\lambda}\mu_{\lambda} + B_{\lambda}(f - \nu_{\lambda})}{(f - \nu_{\lambda})^2 + \mu_{\lambda}^2} \right] \\ & + \left[\frac{A_{\lambda}\mu_{\lambda} + B_{\lambda}(f + \nu_{\lambda})}{(f + \nu_{\lambda})^2 + \mu_{\lambda}^2} \right], \end{aligned} \quad (3)$$

where the amplitudes A and B are defined by

$$B_{\lambda} - iA_{\lambda} = 2 \sum_{\lambda'} \frac{R_{\lambda}R_{\lambda'}^*}{f_{\lambda'}^* - f_{\lambda}} \quad (4)$$

The first term in the right side of Eq. (3) describes a distorted resonance at $f = \nu\lambda$ due to the appearance of the resonance interference term $B(f - \nu\lambda)$, which changes sign when passing through the resonant frequency. The second term is a nonresonant contribution resulting from the companion complex conjugate pole. We emphasize that (a) no attempt has been made to distinguish between poles from noise sources and those from the reactor response function, and (b) the resonance widths, $\mu\lambda$, cannot be taken as the "natural" real parts of the poles, due to instrumental resolution broadening effects. Hence, parameterization (3) must be regarded as a semi-empirical representation of the reactor PSD.

To detect changes in the structure of the reactor signature, we utilize the Wald-Wolfowitz runs test.³ This test compares the number of runs, R (defined as unbroken sequences of points above and below a reference line), obtained from an ordered set of points with an expected value, $E(R)$, and its variance, $V(R)$, derived from a set of randomly distributed points. Hence, the Wald-Wolfowitz test is sensitive to the structure of the data remaining after removal of the fitted structure [Eq. (3)].

Our methodology is based on the following steps: (a) the baseline PSD data are fitted according to parameterization (3) and a Wald-Wolfowitz test is performed to determine the "background" number of runs, R_0 , as well as $E(R_0)$ and $V(R_0)$ for randomly distributed data points; (b) subsequent PSDs are fitted by the same pole structure as the one corresponding to the baseline (with renormalization of amplitudes) and a runs test is performed to ascertain whether the number of runs, R_0 , falls within $R_0 \pm 2V^{1/2}(R_0)$; (c) those PSDs failing the runs test in step b are refitted and a new pole structure is compiled.

This methodology has been implemented in the FORTRAN IV code CARDIOGRAMA, which includes algorithms for automatic search for peaks in the PSD. Figure 3.8.1 shows a CARDIOGRAMA fit to the PSD of a pressure signal from a 1000-MW(e) pressurized water reactor (PWR). Utilizing this fit as the baseline PSD, sensitivity tests were then performed on simulated anomalous data, which were constructed by altering the pole parameters 1%. These PSD changes, though not visible to the eye, were readily detected by the Wald-Wolfowitz test.

In summary, the program CARDIOGRAMA has been implemented to (a) compile reactor PSDs and classify them on a basis of the number and location of their poles in the frequency domain, and (b) provide an algorithm for rapid, sensitive identification of changes in reactor signatures.

*Research sponsored by U.S. Department of Energy.

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Fig. 3.8.1. The Points in This Figure Show the PSD, or a Function of Frequency, Obtained from Pressure Signal Fluctuations in a 1000-MW(e) PWR Plant. The continuous line is the functional fit performed on the basis of Eq. (3) in the text.

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3.9. ON THE THEORY OF THE STOCHASTIC NEUTRON FIELD IN ZERO POWER NUCLEAR REACTORS*

J. L. Munoz-Cobos[†] R. B. Perez

(Reprint of *Trans. Am. Nucl. Soc.* 39, 950, 1981)

In this work, which could as well be titled "Zero Power Noise Revisited," we examine in detail the stochastic theory of neutron transport within the framework of the probability balance method as developed by Pal¹ and Bell.² Our goals are to (a) determine whether or not the calculation of zero power noise can be regarded as a "completely solved problem; only computational difficulties remain,"³ and (b) develop some basic ideas for practical computational algorithms. The motivation for this work arises from difficulties encountered in the interpretation of some zero

power noise experiments.⁴ We use standard notation; the set of phase-space position coordinates is represented by a number, i.e., $(r, \Omega, v) = (1)$, and the inner product between the functions A , B , defined as their product integrated over phase-space and time, will be represented by $\langle A|B \rangle$. The n 'th moment of the neutron field $N(1, t)$ (neutrons per phase-space cell) is obtained by operating with $(\partial^n / \partial Z^n)_{Z=1}$ on the source generating function, $G_s(Z, 1, t)$. The source generating function is defined in terms of the probability $P_N(1, t)$ of having $N(1, t)$ neutrons, by

$$G_s(Z, 1, t) = \sum_{N=0}^{\infty} Z^N \cdot P_N(1, t) \quad (1)$$

On the basis of a conservation equation for $P_n(1, t)$ in terms of $K_n(1', t'|1, t)$, defined as the probability of having n neutrons per unit cell in phase space $(1, t)$ upon injection of one neutron at $(1', t')$, we show that $G_s(Z, 1, t)$ is given by

$$G_s(Z, 1, t) = \exp \left\{ \sum_{i=0}^I \epsilon_i \int_{t_0}^t dt' \int_{r'} dr' S_0(r') \right. \\ \left. X \left[\left\{ \int_{\Omega} \int_{v'} dv' \frac{d\Omega'}{4\pi} X_{\chi_s(v')} G_K(Z, 1', t'|1, t) \right\}^i - 1 \right] \right\} \quad (2)$$

where ϵ_i is the probability of emission of i neutrons per source disintegration. $S_0(r, t)$ is the number of source disintegrations per second and unit volume, χ_s , the normalized source fission spectrum, and $G_K(z, 1', t'|1, t)$ is the generating function

$$G_K(Z, 1', t'|1, t) = \sum_{n=0}^{\infty} Z^n K_n(1', t'|1, t) \quad (3)$$

The result, Eq. (2), reduces to the formula, Eq. (24) in Ref. 2, for $i = 1$, i.e., for nonfission sources, and $t_0 = -\infty$.

From a probability balance for the kernel, K_n , and the utilization of Pal's methodology, one obtains the following expression for G_K :

$$\left[\frac{1}{v'} \frac{\partial}{\partial t'} + \Omega' \cdot \nabla_{r'} - \Sigma(r', v') \right] G_K(1', t'|1, t) \\ = -\Sigma(r', v') \left\{ C_0(r', v', t') + \int_{v'} \int_{\Omega} dv'' d\Omega'' C_1(1', t'|v'' \Omega'') \right\}$$

$$\begin{aligned}
& \times G_K(Z, r', t', v'', \Omega'' | 1, t) + \sum_{j=2}^I c_j(r', t', v') \\
& \times \left[\int_{r''} \int_{\Omega''} dv'' d\Omega'' \chi(v'') G_K(Z, r', t', v'', \Omega'' | 1, t) \right] j \\
& - \frac{Z}{v'} \delta(1 - v') \delta(t - t'), \quad (4)
\end{aligned}$$

where $C_0(r, v, t)$ is the neutron absorption probability, $C_1(1', t' | v'', \Omega'')$ is the probability that a neutron having a collision at $(1', t')$ leads to one neutron with velocity v'' in the direction Ω'' , $C_j(r', t' | v', \Omega')$ is the probability of a fission event that generates j neutrons, and $\chi(v'')$ is the normalized fission spectrum. The result, Eq. (4), is the energy-dependent Pal's equation [Eq. (7) in Ref. 2], with the addition of a δ -function term. This contribution, which corresponds to a source term in Eq. (4), arises from one of the terms in the probability balance for K_n that accounts for uncollided neutrons.

Upon application of the operator $(\partial/\partial Z)Z = 1$ in Eq. (2) and (4), one obtains for the average flux $\bar{\phi}(1, t) = vN(1, t)$ the result

$$\left(\frac{1}{v} \frac{\partial}{\partial t} - L \right) \bar{\phi} = \bar{S}, \quad (5)$$

where L is the Boltzmann transport operator and

$$\bar{S}(1, t) = \frac{\bar{v}_s}{4\pi} S_0(r, t) \chi_s(v).$$

The result, Eq. (5), confirms the sound basis of the Pal and Bell formalism in that the Boltzmann equation for the average flux must emerge from a correct stochastic theory of the neutron field. However, for this to occur, the δ -function in Eq. (4) must be included.

Operating with $(\partial^2/\partial Z^2)Z = 1$ in Eqs. (2) and (4), one derives the following expression for the relative variance, $V(N) = [N^2 - (\bar{N})^2]/\bar{N}$:

$$V(N) = 1 + \frac{1}{\bar{N}} \langle \bar{\phi} | Q \rangle + \frac{1}{\bar{N}} Q_s, \quad (6)$$

where Q and Q_s are "sources" given, respectively, by

$$Q(1, t | 1', t') = \frac{1}{v^2} \sum_j (r', v') \overline{v(v-1)} \Gamma_0^2(1, t | r', t'), \quad (7)$$

$$Q_s(1, t) = \overline{v_s(v_s-1)} \int_{t_0}^t dt' \int_{r_0}^r dr' S_0(r) \Gamma_s^2 \frac{(1, t | r', t')}{v^2}, \quad (8)$$

and where the generalized functions, Γ_0 and Γ_s , satisfy the equations

$$\left(\frac{1}{v} \frac{\partial}{\partial t} - L \right) \Gamma_0 = F_0, \quad (9)$$

$$\left(\frac{1}{v} \frac{\partial}{\partial t} - L \right) \Gamma_s = F_s, \quad (10)$$

with the sources

$$F_0(v, r, r', t, t') = \frac{1}{4\pi} \chi(v) \delta(r - r') \delta(t - t'), \quad (11)$$

$$F_s(v, r, r', t, t') = \frac{1}{4\pi} \chi_s(v) \delta(r - r') \delta(t - t'), \quad (12)$$

and the appropriate boundary and initial conditions. The results, Eq. (6), gives the "excess" relative variance over the Poissonian relative variance of unity as the sum of two contributions: (a) the variance due to the inherent fluctuations of the system, which appears as the convolutions (inner product) of the average flux and the "source" Q , and (b) the variance due to the external spontaneous fission source. It is noteworthy to observe that the term in Q_s , appearing in Eq. (6) for the relative variance, would have been formulated incorrectly from the usual expression² for the generating function $G_s(Z, 1, t)$ when applied to spontaneous fission sources.

Examination of Eqs. (6) through (10) reveals that the relative variance can be calculated via existing computational algorithms for nuclear

reactor calculations to obtain $\bar{\phi}$ and the generalized functions Γ_0 and Γ_s , plus the quadratures in Eqs. (6) and (8). This procedure takes into account spatial and energy effects within the framework of approximations involved in standard "real-life" reactor calculations.

In conclusion, we find that (a) the Pal and Bell formalism with the previously mentioned provisos leads to what appears to be a complete description of zero power noise, and (b) the computational difficulties referred to at the beginning can be ameliorated by concentrating of the calculation of the observables $\bar{\phi}$ and $V(N)$ by the technique set forth in this work, rather than by attempting to calculate^{2,3} the generating function, G_k .

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3.10. PARTICLE AND RADIATION LEAKAGE IMPORTANCE: DEFINITION, ANALYSIS, AND INTERPRETATION*

D. G. Cacuci J. J. Wagschal[†] A. Yaari[†]
[Abstract of *Nucl. Sci. Eng.* **81**, 443 (1982)]

The concept of *leakage importance* function is introduced and analyzed for physical systems governed by the Boltzmann transport equation. The homogeneous equation with inhomogeneous boundary conditions satisfied by the importance function is derived by using adjoint operators. A standard discrete ordinates transport code is used to solve this equation, and some important numerical aspects are highlighted. Idealized nuclear systems are analyzed to illustrate that the leakage importance function gives a measure of the relative importance of each source particle in phase-space in contributing to the leakage, and that the leakage importance function provides

insight regarding the specific physical process that leads to leakage.

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3.11. LEAKAGE IMPORTANCE*

A. Yaari[†] J. J. Wagschal[†] D. G. Cacuci

(Abstract of *Trans. Isr. Nucl. Soc.* **9**, 1.17, 1982)

This work discusses properties of an importance function for particles that contribute specifically to the leakage out of a finite-dimensional physical system governed by the Boltzmann equation. This leakage importance function provides insight regarding both the specific physical process that leads to leakage and the importance of source particles in various points in phase-space. Such insight is not provided by the solution of the forward Boltzmann equation. The availability of a leakage importance function is expected to provide a deeper understanding of physical phenomena involving neutron, gamma, or neutrino transport in reactor problems and astrophysics.

*Research sponsored by U.S. DOE Division of Reactor Research and Technology.

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3.12. SURVEILLANCE DOSIMETRY: ACHIEVEMENTS AND DISAPPOINTMENTS*

J. J. Wagschal[†] R. E. Maerker
B. L. Broadhead[†]

(Abstract of a paper presented at the Fourth ASTM-Euratom Symposium on Reactor Dosimetry, National Bureau of Standards, Washington, DC, March 21-26, 1982)

A methodology has been developed for reducing the uncertainties in estimates of neutron fluence spectra within the pressure vessel of a PWR from dosimetry measurements performed at a surveillance location. This new procedure, named the LEPRICON methodology, has three desirable features not presently available in other spectral unfolding codes: (1) the derivation of flux covari-

ances at each of the two locations; (2) the correlations of these two matrices to one another (i.e., the translation problem); and (3) the development of a flexible data base which can be expanded as more dosimetry measurements become available and are analyzed. The technique is applied to the PCA, where reductions in the uncertainties of the estimated fluxes in the pressure vessel of the order of a factor of two are demonstrated.

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3.13. PEBBLE: A TWO-DIMENSIONAL STEADY-STATE PEBBLE BED REACTOR THERMAL HYDRAULICS CODE*

D. R. Vondy

(Abstract of ORNL-5698, September 1981)

This report documents the local implementation of the PEBBLE code to treat the two-dimensional steady-state pebble bed reactor thermal hydraulics problem. This code is implemented as a module of a computation system used for reactor core history calculations. Given power density data, the geometric description in (RZ), and basic heat removal conditions and thermal properties, the coolant properties, flow conditions, and temperature distributions in the pebble fuel elements are predicted. The calculation is oriented to the continuous fueling, steady state condition with consideration of the effect of the high energy neutron flux exposure and temperature history on the thermal conductivity. The coolant flow conditions are calculated for the same geometry as used in the neutronics calculation, power density and fluence data being used directly, and temperature results are made available for subsequent use. Thus full temperature feedback capability is available within the approximations implemented, allowing account of temperature effects in neutronics and nuclide exposure calculations and admitting the calculation of reactivity effects associated with specific changes in the state of the reactor core. Provision is made to save and recover data, adding the

potential to reduce computation costs and enhancing the modeling capability.

*Research sponsored by U.S. DOE Office of Nuclear Power Systems.

3.14. THE GAIN FROM A MIXED FINITE DIFFERENCE FORMULATION FOR THREE-DIMENSIONAL DIFFUSION THEORY NEUTRONICS*

D. R. Vondy T. B. Fowler

(Reprint of Trans. Am. Nucl. 39, 460, 1981)

We demonstrate in Fig. 3.14.1 for a benchmark fast reactor problem¹ the advantage of a mixed differencing scheme for representing the diffusion theory approximation to neutron transport in three-dimensional triangular-Z geometry. Most of the early codes employed the mesh edge difference formulation, as is used in the German D3E code.² Long ago, we chose to use a mesh-centered formulation in application on a routine basis, with mesh points located at the centers of the finite difference elements instead of at the corners where the internal material interfaces intersect, the VENTURE code³ being our latest to use this scheme. From many applications to a variety of reactor core designs, we found that a global property, such as k_{eff} , that is overestimated by one formulation due to the finite difference error is often underestimated by the

Fig. 3.14.1. Results for the Three-Dimensional KFK (German) Fast Reactor Problem.

other. When this is true for both the planar cut and the axial traverse, then a mixed formulation should exhibit some compensation of the finite difference errors. Indeed such is the case as illustrated by results obtained with VALE, our new neutronics code,⁴ with points mesh edge on planes and planes mesh centered.

These results are for a fast reactor core problem modeling hexagonal assemblies. With a mesh-centered formulation on planes, the coarsest formulation considered was six points per hexagonal assembly, transport being approximated with coupling to the three nearest neighbors. With a mesh edge formulation on planes, the coarsest approximation would be one point in the center of an assembly and one at each corner (shared with three neighbor assemblies), effectively three points per assembly (although boundaries add points to the total), transport being approximated with coupling to the six nearest neighbors, which should be an improvement but adds some computation cost per point. The VALE results are for 3 point/hex and 18 planes, 12 point/hex and 36 planes, and 48 point/hex and 72 planes. The coarsest problem solved by the VENTURE code contained 6 point/hex and 18 planes, while the large fine-mesh problem treated a total of 1 123 632 space-energy points.

The advantage of more accurate results is, of course, obvious. A coarse-mesh model must be used in many applications due to the high cost of solving three-dimensional problems. It should be noted that the ordering of the differencing schemes of mesh-centered, mesh-centered planes with points located mesh edge on planes, and mesh edge is in increasing programming complexity, increasing data handling burden, and increasing computation cost per mesh point treated.

*Research sponsored by U.S. DOE Division of Reactor Research and Technology.

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3.15. A DOUBLE ERROR MODE EXTRAPOLATION PROCEDURE*

D. R. Vondy T. B. Fowler

[Abstract of Nucl. Sci. Eng. 80, 198 (1982)]

A procedure is described for removing two error vectors from the approximation to the solution obtained iteratively. Such a scheme for accelerating convergence can reduce the computation cost of solving neutronics problems as is demonstrated for finite-difference diffusion theory.

*Research sponsored by U.S. DOE Office of Nuclear Reactor Programs.

3.16. NEUTRONICS CODE VALE FOR TWO-DIMENSIONAL TRIANGULAR (HEXAGONAL) AND THREE-DIMENSIONAL GEOMETRIES*

D. R. Vondy T. B. Fowler

(Abstract of ORNL-5792, August 1981)

This report documents the computer code VALE designed to solve multigroup neutronics problems with the diffusion theory approximation to neutron transport for a triangular arrangement of mesh points on planes in two- and three-dimensional geometry. This code parallels the VENTURE neutronics code in the local computation system, making exposure and fuel management capabilities available. It uses and generates interface data files adopted in the cooperative effort sponsored by Reactor Physics RRT Division of the U.S. DOE. The programming in Fortran is straightforward, although data is transferred in blocks between auxiliary storage devices and main core, and direct access schemes are used. The size of problems which can be handled is essentially limited only by cost of cal-

culation since the arrays are variably dimensioned. The memory requirement is held down while data transfer during iteration is increased only as necessary with problem size.

There is provision for the more common boundary conditions including the repeating boundary, 180° rotational symmetry, and the rotational symmetry conditions for the 30°, 60°, and 120° triangular grids on planes.

A variety of types of problems may be solved: the usual neutron flux eigenvalue problem, or a direct criticality search on the buckling, on a reciprocal velocity absorber (prompt mode), or on nuclide concentrations. The adjoint problem and fixed source problem may be solved, as well as the dominating higher harmonic, or the importance problem for an arbitrary fixed source.

*Research sponsored by U.S. DOE Office of Reactor Research and Technology.

3.17. THE BOLD VENTURE COMPUTATION SYSTEM FOR NUCLEAR REACTOR CORE ANALYSIS, VERSION III*

**D. R. Vondy T. B. Fowler
G. W. Cunningham, III[†]**

(Abstract of ORNL-5711, June 1981)

This report is a condensed documentation for VERSION III of the BOLD VENTURE COMPUTATION SYSTEM for nuclear reactor core analysis. An experienced analyst should be able to use this system routinely for solving problems by referring to this document. Individual reports must be referenced for details. This report covers basic input instructions and describes recent extensions to the modules as well as to the interface data file specifications. Some application considerations are discussed and an elaborate sample problem is used as an instruction aid. Instructions for creating the system on IBM computers are also given.

*Work sponsored by U.S. DOE Office of Reactor Research and Technology.

[†]UCC-ND Computer Sciences Division.

3.18. VALIDATION TESTS OF THE VENTURE FINITE-DIFFERENCE DIFFUSION THEORY NEUTRONICS CODE*

D. R. Vondy T. B. Fowler

(Abstract of ORNL/TM-7634, March 1981)

The VENTURE code has been programmed for digital computer neutronics calculations. This report discusses the testing done to prove that reliable solutions are obtained. Some of the results produced and discussion about them may be of general interest, especially in application for reactor analysis. The authors believe that the extensive testing done with this code allows a high level of confidence to be placed on results produced if due consideration is given to limits of the model, to discrete mesh requirements, to basic requirements for adequate nuclear properties, and to satisfying adequate convergence criteria for the iterative procedures of solution.

*Research sponsored by U.S. DOE Office of Reactor Research and Technology.

3.19. ESTIMATING THE EIGENVALUE OF THE DOMINANT ERROR VECTOR OF AN ITERATION PROCESS*

D. R. Vondy T. B. Fowler

(Abstract of Nucl. Sci. Eng. 80, 473 (1982))

The cost of solving a reactor core neutronics problem is reduced by accelerating the iterative process. Special techniques are used to estimate the eigenvalue of the dominant error vector to support acceleration. Eigenvalue estimates can be tested for reliability to help make acceleration effective.

*Research sponsored by U.S. DOE Office of Nuclear Reactor Programs.

3.20. ON THE FINITENESS OF THE NUMBER OF DISCRETE EIGENVALUES IN NEUTRON TRANSPORT THEORY*

D. G. Cacuci

(Abstract of *J. Math. Phys.*, in press)

This work examines the finiteness of the number of discrete eigenvalues occurring in neutron slowing down and transport in an infinite medium consisting of a single nuclide scattering elastically, isotropically, and with energy-independent cross sections. This finiteness is related to convergence conditions for the Legendre moments of the scattering function. It is shown that the convergence criterion obtained by Case for one-velocity neutron transport is not satisfied by hydrogen at large lethargies. A new convergence criterion, weaker than that of Case, is derived and shown to be satisfied by all nuclides, including hydrogen, at all lethargies. Although derived within the more general framework of energy-dependent transport, the new convergence criterion also holds for the Legendre moments of the scattering function in one-velocity transport.

*Research sponsored by U.S. DOE Office of Basic Energy Sciences.

3.21. EIGENVALUE-DEPENDENT NEUTRON ENERGY SPECTRA: DEFINITIONS, ANALYSES, AND APPLICATIONS*

D. G. Cacuci[†] Y. Ronen[†] Z. Shayer[†]
J. J. Wagschal[‡] Y. Yeivin[§]

[Abstract of *Nucl. Sci. Eng.* 81, 432 (1982)]

An analysis of spectral effects that arise from solving the k -, α -, γ -, and δ -eigenvalue formulations of the neutron transport equation is presented. Hierarchies of neutron spectra softness are established and expressed in terms of spatial-dependent local indices that are defined for both the core and the reflector of nuclear system configurations. Conclusions regarding the general behavior of the spectrum-dependent

integral spectral indices and initial conversion ratios given by the k -, α -, γ -, and δ -eigenvalue equations are also presented. Spectral effects in the core and in the reflector are distinguished by defining separate integral spectral indices for the core and for the reflector.

It is shown that the relationship between the spectra given by the k -, α -, γ -, and δ -eigenvalue equations and the spectrum in a corresponding critical configuration depends on the specific physical process that causes deviation from criticality. Nevertheless, some general recommendations are offered regarding the use of a particular eigenvalue equation for specific applications. All conclusions are supported by numerical experiments performed for an idealized thermal system.

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3.22. ANALYSIS AND SOLUTION OF THE ILL-POSED INVERSE HEAT CONDUCTION PROBLEM*

C. F. Weber[†]

(Abstract of *ORNL/CSD/TM-144*, January 1982)

The inverse conduction problem arises when experimental measurements are taken in the interior of a body and it is desired to calculate temperature and heat flux values on the surface. The problem is shown to be ill-posed, as the solution exhibits unstable dependence on the given data functions. A special solution procedure is developed for the one-dimensional case which replaces the heat conduction equation with an approximating hyperbolic equation. If viewed from a new perspective, where the roles of the spatial and time variables are interchanged, then an initial value problem for the damped wave equation is obtained. Since this formulation is well-posed, both analytic and numerical solution procedures are readily available. Sample calculations confirm that this approach produces con-

sistent, reliable results for both linear and non-linear problems.

*Research sponsored by U.S. Nuclear Regulatory Commission PWR Blowdown Heat Transfer Program.

†UCC-NV Computer Sciences Division.

3.23. RESONANCE SHIELDING IN THERMAL REACTOR LATTICES*

W. Rothenstein[†] J. Barben

E. Tavir[†] M. Aminpour[†]

[Abstract of *Ann. Nucl. Energy*, Vol. 9, 141 (1982)]

The theoretical foundations of a new methodology for the accurate treatment of resonance absorption in thermal reactor lattice analysis are presented. This methodology is based on the solution of the point energy transport equation in its integral or intergo-differential form for a heterogeneous lattice using detailed resonance cross section profiles. The methodology is applied to LWR benchmark analysis, with emphasis on temperature dependence of resonance absorption during fuel depletion, spatial and mutual self-shielding, integral parameter analysis and treatment of cluster geometry. The capabilities of the OZMA code, which implements the new methodology are discussed. These capabilities provide a means against which simpler and more rapid resonance absorption algorithms can be checked.

*Research sponsored by Electric Power Research Institute.

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3.24. FUEL MANAGEMENT POSITIONING AND ACCOUNTING MODULE: FUELMANG*

T. W. Medlin[†] K. L. Hill[†] G. L. Johnson[†]

J. E. Jones[†] D. R. Vordy

[Abstract of ORNL-5718 (TEC R-81-044), January 1982]

This report documents the code module FUELMANG for fuel management of a reactor. This code may be used to position fuel during the calculation of a reactor history, maintain a mass balance history of the fuel movement, and calculate the unit fuel cycle component of the electrical generation cost. In addition to handling fixed feed fuel without recycle, provision has been made for fuel recycle with various options applied to the recycled fuel. A continuous fueling option is also available with the code. A major edit produced by the code is a detailed summary of the mass balance history of the reactor and a fuel cost analysis of that mass balance history. The code is incorporated in the system containing the VENTURE diffusion theory neutronics code for routine use. Fuel movement according to prescribed instructions is performed without the access of additional user input data during the calculation of a reactor operating history. Local application has been primarily for analysis of the performance of proposed LMFBR and gas-cooled thermal reactor core concepts.

*Research sponsored by U.S. DOE Office of Advanced Nuclear Systems and Projects.

†Technology for Energy Corporation

Section 4

METHODS FOR SENSITIVITY AND UNCERTAINTY ANALYSES

4.0. INTRODUCTION

During the period covered by this report, the development of sensitivity and uncertainty analysis methods has progressed on both practical and theoretical levels, as is evident from the abstracts and summaries collected here. In addition, other work, not yet published, promises to simplify sensitivity analysis and to allow it to be applied in new areas.

FORSS System, ORACLE-I Library. -- In reactor physics applications, the FORSS sensitivity and uncertainty analysis system has matured. As has been previously reported, the FORSS system consists of a collection of codes that include cross-section processing, uncertainty file processing, neutronics calculations, source calculations, sensitivity profile generation, uncertainty analysis, and data adjustment capabilities.

The FORSS system has been used to generate ORACLE-I, a recently completed version of the Oak Ridge adjusted cross-section and covariance library ORACLE. This library is based on the a priori 174-group VITAMIN-E cross-section library, a priori covariances derived from ENDF/B-V and other sources, and 27 benchmark integral experiments. ORACLE-I not only consists of adjusted cross sections in the 174-group structure but also includes adjusted cross-section covariances in a 26-group structure.

DEPTH-CHARGE System. -- Development of the DEPTH-CHARGE modular system has been completed and implemented within the VENTURE-BURNER computational system at ORNL. The DEPTH module calculates, in conjunction with VENTURE, first-order response sensitivities to variations in the nuclide concentrations and nuclear data. These sensitivities are determined by using adjoint functions. The CHARGE module is used for general data manipulation and input preparation and for automatic generation of adjoint sources, estimated perturbed responses, and relative sensitivities. The DEPTH-CHARGE system is an efficient tool for solving realistic multi-cycle reactor depletion problems.

Adaptation of FORSS, ORACLE-I, and DEPTH-CHARGE Systems to Specific Reactor Problems. -- A major application of DEPTH-CHARGE, FORSS, and ORACLE-I has been the sensitivity and uncertainty analysis of key performance parameters for a large heterogeneous Liquid Metal Fast Breeder Reactor core model. Four responses (k_{eff} , breeding ratio, peak power density ratio, and sodium void reactivity) have been analyzed. The uncertainty in k_{eff} due to nuclear data uncertainties is reduced from 3% (with unadjusted cross sections) to 0.4% with the adjusted library.

Other applications of FORSS and DEPTH-CHARGE include sensitivity studies of fuel irradiation experiments performed at France's Phenix reactor and analysis of light-water reactors. In addition, it is being adapted for application to problems of interest to the High Temperature Gas-Cooled Reactor (HTGR). Specifically, it will be used to look at the impact of burnup on the Fort St. Vrain reactor and to examine the effect of proposed new fuel block configurations for that reactor.

Sensitivity Capabilities for Thermal-Hydraulic Problems. -- The initiation of the development of sensitivity and uncertainty analysis capabilities for thermal-hydraulic problems was reported in our previous report. The successful completion of those initiatives has led to the implementation of a sensitivity analysis capability for the neutronic/thermal-hydraulic fast reactor safety code MELT-IIIB, and to the formulation of an advanced uncertainty analysis capability for time-dependent non-linear systems. Sensitivities of responses to neutronic/thermal-hydraulic parameters in MELT-IIIB

are calculated by using adjoint functions produced by MELTADJ - a module that solves appropriately constructed adjoint equations. The combined MELT-IIIB and MELTADJ system has been used to perform sensitivity analyses of two transients in the Fast Flux Test Facility (FFTF). These analyses underscored the practical advantages of using adjoint functions for sensitivity analysis.

An advanced method for systematic uncertainty analysis of time-dependent nonlinear systems has been developed. This method includes a capability for reducing uncertainties in system parameters and responses by using Bayesian inference techniques to consistently combine prior knowledge with additional experimental information. The method was applied to the uncertainty analysis of a transient upflow in a high-pressure water heat transfer experiment and the results were compared with those obtained by standard response techniques. With the new method, major reductions in the discrepancies between calculated and measured values were realized, and incorporating in-bundle measurements in the uncertainty analysis significantly reduced the system uncertainties.

Second-Order Uncertainty Analysis Method. -- A method for second-order uncertainty analysis of problems involving generally correlated but normally distributed parameters has also been formulated. All quantities required to apply this method are expressed analytically in terms of the customary covariance matrices and in terms of first- and second-order sensitivities. This method has been successfully applied to evaluate variances of the initial conversion ratio and infinite multiplication factor for a PWR unit cell.

Recent Developments: Comprehensive Sensitivity Theory for Nonlinear Systems. -- The successful use of adjoint functions to develop methods for sensitivity analysis of specific problems as mentioned in the foregoing has motivated an investigation of the mathematical foundations underlying sensitivity theory for nonlinear systems. Concepts of nonlinear functional analysis have been used to establish the necessary and sufficient conditions underlying this theory. Consequently, it has been possible not only to ascertain the limitations inherent in existing analytical approaches to sensitivity analysis, but also to rigorously formulate a considerably more general sensitivity theory for physical problems characterized by systems of nonlinear equations and by nonlinear functionals as responses. The rigor and generality of this formalism stem from the use of Gateaux (G-) differentials. In particular, the use of G-differentials gives precise mathematical and physical meaning to the sensitivity of a functional-type response to parameters that are functions rather than scalars, and allows treatment of discontinuous operators and parameters.

This sensitivity theory has subsequently been extended to include treatment of additional types of responses. There are physical systems where a critical point of a function that depends on the system's state vector and parameters defines the location in phase space where the response functional is evaluated. Examples of such critical points include maxima, minima, and saddle points. The G-differentials giving the sensitivities of both the functional and the critical point to changes in the system's parameters have been obtained by alternative formalisms, namely, the "Forward Sensitivity Formalism" and the "Adjoint Sensitivity Formalism."

Sensitivity theory is also extended to include treatment of general operators, acting on the system's state vector and parameters, as responses. The Adjoint Sensitivity Formalism requires expanding the indirect effect term, an element of a Hilbert space, in terms of elements of an orthonormal basis. Since as many calculations of adjoint functions are required as there are nonzero terms in this expansion, careful consideration of truncating the expansion is needed to assess the advantages of the Adjoint Sensitivity Formalism over the Forward Sensitivity Formalism.

New Applications of Sensitivity Analysis. -- These theoretical developments have opened the way for new applications of sensitivity analysis. For example, sensitivity analyses of extremum-type

responses have been performed for both a simple LMFBR model (involving the linear Boltzmann equation and a peak power density response) and a complex transient in the FFTF modeled by using the fast-reactor safety code MELT-III. In the latter application, two responses of practical importance (the maximum fuel temperature in the hot channel and the maximum normalized reactor power level) have been considered. For the purposes of sensitivity analysis, a complete characterization of such responses requires consideration of both the numerical value of the response at the maximum and the location in phase space where the maximum occurs. This is because variations in the system parameters alter not only the value at this maximum but also alter the location of the maximum in phase-space.

The comprehensive sensitivity theory previously mentioned has also been applied to develop a rigorous formalism for sensitivity analysis of functional-type responses associated with the well-posed system of quasilinear partial differential equations (PDE) of hyperbolic type that describe one-dimensional two-phase flows. This formalism uses adjoint functions to determine efficiently sensitivities to many parameter variations. The physical meaning of this sensitivity analysis formalism is illustrated by an application to the homogeneous equilibrium model for two-phase flow. Although this formalism does not address transition phenomena between single- and two-phase flow regimes, and it ignores higher-order effects of parameter variations, it does provide a complete theoretical framework for implementing an efficient sensitivity analysis capability into one-dimensional two-phase flow models.

Application to CO₂ Research Program. -- Sensitivity analysis is also being performed in conjunction with the Division's new CO₂ research program. One of the major outstanding questions in this program centers on the sensitivity of the results of atmospheric general circulation models (GCMs) to uncertainties in input parameters. The comprehensive sensitivity theory mentioned above has been successfully applied to a radiative-convective model (RCM) that contains many of the nonlinear processes found in GCMs. A single adjoint calculation, which requires about the same computation time as the original model, suffices to calculate sensitivities of surface-air temperature to all 312 model parameters. The sensitivities accurately predict the effect on surface-air temperature of small variations in the model parameters. Several of the sensitivities to parameters customarily considered in previous works (e.g. solar constant, surface albedo, relative humidity, CO₂ concentration) are reproduced, but the largest sensitivities are to constants used to compute the saturation vapor pressure of water.

Current Research Directions. -- Current research is divided between further developing sensitivity theory and investigating new areas of application. Theoretical efforts include investigations of problems involving structural changes such as those occurring in fault trees, and development of global sensitivity analysis capabilities for assessing effects of large parameter variations, including second- and higher-order effects.

A recently developed language that reads computer codes written in FORTRAN and translates them into their adjoint form, called GRESS (for GRAdient Enhanced Software System), will soon be available. The scope of GRESS is to automatically produce the adjoint solution necessary to calculate sensitivities, and its availability should shorten the time requirements for undertaking major sensitivity analysis tasks. Its feasibility has been demonstrated by application to a small LEAP-type energy model.

Existing theoretical developments in sensitivity and uncertainty analysis are expected to be applied to new areas, including problems involving phase changes, problems of waste disposal, and climatic projections based on GCMs. For example, the adjoint method of sensitivity analysis will

be applied to the Oregon State University two-level GCM. This investigation is expected to alert the modeling community to the most important measurements or parameterization improvements that need to be made to reduce uncertainties in the GCM results, and to provide a systematic sensitivity and uncertainty analysis capability to analyze the climatic impact of increased CO₂.

4.1. SENSITIVITY THEORY FOR NONLINEAR SYSTEMS:

I. NONLINEAR FUNCTIONAL ANALYSIS APPROACH*

D. G. Cacuci

[Abstract of ORNL/TM-7304, August 1981; also *J. Math. Phys.* 22 (12), 2794 (December 1981); also EPRI NP-81-10-LD (July 1981)]

Concepts of nonlinear functional analysis are employed to investigate the mathematical foundations underlying sensitivity theory. This makes it possible not only to ascertain the limitations inherent in existing analytical approaches to sensitivity analysis, but also to rigorously formulate a considerably more general sensitivity theory for physical problems characterized by systems of nonlinear equations and by nonlinear functionals as responses. Two alternative formalisms, labeled the "Forward Sensitivity Formalism" and the "Adjoint Sensitivity Formalism," are developed in order to evaluate the sensitivity of the response to variations in the system parameters. The "Forward Sensitivity Formalism" is formulated in normed linear spaces, and the existence of the Gateaux-differentials of the operators appearing in the problem is shown to be both necessary and sufficient for its validity. Although conceptually straightforward and applicable to a considerably wider range of problems than heretofore possible, this formalism is computationally expensive to employ. The alternative "Adjoint Sensitivity Formalism" is considerably more economical from a computational standpoint, but, as proven here, it can only be formulated under more restrictive conditions. In particular, the requirement that operators acting on the state vector and on the system parameters must admit densely defined Gateaux-derivatives is shown to be of fundamental importance for the validity of this formalism. The present analysis significantly extends the scope of sensitivity theory and provides a basis for still further generalizations.

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4.2. SENSITIVITY THEORY FOR NONLINEAR SYSTEMS: II. EXTENSIONS TO ADDITIONAL CLASSES OF RESPONSES*

D. G. Cacuci

[Abstract of ORNL/TM-7511, August 1981; also *J. Math. Phys.* 22 (12), 2803 (1981)]

This work extends a recent, functional-analytic formulation of sensitivity theory to include treatment of additional types of responses. There are physical systems where a critical point of a function that depends on the system's state vector and parameters defines the location in phase-space where the response functional is evaluated. The Gateaux-differentials giving the sensitivities of both the functional and the critical point to changes in the system's parameters are obtained by alternative formalisms. The Forward Sensitivity Formalism is the simpler and more general but may be prohibitively expensive for problems with large data bases. The Adjoint Sensitivity Formalism, although less generally applicable and requiring several adjoint calculations, is likely to be the only practical approach.

Sensitivity theory is also extended to include treatment of general operators, acting on the system's state vector and parameters, as responses. In this case, the Forward Sensitivity Formalism is the same as for functional responses, but the Adjoint Sensitivity Formalism is considerably different. The Adjoint Sensitivity Formalism requires expanding the indirect effect term, an element of a Hilbert space, in terms of elements of an orthonormal basis. Since as many calculations of adjoint functions are required as there are nonzero terms in this expansion, careful consideration of truncating the expansion is needed to assess the advantages of the Adjoint Sensitivity Formalism over the Forward Sensitivity Formalism.

*Research sponsored by Electric Power Research Institute.

4.3. APPLICATION OF ZERO-POWER REACTOR EXPERIMENTS TO REACTOR DESIGN*

J. H. Marable

(Abstract of Slide Presentation at Georgia Institute of Technology, April 30, 1981)

Because of the accuracy limitations of existing nuclear data bases, it is necessary to incorporate the results of integral experiments in reactor design calculations. Such integral experiments are largely made in zero-power reactor assemblies (e.g. physics benchmark assemblies, engineering benchmark assemblies, or engineering mockup criticals), but integral experiments in standard neutron flux fields are also utilized. The responses measured are related to core-physics performance and typically include fuel and control-rod enrichments, power distributions, and safety coefficients. Adjustments to calculations are made through either the traditional bias-factor method, which is direct and simple to apply, or the least-squares adjustment method, which provides a logical combination of all the data but requires extensive covariances and sensitivities. The second and more recent approach leads to (1) a defensible technique for calculating uncertainties of core-physics performance, (2) the possibility of reducing design margins and design uncertainties, (3) the possibility of optimizing designs in a more reliable and consistent manner, and (4) the possibility of reducing the expensive zero-power reactor experimental program presently required for development of a real power reactor.

*Research sponsored by U.S. DOE Office of Reactor Research and Technology.

4.4. TAYLOR SERIES EXPANSION WITH GENERALIZED PERTURBATION THEORY FOR PEAK POWER INVESTIGATIONS*

L. A. Belblidia[†] A. Gandini[‡]
J. M. Kallfelz[†] V. A. Perone[†]

(Reprint of Trans. Am. Nucl. Soc. 39, 957, 1981)

Generalized perturbation theory (GPT) is often used to calculate the influence on integral reactor performance parameters of design and cross-section changes, and to determine integral parameter uncertainties.¹ In a previous study of a Conceptual Design Study-type heterogeneous core, integral parameters (henceforth called "responses," R) studied included the power densities at the locations of peak power density in each driver zone for the unperturbed case. The influence of "near-range" shifts in the radial location of the zonal peak power on the sensitivities, $\delta R / \delta \sigma$, was estimated to be small based on associated calculations.¹ However, such spatial shifts were not treated explicitly.

A Taylor series expansion method with GPT has recently been proposed to treat such shifts explicitly.^{2,3} Cacuci⁴ has also developed a theory involving Gateaux-differentials to investigate response maxima. The purpose of the work reported in this paper is to investigate the appropriateness of the Taylor series GPT approach for peak power investigations in a heterogeneous liquid-metal fast breeder reactor (LMFBR) core. The Taylor series approach involves application of GPT to functionals involving derivative operators,² and has the advantage that it can be applied directly to GPT production codes⁵⁻¹⁰ without further programming of these codes.³

For the present investigations, we have considered a heterogeneous three-region slab (x) model, with internal blanket (IB), driver (D), and external blanket (EB) concentrations of sodium, ^{238}U , and ^{239}Pu , the same as those in the LMFBR model of Ref. 11. The three-group cross sections used are from the CITATION test case.¹² The outer boundaries were 10.5, 60.0, and 91.0 cm from the reactor center, and the driver zone mesh spacing was 1.5 cm. The response considered was

$$R(x) = \frac{\sum_j \phi_j(x)}{\int_{\text{reactor}} \sum_j \phi_j(x') dx'} = \frac{Q_1(x)}{Q_2} \quad (1)$$

which is related to the ratio of point power density to total power considered in Ref. 1.

Using the basic equation for the expansion about point x_k ,

$$R'(x) = R'(x_k) + R^{(1)'}(x_k)(x-x_k) + \frac{1}{2!} R^{(2)'}(x_k)(x-x_k)^2 + \dots \quad (2)$$

the change in the response at x , $\delta R(x)$, due to a perturbation can be found from

$$\delta R(x) = \delta R(x_k) + \delta R^{(1)'}(x_k)(x-x_k) + \frac{1}{2} \delta R^{(2)'}(x_k)(x-x_k)^2 + \dots \quad (3)$$

where $R^{(i)'}$ and $\delta R^{(i)'}$ are the perturbed value and the change of the i 'th derivative. Considering only the first three terms in this series, the $\delta R^{(i)'}$ were determined at mesh point x_k using the Italian GPT codes⁸⁻¹⁰ and the finite difference approximation, i.e., using δR at x_k , x_{k-1} , and x_{k+1} . (For the following discussion, x_k is the location of the maximum unperturbed R -value for the space mesh considered.)

This method is also applicable to predictions of changes of the location of the maximum R . It can be shown that the location of the perturbed point of maximum R , x'_M , can be written as

$$x'_M - x_k = - \frac{Q_1^{(1)'}(x_k)}{Q_2^{(2)'}(x_k)} \quad (4)$$

where $Q_i^{(i)'}$ is the i 'th derivative of the numerator of the perturbed response function, R' .

In general, the point x_k is not the actual point of the unperturbed maximum response. The "exact" location, x_M , of this point is found, to the same degree of accuracy, by fitting $R(x_{k-1})$, $R(x_k)$, and $R(x_{k+1})$ by a quadratic polynomial function and setting its first derivative to zero. The spatial shift is then

$$x'_M - x_M = (x'_M - x_k) + (x_k - x_M) \quad (5)$$

For the present study, two perturbations were considered:

1. 3% decrease in the core ^{239}Pu density (N_c^{49}) and an addition of ^{239}Pu with a number density of $0.045 N_c^{49}$ to the outer blanket.
2. 5% decrease in N_c^{49} and 5% decrease in the inner and outer blanket ^{238}U density (N_b^{28}).

To examine the accuracy of the Taylor series expansion used in GPT for the above perturbations, $\delta R_p(x)$ determined by normal GPT at x was compared to $\delta R_{TP}(x)$ from Eq. (3). For expansions up to 6 cm (much greater than $x'_M - x_M$), they generally agree to $\sim 5\%$. An exception is the largest x -value for case 2; δR is apparently approaching a sign change with increased x , and the second-order Taylor expansion is less accurate.

Spatial shifts ($x'_M - x_M$) were also computed for two cases. For the Taylor series shift, S_{TP} , x'_M was determined from Eq. (4), while for the direct result, S_D , x'_M was derived from a quadratic fit to the perturbed flux distribution from direct calculations.

In conclusion, the results listed in Table 4.4.1 show good agreement and demonstrate the usefulness of the Taylor series expansion method. Also shown are the facts that for the perturbations considered, the spatial shifts and the error they introduce in the relative change of the response

Table 4.4.1. Taylor Series Expansion Accuracy and Spatial Shift Predictions for Three-Region Reactor Model for Base Point of Expansion, X_b , at 19.5 cm

	Case 1				Case 2			
S_D (cm)	0.128				-0.687			
S_{TP} (cm)	0.151				-0.658			
SE^a (%)	-0.198				2.672			
X (cm)	13.5	16.5	22.5	25.5	13.5	16.5	22.5	25.5
(T-P)/P (%)	-5.478	-1.471	-1.791	-7.403	5.440	1.660	2.832	15.048

$$^aSE \equiv \{[R'(X'_M) - R'(X_M)]/[R'(X'_M) - R(X_M)]\} \times 100.$$

$$^b(T-P)/P \equiv \{[\delta R_{TP}(X) - \delta R_P(X)]/\delta\} \times 100; \text{ see text for } \delta R \text{ definitions.}$$

are small. It should be noted that only three perturbation theory calculations are needed to predict the sensitivity at various locations. Therefore, for a spatial scan of power density sensitivities, this method can reduce considerably the number of necessary GPT calculations.

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4.5. THE DEVELOPMENT OF DEPLETION PERTURBATION THEORY FOR A REACTOR NODAL CODE*

S. M. Bowman[†]

(Abstract of ORNL/TM-7589, September 1981; also Master of Engineering Thesis, University of Tennessee)

A generalized depletion perturbation theory (DPT) formulation for light water reactor (LWR) depletion problems is developed and implemented into the three-dimensional LWR nodal code SIMULATE. This development applies the principles of the original derivation by M. L. Williams to the nodal equations solved by SIMULATE. The present formulation is first described in detail, and the nodal coupling methodology in SIMULATE is used to determine partial derivatives of the coupling coefficients. The modifications to the original code and the new DPT options available to the user are discussed.

Finally, the accuracy and the applicability of the new DPT capability to LWR design analysis is examined for several LWR depletion test cases.

The cases range from simple static cases to a realistic PWR model for an entire fuel cycle. Responses of interest included k_{eff} , nodal peaking, and peak nodal exposure. The nonlinear behavior of responses with respect to perturbations of the various types of cross sections were also investigated. The time dependence of the sensitivity coefficients for different responses were examined and compared.

Comparison of DPT results for these examples to direct calculations reveals the limited applicability of depletion perturbation theory to LWR design calculations at the present. The reasons for these restrictions are discussed, and several methods which might improve the computational accuracy of DPT are proposed for future research.

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4.6. THE DEPTH-CHARGE STATIC AND TIME-DEPENDENT PERTURBATION/SENSITIVITY SYSTEM*

J. R. White†

(Abstract of ORNL/CSD-78, September 1981)

This report provides the background theory, user input, and sample problems required for the efficient application of the DEPTH-CHARGE system -- a code block for both static and time-dependent nuclide perturbation theory and data sensitivity analyses. The DEPTH-CHARGE system is of modular construction and has been implemented within the VENTURE-BURNER computational system at Oak Ridge National Laboratory. The DEPTH module (coupled with VENTURE) solves for the three adjoint functions of Depletion Perturbation Theory and calculates the desired time-dependent derivatives of the response with respect to the nuclide concentrations and nuclear data utilized in the reference model. The CHARGE code is a collection of

utility routines for general data manipulation and input preparation and considerably extends the usefulness of the system through the automatic generation of adjoint sources, estimated perturbed responses, and relative data sensitivity coefficients. Combined, the DEPTH-CHARGE system provides, for the first time, a complete generalized first-order perturbation/sensitivity theory capability for both static and time-dependent analyses of realistic multidimensional reactor models.

*Research sponsored by U.S. DOE Division of Reactor Research and Technology.

†UCC-ND Computer Sciences Division.

4.7. DIFFERENTIAL SENSITIVITY THEORY APPLIED TO MOVEMENT OF MAXIMA RESPONSES*

P. J. Maudlin† C. V. Parks† D. G. Cacuci

(Reprint of Trans. Am. Nucl. Soc. 38, 333, 1981)

Differential sensitivity theory (DST) is a recently developed methodology to evaluate response derivatives $dR/d\alpha$ by using adjoint functions that correspond to the differentiated (with respect to an arbitrary parameter α) linear or nonlinear physical system of equations.^{1,2} However, for many problems, where responses of importance are local maxima such as peak temperature, power, or heat flux, changes in the phase-space location of the peak itself are of interest. This paper presents the DST procedure for predicting phase-space shifts of maxima responses as applied to the MELT-III fast reactor safety code.³

An adjoint version of the MELT-III code has been developed to allow evaluation of $dR/d\alpha$ via DST methodology.⁴ The adjoint system solved is of the form

$$L^* u^* = s^* \quad (1)$$

where L^* , u^* , and s^* are the adjoint operator, function, and source, respectively. The general sensitivity expression is of the form

$$\frac{dR}{d\alpha} = \int_V s^* u^* dV dt + BT \quad (2)$$

where s is the source term for the differentiated physical system and BT denotes boundary terms.

Consider a peak fuel response,

$$R = \int_V T(r, z, t) \delta(r - r_0) \delta(z - z_0) \delta(t - t_0) dV dt \quad (3)$$

where the phase-space location (r_0, z_0, t_0) is dependent on α . For brevity, consider only response shifts in the time domain. As shown in Ref. 5, an expression for $dt_0/d\alpha$ can be derived:

$$\frac{dt_0}{d\alpha} = - \left[\frac{\frac{\partial}{\partial t_0} \left(\frac{dR}{d\alpha} \right)}{\frac{\partial^2 T}{\partial t^2}} \right]_{(r_0, z_0, t_0)} \quad (4)$$

Note that a similar expression involving other maxima responses and phase-space variables can be derived.

The denominator of Eq. (4) is easily evaluated from the solution of the physical problem. The numerator of Eq. (4) is best obtained by differentiating Eq. (3):

$$\frac{\partial}{\partial t_0} \left(\frac{dR}{d\alpha} \right) = \int_V \left[\frac{\partial u^*}{\partial t_0} \right] dV dt + \frac{\partial}{\partial t_0} (BT), \quad (5)$$

where $\partial u^* / \partial t_0$ satisfies

$$L^* \left[\frac{\partial u^*}{\partial t_0} \right] = \left[\frac{\partial s^*}{\partial t_0} \right] \quad (6)$$

Note that for the response of Eq. (3), the only nonzero element of s^* is $\delta(r - r_0) \delta(z - z_0) \delta'(t - t_0)$ (Ref. 4). Thus, the only nonzero element of $\partial s^* / \partial t_0$ is $-\delta(r - r_0) \delta(z - z_0) \delta'(t - t_0)$, and so Eq. (6) can be solved with the adjoint MELT code by correctly specifying the adjoint source. The $\delta'(t - t_0)$ term used in the adjoint source was numerically applied in a manner consistent with the definition of Ref. 6.

A Fast Flux Test Facility protected transient involving a 0.23 dollar/s ramp reactivity insertion

with scram on high power was selected for investigation. The peak fuel temperature occurred at $t_0 = 0.87$ s. Adjoint calculations were performed for solution of Eqs. (1) and (6), from which $dR/d\alpha$ and $dt_0/d\alpha$ were obtained via Eqs. (2), (4), and (5). Figure 4.7.1 shows a profile of the portion of the adjoint solution of Eq. (6) associated with the coolant energy conservation equation. The ripple at $t \approx 0.87$ s is caused by the adjoint source, while the contour changes at $t \approx 0.5$ s are related to a source connected with the reactor trip.⁴ Table 4.7.1 shows results for the parameters that cause the greatest time shift in the response. The second and third columns show the first-order DST predictions for the response magnitude change and the time shift. The fourth and fifth columns indicate magnitude changes and time shifts obtained by direct recalculation with $\alpha + \Delta\alpha$ as input. The results provide adequate validation of the time shifts predicted with DST methodology.

ORNL Prog. 80-17999

Fig. 4.7.1. Adjoint Profile Corresponding to the Coolant Energy Conservation Equation.

In conclusion, it should be noted that only two adjoint calculations were necessary to calculate the response magnitude change and time shift for all the MELT parameters. This paper has shown that once the adjoint code is available, only a simple source modification is needed to allow prediction of the phase-space movement of maxima responses.

Table 4.7.1. Sensitivity Comparison Featuring Peak Fuel Temperature Response

Input Parameter, α	DST		Recalculation	
	$\frac{\Delta R}{\Delta \alpha / \alpha}$	$\frac{\Delta t_o}{\Delta \alpha / \alpha}$	$\frac{\Delta R}{\Delta \alpha / \alpha}$	$\frac{\Delta t_o}{\Delta \alpha / \alpha}$
	$\frac{K}{.5\%}$	$\frac{s}{.5\%}$	$\frac{K}{.5\%}$	$\frac{s}{.5\%}$
Initial Fuel Temperature	10.2	0.021	10.0	0.02
Scram Power	1.27 $\frac{K}{.3\%}$	0.012 $\frac{s}{.3\%}$	1.3 $\frac{K}{.3\%}$	0.01 $\frac{s}{.3\%}$
Initial Power	-0.08 $\frac{K}{.4\%}$	-0.012 $\frac{s}{.4\%}$	-.1 $\frac{K}{.4\%}$	-0.01 $\frac{s}{.4\%}$
Fuel Conductivity	-17.75 $\frac{K}{5\%}$	-0.024 $\frac{s}{5\%}$	-18.3 $\frac{K}{5\%}$	-0.025 $\frac{s}{5\%}$

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4.8. APPLICATION OF DIFFERENTIAL SENSITIVITY THEORY TO A NEUTRONIC/THERMAL-HYDRAULIC REACTOR SAFETY CODE*

C. V. Parks[†] P. J. Maudlin[‡]

(Abstract of ORNL/CSD/TM-154, July 1981)

A recently proposed sensitivity technique called differential sensitivity theory is applied to

the neutronic/thermal-hydraulic fast reactor safety code MELT-III. This application centralizes upon the development and solution of the appropriate adjoint and sensitivity equations, resulting in an adjoint version of the MELT code called MELTADJ. Proper integration of the forward MELT solution with the corresponding adjoint MELTADJ solution formally yields sensitivity information for all input parameters.

Two transients in the Fast Flux Test Facility were investigated by performing input parameter sensitivity analyses. Sensitivities obtained via MELTADJ are compared with those from MELT recalculations using perturbed input. These investigations indicate sufficiently good agreement between differential sensitivity theory and recalculation to validate the development of MELTADJ.

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4.9. THERMAL-HYDRAULIC DIFFERENTIAL SENSITIVITY THEORY*

P. J. Maudlin[†] C. V. Parks[‡] C. F. Weber[‡]

(Abstract of ORNL/CSD/TM-126, September 1981)

Differential sensitivity theory is applied to a nonlinear, three-dimensional space- and time-

dependent description of the thermal-hydraulic conservation equations. The resulting sensitivity equations, which are derived using adjoint functions, can be readily utilized for input parameter sensitivity analysis of large or long-running thermal-hydraulic computer codes without any prior engineering judgment. The procedure for applying this sensitivity theory is illustrated using several classical analytical problems.

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4.10. UNCERTAINTY ANALYSIS OF TIME-DEPENDENT NONLINEAR SYSTEMS: THEORY AND APPLICATION TO TRANSIENT THERMAL-HYDRAULICS*

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M. A. Bjerke[‡] C. B. Mullins[‡]

[Abstract of Nucl. Sci. Eng. 81, 23 (1982)]

An advanced methodology for performing systematic uncertainty analysis of time-dependent nonlinear systems is presented. This methodology includes a capability for reducing uncertainties in system parameters and responses by using Bayesian inference techniques to consistently combine prior knowledge with additional experimental information. The determination of best estimates for the system parameters, for the responses, and for their respective covariances is treated as a time-dependent constrained minimization problem. Three alternative formalisms for solving this problem are developed. Two "off-line" formalisms, with and without "foresight" characteristics, require the generation of a complete sensitivity data base prior to performing the uncertainty analysis. The "on-line" formalism, in which uncertainty analysis is performed interactively with the system analysis code, is best suited for treatment of large-scale highly nonlinear time-dependent problems.

This methodology is applied to the uncertainty analysis of a transient upflow of a high pressure water heat transfer experiment. For comparison,

an uncertainty analysis using sensitivities computed by standard response techniques is also performed. The results of the analysis indicate the following:

1. Major reduction of the discrepancies in the calculation/experiment ratios is achieved by using the new methodology.
2. Incorporation of in-bundle measurements in the uncertainty analysis significantly reduces system uncertainties.
3. Accuracy of sensitivities generated by response-surface techniques should be carefully assessed prior to using them as a basis for uncertainty analyses of transient reactor safety problems.

Conclusions about the future applicability of the uncertainty analysis methodology presented in this work are also discussed.

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4.11. ADJOINT SENSITIVITY ANALYSIS FOR TRANSIENT TWO-PHASE FLOW*

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(Abstract of paper accepted for publication in Nuclear Science and Engineering)

A rigorous formalism is presented for sensitivity analysis of functional-type responses associated with the well-posed system of quasilinear partial differential equations (PDE) of hyperbolic type that describe one-dimensional two-phase flows. The rigor and generality of this formalism stem from the use of G-differentials. In particular, it is possible to treat discontinuities, and parameters that are functions rather than scalars. This formalism uses adjoint functions to determine efficiently sensitivities to many parameter variations. The adjoint system satisfied by these adjoint functions is explicitly determined and shown to be solvable as a well-posed system of

linear first-order PDE's possessing the same characteristics as the original quasilinear PDE's. For completeness, a general solution of this adjoint system is obtained by using the method of characteristics. The physical meaning of this sensitivity analysis formalism is illustrated by an application to the homogeneous equilibrium model for two-phase flow. Although this formalism does not address transition phenomena between single- and two-phase flow regimes, and ignores higher-order effects of parameter variations, it provides a complete theoretical framework for implementing an efficient sensitivity analysis capability into one-dimensional two-phase flow models.

*Research sponsored by U.S. DOE Division of Reactor Research and Technology.

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4.12. SENSITIVITY ANALYSIS OF TRANSIENT TWO-PHASE FLOW SYSTEMS*

E. Wacholuer[†] S. Kaiserman[†] D. G. Cacuci

(*Trans. Isr. Nucl. Soc.* 9, 122, 1982)

A sensitivity analysis of a sample transient two-phase flow problem is presented. The phase-change occurs across a time-dependent boiling boundary. The method of characteristics is used to obtain an exact solution for the differentiated form of the nonlinear partial differential equations modeling this sample problem. Time- and space-dependent sensitivity coefficients for the state functions are calculated, analyzed, and used to predict effects of parameter variations on the state functions. Verifications of these predictions are performed by direct recalculations with altered parameter values.

*Research sponsored by U.S. Department of Energy Division of Reactor Research and Technology.

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4.13. APPLICATIONS OF DIFFERENTIAL SENSITIVITY THEORY FOR EXTREMUM-TYPE RESPONSES*

D. G. Cacuci[†] P. J. Mandlin[†] C. V. Parks[†]

(*Abstract of ORNL/TM-7717 May 1982*)

A recently developed sensitivity theory for nonlinear systems with responses defined at critical points, e.g. maxima, minima, or saddle points, of a function of the system's state variables and parameters is applied to a protected transient with scram on high power level in the Fast Flux Test Facility. The single-phase segment of the fast reactor safety code MELT-III B is used to model this transient. Two responses of practical importance, viz. the maximum fuel temperature in the hot channel, and the maximum normalized reactor power level, are considered. For the purposes of sensitivity analysis, a complete characterization of such responses requires consideration of both the numerical value of the response at the maximum and the location in phase-space where the maximum occurs. This is because variations in the system parameters alter not only the value at this maximum but also the location of the maximum in phase-space.

Expressions for the sensitivities of the numerical value of each maximum-type response, and expressions for the sensitivities of the phase-space location at which the respective maximum occurs, are derived in terms of adjoint functions. The adjoint systems satisfied by each of these adjoint functions are derived and solved. It is shown that the complete sensitivity analysis of each maximum-type response requires (a) the computation of as many adjoint functions as there are non-zero components of the maximum in phase-space, and (b) the computation of one additional adjoint function for evaluating the sensitivities of the numerical value of the response. The same computer code can be used to calculate all the

required adjoint functions. Once these adjoint functions are available, the sensitivities to *all* possible variations in the system parameters are obtained by quadratures. The sensitivities obtained by this efficient method are used to predict both changes in the numerical values of these maximum-type responses and the new phase-space location at which the perturbed maxima occur when the system parameters are varied. These predictions are shown to agree well with direct recalculations using perturbed parameter values.

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4.14. SENSITIVITY ANALYSIS OF A RADIATIVE-CONVECTIVE MODEL BY THE ADJOINT METHOD*

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M. E. Schlesinger[†]

(Abstract of Journal of Atmospheric Sciences, in press)

The adjoint method of sensitivity analysis is demonstrated on a radiative-convective climate model. A single adjoint calculation, which requires about the same computation time as the original model, suffices to calculate sensitivities of surface-air temperature to all 312 model parameters. The customary uses of these sensitivities are discussed and illustrated. The sensitivities accurately predict the effect on surface-air temperature of small variations in the model parameters. Relative sensitivities are used to rank the importance of all the parameters. Several of the sensitivities to parameters customarily considered in previous works (e.g. solar constant, surface albedo, relative humidity, CO₂ concentration) are reproduced, but the largest sensitivities are to constants used to compute the saturation vapour pressure of water. The uncertainties in the model results are expressed formally in terms of sensitivities and parameter covariances.

The radiative-convective model contains complex on-off nonlinear processes of the type found in general circulation models. Therefore, the fact that the adjoint method works successfully and

efficiently for the radiative-convective model provides valuable information about subsequent application of the method to general circulation models.

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4.15. DERIVATION AND APPLICATION OF SECOND ORDER UNCERTAINTY ANALYSIS FOR NORMALLY DISTRIBUTED PARAMETERS*

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(Abstract of Ann. Nucl. Energy (submitted))

A second order uncertainty analysis methodology for problems involving generally correlated but normally distributed parameters is presented. All quantities required to apply this uncertainty analysis methodology are expressed analytically in terms of the customary covariance matrices and in terms of first and second order sensitivities. An application of this methodology to evaluate the variances of the initial conversion ratio and the infinite multiplication factor for a PWR unit cell is also presented.

*Research sponsored by Electric Power Research Institute.

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4.16. EXPERIMENTAL AND CALCULATIONAL RESULTS OF AN IRRADIATION EXPERIMENT IN EBR-II*

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S. Ramas[‡] P. H. Stelson[‡] R. L. Walker[‡]
M. L. Williams

(Reprint of Trans. Am. Nucl. Soc. 38, 669, 1981)

During the period from 1967 to 1970, several actinide samples were irradiated by Argonne National Laboratory (ANL) in the Experimental Breeder Reactor II (EBR-II) fast reactor for $\sim 10^{23}$ n/cm². In 1978, Oak Ridge National Laboratory began an experimental determination of the exposure products in the irradiated samples, and to date, 12 of the samples have already

been analyzed. In a previous paper,¹ preliminary experimental results for a ^{239}Pu sample were compared with the exposure products predicted by calculations. In this paper, the comparison is extended to samples of ^{238}U , ^{233}U , ^{232}Th , ^{240}Pu , ^{237}Np , and a second ^{239}Pu sample, with the calculations utilizing the recently released ENDF/B-V data and improved modeling approximations. The experimental results presented in this paper represent a fairly comprehensive set of measurements of actinide burnup in a fast reactor spectrum.

One of the main goals of the present study is to establish and examine techniques (both experimental and calculational) for use in a future program to irradiate many "exotic" actinide samples (e.g., americium and curium) in the United Kingdom Prototype Fast Reactor (PFR), and to study their exposure products.

To achieve this goal, the following specific areas were examined:

1. the adequacy of experimental methods for analyzing small samples containing impurities of the built-up actinides to be studied
2. the adequacy of using simple modeling methods for burnup calculations
3. use of depletion sensitivity theory to focus investigation on the most important nuclear data.

In addition, a comparison was made of ENDF/B-V and IV one-group cross sections, each averaged over its appropriate EBR spectrum, to examine the effects of recent cross-section improvements.

The EBR experimental configuration consisted of an experimental pin in which 12 nickel capsules containing small amounts of actinides (~100 mg) were stacked vertically along the entire core height. Eight other samples were stacked in the axial blanket in the same pin below the first ones. The pin was located at approximately two-thirds of the core radius out from the center.

To determine the nuclide composition of the discharged samples, they were first dissolved in

HNO_3 . Then, isotopic measurements were made using a two-stage thermal emission mass spectrometer² equipped with a pulse-counting detection system for high sensitivity. The quantitative determination of plutonium and uranium was done by isotope dilution spectrometry using enriched spikes of ^{233}U , ^{236}U , and ^{242}Pu . The resin-bead method³ was used for the uranium and plutonium isotopic analyses. Alpha counting was used for ^{237}Np , ^{238}Pu , and ^{231}Pa measurements.

The calculational procedure consisted of two main steps. In a first step, one-group cross sections were collapsed from the 174-group VITAMIN-E library, which is based on ENDF/B-V. For resonance shielding, the Bondarenko method was used; however, this effect was found to be small. The flux spectrum used for the collapsing was computer with transport theory using a simplified, three-region spherical model⁴ of EBR-II. Several modifications of the basic model were also examined to account for changes made to the blanket during irradiation. It was generally found that the spectra near the core center were insensitive to geometric approximations; however, this did not apply to regions near the core-blanket interface.

Assuming a nearly time-independent flux spectrum and shape (this might be justified by examining the burnup history⁵ of EBR-II), the one-group cross sections derived in the first step were then used in successive depletion calculations. Based on the power history of the reactor, a separate calculation for each sample was done with the ORIGEN code,⁶ also taking into account the spatial distribution of the flux.

The calculated-to-experimental (C/E) ratios for the discharged samples (including a decay period of 8 to 10 yr) are given in Table 4.16.1. As can be seen, the degree to which the calculations and experiments agree is not uniform for all samples. For samples near the core midplane (^{239}Pu , ^{240}Pu , and ^{233}U), the calculations are usually within 10% of the measured values. However, for samples near the axial blanket (^{238}U , ^{232}Th , and ^{237}Np), the agreement is poor (perhaps because of oversimplified approximations made in modeling the reactor). The same calculations carried out with ENDF/B-IV data did not signi-

Table 4.16.1. C/E Results for Samples Irradiated in the EBR-II Core

Major Isotope	^{232}Th	^{238}U	$^{239}\text{Pu}-1^a$	$^{239}\text{Pu}-2$	$^{233}\text{U}^c$	^{240}Pu	^{237}Np
Axial ^b Location	1	2	6	8	9	10	12
Nuclide	C/E (calculated to experimental) Ratios						
^{231}Pu	1.120						
^{233}U	.626				1.070		
^{234}U	.442				.975		
^{235}U					1.040		
^{236}U					.285		
^{238}U		.998					
^{237}Np							1.073
^{238}Pu							.590
^{239}Pu		.818	1.000	1.020		1.025	
^{240}Pu		.719	.953	.983		1.020	
^{241}Pu		1.252	1.020	.976		.897	
^{242}Pu			.968	1.100		1.010	
^{241}Am				1.400		.909	

^aExperimental analysis done at ANL.^bAxial locations are counted from top to bottom.

significantly change this situation, although differences up to several percent were observed between the libraries.

To better understand the reasons for the disagreement between some experiments and calculations, sensitivity coefficients for all input parameters were computed. It could be shown that "built-up" nuclides are mostly sensitive to their initial charge, so that inaccurate determination of the initial composition or disregarded impurities might lead to misinterpretation of the measurements. Extreme care should therefore be taken to identify all sources of contamination in the experimental measurements before as well as after irradiation. This might be the most important message of this study for future fast reactor irradiation experiments (i.e., the PFR program).

Other conclusions from this study are the following:

1. For small samples with moderate burnup, meaningful experimental results can only be expected for two to three nuclides up the buildup chain.
2. Whenever possible, irradiated samples should be located in well-defined reactor regions (i.e., reactor center) to allow simple reactor modeling to be adequate.

Even multidimensional calculations will become useless when the sample environment is too complex or changing too often.

3. Insertion of nonactinide dosimeter capsules with the experimental samples is desirable to better determine the flux magnitude (and to a lesser extent, the flux spectrum).
4. A combination of mass spectrometry and alpha counting was found to give adequate experimental results. Alpha counting is especially important for measuring ^{238}Pu [needed to determine the $^{239}\text{Pu}(n,2n)$ reaction and $^{241}\text{Am}(n,\gamma)$ branching ratio], which is easily contaminated by ^{238}U .

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4.17. SENSITIVITIES AND UNCERTAINTIES OF A LARGE LMFBR USING THE ADJUSTED CROSS-SECTION LIBRARY ORACLE-1*

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[Reprint of *Trans. Am. Nucl. Soc.* 41, 578, 1982; also summary of invited paper to be presented at the *ANS Topical Meeting on Advances in Reactor Physics and Core Thermal Hydraulics*, Kiamesha Lake, NY, September 22-24, 1982 (with D. G. Cacuci)]

The principal purpose of this work is to investigate uncertainties and sensitivities associated with calculated values of significant performance parameters in fast reactors. The joint work reported here is a continuation of a cooperative effort between General Electric Company, Georgia Institute of Technology, and the Oak Ridge National Laboratory. Earlier work is reported in Refs. 1-4. Related work has been associated with the development of a least-squares adjusted cross section and covariance library ORACLE-1⁵ for application to fast reactor design.

A final version of ORACLE-1 has been completed. (A preliminary version was described in Ref. 5.) This library is based on the *a priori* 174-group Vitamin-E cross section library, *a priori* covariances derived from ENDF/B-5 and other sources, and twenty-seven benchmark integral experiments. This adjusted library consists of not only adjusted cross sections in 174-group structure, but also includes adjusted cross section covariances in 26-group structure.

ORACLE-1 cross sections and covariances have been applied to a model of a LDP-type heterogeneous reactor⁴ using the VENTURE,⁶

DEPTH-CHARGE,⁷ and FORSS⁸ code systems. Previously reported work⁴ considered the application to this model of cross sections and covariances based on ENDF/B-IV. In the work reported here not only have the cross section and covariance been changed, but the performance parameters considered in the sensitivity and uncertainty analysis include sodium-void reactivity as well as k_{eff} , breeding ratio, and power-density ratios.

The general characteristics of k_{eff} and breeding-ratio sensitivities, including the influence of the plutonium enrichment "k-reset" mechanism, have been discussed in previous references.^{2,3,4} In general, these sensitivity coefficients are within about fifteen percent of those reported in Ref. 4, which were calculated with ENDF/B-IV data for the same reactor model.

Power density (PD) sensitivities were calculated at the point of peak power for each of three driver zones. Note that an X-Y model would be more appropriate than the R-Z model for detailed PD studies. These PD sensitivities agree with values reported in Ref. 4 to within ~20% and have the following general characteristics:

1. The PD sensitivities are quite large for those reactions that influence the neutron transport between zones.
2. The PD sensitivities for a given reaction are quite different in each driver zone.
3. For the significant reactions, the PD sensitivity coefficients for the middle driver zone are generally much smaller than for the other two driver zones.

Selected total (energy-integrated) relative sensitivity coefficients for the sodium-void reactivity are presented in Table 4.17.1. These sensitivities were calculated using a capability developed only recently. As expected, these sensitivities are large for nuclear data which have large k_{eff} sensitivity coefficients. Besides various heavy-metal reactions, the scatter sensitivities for sodium, oxygen, and iron are large. The values of these sensitivities for the scatter cross sections were strongly influenced by the changes in the associated tran-

Table 4.17.1. Selected Total Relative Sensitivity Coefficients for the Sodium-Void Reactivity of the LDP Core Using the ORACLE-I Data Set

Nuclide	Reaction	Na Void Sensitivity	Nuclide	Reaction	Na Void Sensitivity
U-235	ν	-1.1 E-2	Pu-239	ν	-1.7
	σ_f	-2.9 E-3		σ_f	-9.9 E-1
	σ_c	6.5 E-3		σ_c	9.0 E-1
	σ_{el}	2.6 E-4		σ_{el}	1.6 E-2
	σ_{inel}	-1.1 E-3		σ_{inel}	-3.6 E-2
U-238	ν	1.1	Pu-240	ν	2.9 E-1
	σ_f	6.5 E-1		σ_f	1.9 E-1
	σ_c	9.8 E-1		σ_c	2.8 E-1
	σ_{el}	1.4 E-1		σ_{el}	3.7 E-3
	σ_{inel}	-6.4 E-1		σ_{inel}	-1.5 E-2
Pu-241	ν	-7.1 E-1	Pu-242	ν	2.9 E-2
	σ_f	-4.5 E-1		σ_f	1.9 E-2
	σ_c	8.5 E-2		σ_c	2.5 E-2
	σ_{el}	1.3 E-3		σ_{el}	4.7 E-4
	σ_{inel}	-8.1 E-3		σ_{inel}	-2.1 E-3
Na-23	σ_c	1.1 E-1	O-16	σ_{el}	-5.6 E-1
	σ_{el}	-1.8 E-1		σ_{inel}	-6.6 E-3
	σ_{inel}	6.5 E-1			
Fe	σ_c	1.0 E-1	Ni	σ_c	2.1 E-2
	σ_{el}	5.3 E-3		σ_{el}	-6.5 E-2
	σ_{inel}	-2.2 E-1		σ_{inel}	-4.7 E-2
Cr	σ_c	3.6 E-2	Mo	σ_c	4.8 E-2
	σ_{el}	3.0 E-2		σ_{el}	5.7 E-3
	σ_{inel}	-5.0 E-2		σ_{inel}	-1.2 E-2

sport cross sections. Our convention is to treat partial cross sections as independent variables and total cross sections as dependent variables.

Combining the ORACLE-I-based sensitivities and covariances resulted in performance parameter uncertainties shown in Table 4.17.2. For the sake of comparison, previously reported uncertainties⁴ based on unadjusted ENDF/B-IV data are also shown. The reduction of the uncertainties is largely due to the effect of including the results of integral experiments. It must be emphasized that these uncertainties include only the effects of uncertainties in nuclear data as modified by the incorporation of integral experiments. Covariance contributions due to thermal-

hydraulic and other engineering design uncertainties have not been included.

In summary, the above reports the application of the recently-developed adjusted cross-section and covariance library ORACLE-I to the calculation of sensitivities and uncertainties of performance parameters in LDP-type heterogeneous fast reactors. Based on new 2-D calculational capability these parameters include sodium-void reactivity as well as k_{eff} , a breeding ratio, and power density ratios. A comparison with previously reported uncertainties based on unadjusted data shows that the inclusion of integral experiments is significant for the reduction of uncertainties due to nuclear data.

**Table 4.17.2. Performance Parameter
Uncertainties Based on ORACLE-I**

Parameter	Standard Deviation, % (No K-Reset)	
	ORACLE-I	ENDF/B-IV (Ref. 4)
k_{eff}	0.4	3
BR	2	7
PD (Inner Driver)	1	2
Sodium-Void (All Drivers)	14	--

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†UCC-ND Computer Science Division.

‡Georgia Institute of Technology.

§General Electric Company, Sunnyvale, California.

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Section 5

INTEGRAL EXPERIMENTS AND NUCLEAR ANALYSES

(Supporting the Design and Utilization of Nuclear Systems)

Integral Experiments Supporting Fusion Reactor Designs

Nuclear Analyses Supporting Fusion Reactor Designs

High-Energy Particle Transport Calculations

Integral Experiments Supporting Gas-Cooled Fast Breeder Reactor Designs

Nuclear Analyses Supporting Gas-Cooled Reactor Designs

Nuclear Analyses Supporting Utilization of Light-Water Reactors

Integral Experiment Analyses Supporting Surveillance Dosimetry Improvement Program

Miscellaneous Nuclear Analyses

5.0. INTRODUCTION

The nuclear data discussed in Section 1, the multigroup libraries described in Section 2, the radiation transport calculational methods presented in Section 3, and the sensitivity and uncertainty analyses techniques described in Section 4 all contribute to a capability for analyzing systems in which the production and/or transmission of radiation must be determined. This section presents papers describing results obtained when that nuclear analysis capability is applied to specific systems. It also includes integral experiments that provide information on the production, reduction, and transport of radiation.

During this reporting period the systems of interest have been associated with the magnetic fusion energy program, with certain systems involving high-energy particle transport, and with the several types of fission reactors that are operating, under design, or being considered as an option in the future. Not included here are analyses performed for the Liquid Metal Fast Breeder program. These will be found in Section 9, which is published separately as Volume II of this report.

Magnetic Fusion Energy Program

The division's magnetic fusion energy program includes integral experiments and their analyses, neutronics calculations in support of radiation damage studies and tritium recovery, and a variety of other design calculations.

The integral experiments utilize a 14-MeV neutron source produced by the interactions of 250-keV deuterons with tritium saturated in a titanium target. They are designed to provide data against which ORNL and other organizations involved in fusion reactor design can test their methods and nuclear data. The principle design problems are associated with neutron streaming through the large ducts that are required in fusion reactor blankets and shields to accommodate neutral beam injectors, vacuum pumps, etc. As noted in papers included in this section, measurements have been made of 14-MeV neutrons streaming through iron ducts embedded in concrete, with the ducts both open and partially plugged. In addition, measurements have been made on the transmission of 14-MeV neutrons through several shield configurations comprised of one or more of the following materials: lithium hydride, lead, stainless steel, borated polyethylene, and Havimet. In all cases the experiments have included measurements of spectra of neutrons and the secondary gamma rays produced by the neutron interactions. And, of course, analyses of the experiments have been performed.

Other studies in the magnetic fusion energy program have consisted in radiation damage and tritium recovery calculations carried out to aid in the design of experiments being conducted at the Oak Ridge Research Reactor. The ORNL Metals and Ceramics Division is conducting experiments to obtain radiation damage data for metals that will be used in fusion reactors, and investigators from the Argonne National Laboratory are carrying out experiments to study the recovery of tritium when lithium-bearing materials are irradiated.

High-Energy Physics Program

The division's calculations of the transport of high-energy particles continue to be performed at the request of groups outside of ORNL. With the methods and interaction data that have been developed and the experience that has been gained, calculations are routinely performed at ORNL that were not possible a few years ago. During this reporting period, shielding calculations have been carried out to aid in the design of a proposed neutron oscillation experiment at the Oak Ridge Research Reactor and in the design of a neutrino experiment at the Rutherford Laboratory in England.

Work in the high-energy program is also directed at aiding in the design of instrumentation (ionization calorimeters) to measure the energy of various high-energy particles. The calorimeters that have been built and tested at accelerators have all been found to be satisfactory, which has encouraged the development of more sophisticated (and more complicated) devices, each design being for a different purpose. The agreement of our calculation with the experimental tests is especially gratifying since the instruments are very large and expensive, and "testing" design variations with a series of calculations rather than with numerous mockups is much more economical.

As part of the program during the past year, a model of high-energy fission has been incorporated into the ORNL high-energy transport code HETC (see Section 3), and the validity of the model has been demonstrated by comparing calculated results with experimental data. The use of fissionable targets in pulsed neutron sources is being studied at several institutions, both in the U.S. and in Europe, and this revised code is now available for use in these studies.

Integral Experiments Supporting Gas-Cooled Fast Breeder Reactor Designs

Between 1975 and 1980 a series of experiments were performed at the ORNL Tower Shielding Facility in support of the shield design for a 300-MW(e) Gas Cooled Fast Breeder Demonstration Plant. During this reporting period descriptions of the last few experiments and their analyses have been published. In addition, a report reviewing the entire program has been published. Although the GCFR program has now been cancelled, much of the information in these reports is generic and therefore applicable to other systems. The measurements dealt with neutron streaming in core helium coolant passageways, with the effectiveness of a shield to protect the reactor grid plate, with the adequacy of a radial shield in protecting the prestressed concrete reactor vessel, with neutron streaming between abutting sections of a radial shield, and with the effectiveness of an "exit shield" in reducing neutron fluxes in the plenum region of a gas-cooled reactor.

Nuclear Analyses Supporting Gas-Cooled Reactor Designs

The division continues to perform nuclear analyses in support of the designs of gas-cooled reactors. During this reporting period these have included the GCFR, the U.S. High-Temperature Gas-Cooled Reactor (HTGR) and Germany's Pebble Bed Reactor (PBR).

For the GCFR a detailed analysis was performed for several proposed radial shield designs. These designs were complicated by the fact that the radial shield had to include penetrations to allow natural circulation of the coolant in the event of the loss of forced circulation. The analyses showed that the "final" design met all the design criteria.

For the HTGR a survey analysis of radiation levels within General Atomic's conceptual 900-MW(e) steam-cycle HTGR was performed. The analysis indicated a problem in neutron streaming in the core support structural area, resulting in high doses to the side wall liner. However, large uncertainties existed in the calculations and an experimental investigation has been recommended.

For the PBR a study was made of the stability of the reactor against xenon-driven oscillation. Special calculational methods were implemented. A core as large as 3000 MW(th) could be stable, the size where special control would be necessary being in this neighborhood.

Nuclear Analyses Supporting Utilization of Light-Water Reactors

The division is becoming increasingly involved in studies related to the operation and safety of light-water reactors (see also Section 7). In one study carried out during 1980-81, division staff invented a technique (patent pending) for using a modified version of a standard gamma thermometer to measure the adequacy of a PWR core cooling mechanism without compromising the original function of the instrument. That is, a dual-function gamma thermometer is proposed.

Another study was aimed at determining whether depletion perturbation theory (DPT) would work as well for estimating changes in various reactor responses (power ratios, reactivity, etc.) for LWRs as it does for fast reactors. The conclusion was that static DPT did not work well for localized perturbations, but that time-dependent DPT gave good results for such things as fuel shuffling. For global perturbations, both static and time-dependent perturbation theory produces useful results.

Integral Experiment Analyses Supporting Surveillance Dosimetry Improvement Program

In support of the development of a methodology for making reliable estimates of the neutron fluences within the wall of a PWR pressure vessel (see Paper 3.12), funded by the Electric Power Research Institute, the division has been engaged in analyzing a series of benchmark experiments in which neutron dosimeter measurements are made. In the methodology development, the approach is to compare measured and calculated dosimeter responses and then to systematically adjust the data used in the calculations so that the two responses are in better agreement. The practical result will be a smaller uncertainty associated with the predicted lifetimes of pressure vessels determined from dosimeter measurements made at surveillance locations.

The first calculations were for dosimeter responses in a simple experiment in which a ^{252}Cf source was used. Subsequently calculations were performed for a series of measurements made at the ORNL Pool Critical Assembly, which is a very low-power critical facility. For the period

covered by this report, calculations have primarily been made for responses measured in the Intermediate Energy Standard Neutron Field (ISNF) at the National Bureau of Standards and at the ORNL Poolside Facility with the Oak Ridge Research Reactor (ORR) used as the source. These most recent calculations, funded by the Nuclear Regulatory Commission, serve the double purpose of acting as support calculations by the ORR measurements, as well as providing an expanded data base that can be used in the adjustment methodology. Currently, we are calculating the dosimeter responses obtained at Arkansas Nuclear One, Unit 1.

Miscellaneous Nuclear Analyses

During the past 18 months several investigations have been carried out that have not been directly related to specific reactor programs. Three such calculations have been aimed at improving our ability to calculate radiation penetration through iron and/or concrete shields. One consisted of a Monte Carlo analysis of a time-dependent neutron and secondary gamma-ray integral experiment on a thick concrete and steel shield. Another, still under way, consists in the development of a simple calculational method for predicting total doses due to neutrons penetrating thick concrete shields; the methodology appears to be generic for deep-penetration problems and involves the application of an experimental correction factor to sensitivity predictions based on linear perturbation theory. The purpose of the third investigation was to develop a Monte Carlo sensitivity method that allows cross-section adjustment procedures to be applied more reliably and generally than had previously been possible.

Another brief study has been due to a recent upsurge of interest in the use of reactors for space applications. Several different reactor concepts are being considered, and the division expects to be involved with studies of the reactors in the areas of reactor physics and shielding. Shielding is an especially important constraint for space reactor systems.

Finally, the recent concern of the dose effects of neutrons, especially those from low neutron dose levels, has led to our participation in a series of studies aimed at reevaluating the doses received at Hiroshima and Nagasaki. Our primary involvement is performing air-over-ground transport calculations, as is discussed in two papers included in this report.

Integral Experiments Supporting Fusion Reactor Designs

5.1. CALCULATED NEUTRON AND GAMMA RAY ENERGY SPECTRA FROM 14-MeV NEUTRONS STREAMING THROUGH AN IRON DUCT: COMPARISON WITH EXPERIMENT*

R. T. Santoro R. G. Alsmiller, Jr.
J. M. Barnes[†] G. T. Chapman
J. S. Tang[†]

[Abstract of ORNL/TM-7878, September 1981; also Nucl. Sci. Eng. 80, 586 (1982)]

Integral experiments to measure the streaming of 14 MeV neutrons through a 0.30-m-diameter iron duct (length-to-diameter ratio ~3) imbedded in a concrete shield have been carried out at the Oak Ridge National Laboratory. Calculated and measured neutron and gamma-ray energy spectra are compared at sixteen detector locations on and off the cylindrical axis of the duct. The measured spectra were obtained using an NE-213 liquid scintillator detector with pulse-shape discrimination to simultaneously resolve neutron and gamma-ray events. The calculated spectra were obtained using a computer code network that incorporates two radiation transport methods: discrete ordinates (with P_3 multigroup cross sections) and Monte Carlo (with continuous-point cross sections). The two radiation transport methodologies are required to properly account for neutrons that single scatter from the duct to the detector. The calculated and measured outgoing neutron energy spectra above 850 keV agree within 5 to 50% depending on detector location and neutron energy. The calculated and measured gamma-ray spectra above 750 keV are also in favorable agreement, ~5 to 50%, depending on detector location and gamma ray energy.

*Work Sponsored by U.S. DOE Office of Fusion Energy.
[†]UCC-ND Computer Sciences Division.

5.2. COMPARISON OF MEASURED AND CALCULATED NEUTRON AND GAMMA-RAY ENERGY SPECTRA BEHIND AN IN-LINE SHIELDED DUCT*

R. T. Santoro R. G. Alsmiller, Jr.
J. M. Barnes[†] G. T. Chapman
J. S. Tang[†]

(Abstract of ORNL/TM-8258, May 1981; also J. Fusion Energy, in press)

Integral experiments that measure the transport of ~14 MeV neutrons through a 0.30-m-diameter duct having a length-to-diameter ratio of 2.83 that is partially plugged with a 0.15 m diameter, 0.51 m long shield comprised of alternating layers of stainless steel type 304 and borated polyethylene have been carried out at the Oak Ridge National Laboratory. Measured and calculated neutron and gamma-ray energy spectra are compared at several locations relative to the mouth of the duct. The measured spectra were obtained using a NE-213 liquid scintillator detector with pulse shape discrimination methods used to simultaneously resolve neutron and gamma-ray events. The calculated spectra were obtained using a computer code network that incorporates two radiation transport methods: discrete ordinates (with P_3 multigroup cross sections) and Monte Carlo (with continuous point cross sections). The two radiation transport methods are required to account for neutrons that singly scatter from the duct to the detectors. The calculated and measured neutron energy spectra above 850 keV agree within 5 to 50% depending on detector location and neutron energy. The calculated and measured gamma-ray energy spectra above 750 keV are also in favorable agreement, ~5 to 50%, depending on detector location and gamma ray energy.

*Research sponsored by U.S. DOE Office of Fusion Energy.
[†]UCC-ND Computer Sciences Division.

5.3. CALCULATION OF NEUTRON AND GAMMA-RAY ENERGY SPECTRA FOR FUSION REACTOR SHIELD DESIGN: COMPARISON WITH EXPERIMENT II*

R. T. Santoro R. G. Alsmiller, Jr.
J. M. Barnes[†] G. T. Chapman

(Abstract of ORNL/TM-7927, November 1981; also *J. Fusion Energy*, in press)

Measured and calculated neutron and gamma-ray energy spectra resulting from the transport of ~ 14 MeV neutrons through a 0.30-m-thick lithium hydride slab and through a 0.05-m-thick lead slab followed by 0.30 m of lithium hydride are compared. Also reported are comparisons of the measured and calculated neutron energy spectra behind an 0.80-m-thick assembly comprised of stainless steel type 304 and borated polyethylene. The spatial dependence of the gamma-ray energy deposition rate measured using thermoluminescent detectors is compared with calculated data.

The calculated data obtained using two-dimensional radiation transport methods and ENDF/B-IV cross section data are in good agreement for all of the experimental configurations. Calculated integral neutron energy spectra agree with the measured data within ~ 5 to $\sim 20\%$ depending on neutron energy for the LiH and Pb plus LiH assemblies. The gamma-ray spectra agree within 20% for these slabs. The measured and calculated neutron energy spectra behind the SS-304-borated polyethylene assembly agree within $\sim 5\%$ except at neutron energies below ~ 5 MeV, where background radiation influences the measured spectra. The gamma-ray energy deposition rates as a function of depth agree within a factor of two at all detector locations.

*Research sponsored by U.S. DOE Office of Fusion Energy.
[†]UCC-ND Computer Sciences Division.

5.4. SENSITIVITY ANALYSIS FOR A SPECIFIC FUSION REACTOR SHIELDING EXPERIMENT CONTAINING TUNGSTEN*

R. G. Alsmiller, Jr. R. T. Santoro
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J. L. Lucas[†]

(Abstract of ORNL/TM-8105, in press; also *Technical Note, Nucl. Sci. Eng.* (submitted))

In a previous paper, calculated and experimental results for the transmission gamma-ray spectra when ~ 14 MeV neutrons are incident on a laminated slab configuration of stainless steel, borated polyethylene, and Hevimet (90 wt % W, 6 wt % Ni, 4 wt % Cu) were shown to be in substantial disagreement. In this paper the results of a cross section sensitivity analysis (based on the two-dimensional model of the experiment used in the previous paper) are presented. In the analysis only the nuclear cross section data for the elements in Hevimet were considered. The cross section sensitivities are found to be quite small and it is concluded that it is unlikely that the disagreement is due to errors in the cross section data used for the elements in Hevimet.

*Research sponsored by U.S. DOE Office of Fusion Energy.
[†]UCC-ND Computer Science Division.

5.5. CALCULATIONAL PROCEDURES FOR THE ANALYSIS OF INTEGRAL EXPERIMENTS FOR FUSION REACTOR DESIGN*

R. T. Santoro J. M. Barnes[†]
R. G. Alsmiller, Jr. E. M. Oblow

(Abstract of ORNL-5777, July 1981)

The calculational models, nuclear data, and radiation transport codes that are used in the analysis of integral measurements of the transport of ~ 14 MeV neutrons through laminated slabs of

materials typical of those found in fusion reactor shields are described. The two-dimensional discrete ordinates calculations to optimize the experimental configuration for reducing the neutron and gamma ray background levels and for obtaining an equivalent, reduced geometry of the calculational model to reduce computer code storage and running times are also presented. The equations and data to determine the energy-angle relations of neutrons produced in the reactions of 250-keV deuterons in a titanium-tritide target are given. The procedures used to collapse the 171n-36 γ VITAMIN C cross section data library to a 53n-21 γ broad group library are described. Finally, a description of the computer code network used to obtain neutron and gamma ray energy spectra for comparison with measured data is included. The network incorporates several two-dimensional discrete ordinates codes to treat first collisions (GRTUNCL), multiple collisions (DOT), and last collisions (FALSTF) of the neutrons and gamma rays in the transport sequence from the neutron source to the detector.

*Work sponsored by U.S. DOE Office of Fusion Energy.

[†]UCC-ND Computer Sciences Division.

5.6. ANALYSIS OF THE ORNL RADIATION STREAMING INTEGRAL EXPERIMENTS*

R. T. Santoro R. G. Alsmiller, Jr.
J. M. Barnes[†] G. T. Chapman
J. S. Tang[†]

(Reprint of *Trans. Am. Nucl. Soc.* 38, 561, 1981)

The determination of radiation streaming through the various penetrations that will be present in the blanket and shield of a deuterium-tritium (D-T)-burning fusion reactor is one of the more complicated problems facing nuclear design engineers. Penetrations of varying size and shape will be required for neutral particle injection, radiofrequency heating, vacuum pumping, and plasma diagnostics. The radiation streaming through these openings will degrade the reactor performance by the adverse effects of nuclear heating, induced activation, and radiation damage in vital components. The effects of radiation

streaming must be estimated during the design of the reactor. It is necessary, therefore, to have experimental verification of the nuclear data and radiation transport methods that will be used to carry out these calculations. Integral experiments are being performed at the Oak Ridge National Laboratory to provide this verification. Measurements have been made to determine the streaming of ~ 14 -MeV neutrons through a simple iron duct imbedded in a concrete shield. The verification of the nuclear data and radiation transport methods is made by extensive comparison between the measured and calculated neutron and gamma-ray energy spectra at several locations on and off the axis of symmetry of the duct.

The experiments are performed using ~ 14 -MeV neutrons produced from the interactions of 250-keV deuterons with tritium saturated in a titanium target. The neutron source strength is $\sim 10^9$ n/s. The target is enclosed in an iron can that tailors the neutron distribution to correspond to that incident on the first wall of a fusion reactor. A cylindrical iron duct, with an inner radius of 0.15 m, a wall thickness of 0.075 m, and a length-to-diameter (L/D) ratio of 3, extends from the source can. The assembly is imbedded in a concrete structure that simulates a shield surrounding the plasma. Neutron and gamma-ray pulse-height spectra were measured at 17 detector locations on and off the axis of symmetry of the duct using a 5- x 5-cm cylindrical NE-213 liquid scintillator. The neutron and gamma-ray data were obtained simultaneously using pulse-shape discrimination methods, and spectral unfolding techniques were used to resolve the energy spectra from the pulse-height data.

The calculated neutron and gamma-ray energy spectra are obtained using a computer code network that employs both discrete ordinates and Monte Carlo radiation transport methods. The experimental configuration is represented in r-z geometry with cylindrical symmetry about the deuteron beam-duct axis using 82 axial and 62 radial mesh intervals to describe the geometry. The D-T neutron source used in the calculation accounts for the angle-energy correlation of neutrons emitted from the reaction. The discrete

ordinates calculations are carried out on an S_{12} angular quadrature and a P_3 expansion of a 53-neutron, 21-gamma-ray multigroup cross-section set obtained by collapsing the 171-neutron, 36-gamma-ray VITAMIN-C data library.¹

The radiation transport calculations are initiated using a sequence of discrete ordinates codes consisting of GRTUNCL,² DOT,³ and FALSTF.⁴ GRTUNCL calculates uncollided neutron flux and first collision source distributions throughout the geometric mesh used to define the experimental geometry. These data are used as input to the DOT code, which calculates the collided neutron and secondary gamma-ray flux distributions and also writes a final scattering source for input to the FALSTF code. This code calculates the total neutron and gamma-ray flux at points external to the concrete and iron duct and has the advantage of eliminating ray effects that could arise from intense final scattering sites in the geometry. The neutron energy spectra were obtained from this sequence using a P_3 Legendre expansion of the scattering cross sections. The spectra at detector locations off the axis of symmetry do not compare favorably with those obtained from the experiment. The differences between the measured and calculated data are due to the inadequacy of the P_3 data for representing the elastic and inelastic to discrete level single scattering of neutrons from the iron duct to the detectors.

This phenomenon was corrected by calculating the single scattered neutron contribution using a version of MORSE⁵ that incorporates continuous (point) cross-section data rather than multigroup formatted data and that accesses ENDF/B-IV data files directly. These data files contain higher order Legendre expansions of the neutron scattering cross sections. Thus, the neutron flux at detector location r is obtained by evaluating expressions of the form

$$\phi(r) = \phi_c(r) - \phi_{P_1}^{SS} + \phi_{P_1}^{SS}(r),$$

where $\phi_c(r)$ is the neutron flux calculated by the GRTUNCL-DOT-FALSTF sequence and contains the uncollided, single scattered, and multiply collided neutron contributions. Here, $\phi_{P_1}^{SS}(r)$ is the single scattered contribution to the

flux obtained using discrete ordinates methods, and $\phi_{P_1}^{SS}(r)$ is the single scattered neutron flux obtained from MORSE (point cross-section calculation).

The calculated neutron energy spectra obtained using this procedure are compared with measured data in five detector locations in Fig. 5.6.1. The measured spectra are given by the curves, and the calculated data are given by the points. The two solid lines at each detector location indicate the statistical uncertainty band in

Fig. 5.6.1. Comparison of the Calculated and Measured Neutron Flux Per Unit Energy Versus Neutron Energy for Detector Locations at $z = 208$ cm and $r = 0, 59, 79, 98, \text{ and } 119$ cm.

the unfolded neutron spectrum. The calculated data were obtained by smoothing the flux per unit energy in each multigroup interval with an energy-dependent Gaussian response function characteristic of the neutron response of the detector. (The flux from the MORSE calculation was binned in energy intervals corresponding to the multigroup energy boundaries used in the discrete ordinates calculation.)

The data are compared in Fig. 5.6.1 when the detector is on the axis of symmetry ($r = 0$) and for four locations off the axis ($r = 59, 79, 98$, and 119 cm). The axial source-to-detector distance is 208 cm in all cases. The calculated and measured neutron energy spectra are in good agreement at all detector locations. Comparisons of the neutron and gamma-ray spectra at all 17 detector locations at which measurements were made are presented and discussed.

*Research sponsored by U.S. DOE Office of Fusion Energy.
 †UCC-ND Computer Sciences Division.

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5.7. ANALYSIS OF ORNL FUSION REACTOR BLANKET/SHIELD INTEGRAL EXPERIMENTS*

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(Reprint of *Trans. Am. Nucl. Soc.* 39, 798, 1981)

A series of integral experiments is being performed at the Oak Ridge National Laboratory (ORNL) to obtain experimental verification of the nuclear data and radiation transport methods that are being used in nuclear design calculations

for fusion reactors. In this paper, measured and calculated neutron and gamma-ray energy spectra resulting from the transport of ~ 14 -MeV neutrons through a 0.30 -m-thick lithium hydride slab and through a 0.05 -m-thick lead slab followed by 0.30 m of lithium hydride are compared. Also reported are comparisons of the measured and calculated neutron energy spectra behind a 0.80 -m-thick assembly comprised of Type 304 stainless steel and borated polyethylene. The spatial dependence of the gamma-ray energy deposition rate in the stainless-steel-borated-polyethylene assembly was also measured using thermoluminescent detectors, and these data are compared with calculated results. The lithium hydride and lead plus lithium hydride configurations are intended to simulate blanket module configurations, and the stainless-steel-borated-polyethylene assembly (which is a continuation of previously reported work¹) represents a thick shield assembly.

The experiments are carried out using 14 -MeV neutrons having a source strength of $\sim 10^8$ n/s produced in the interactions of 250 -keV deuterons with tritium saturated in a titanium target. The target is enclosed in an iron can, which has the purpose of tailoring the neutron distribution to correspond to that incident on the first wall of a fusion reactor.² The target, iron can, and the shield assembly are supported in a concrete superstructure. The concrete is sufficiently thick to act as an environmental shield and to minimize the radiation background levels in the vicinity of the radiation detector. The test slabs (1.52×1.52 m²) are inserted in a cavity in the concrete structure. Neutron and gamma-ray spectra are measured at various positions relative to the deuteron-target axis and as a function of the test slab composition using an NE213 liquid scintillator. Neutron and gamma-ray pulse-height spectra are simultaneously obtained using pulse-shape discrimination methods, and unfolding techniques are used to resolve the energy spectra from the pulse-height data.

The calculated neutron and gamma-ray energy spectra are obtained using two-dimensional radiation transport methods. The

experimental configuration is represented in r - z geometry with cylindrical symmetry about the axis of the deuteron beam using 92 axial and 76 radial mesh intervals to describe the geometry. The neutron source used in the calculations takes into account the angle-energy relationship of the neutrons emitted from the deuterium-tritium (D-T) reaction and takes into account deuteron slowing down in the target. Separate calculations are performed to determine the first collision and uncollided flux distributions in the geometry meshing using the GRTUNCL code.³ These data form the source term for input into the DOT discrete-ordinates code,⁴ which completes the radiation transport. The last flight of radiation to the detectors is calculated using the FALSTF code.⁵ The transport calculations are carried out using an S_8 angular quadrature and a P_3 Legendre expansion of the transport cross sections. A 53-neutron 21-gamma-ray multigroup cross section is obtained by collapsing the 171-neutron 36-gamma-ray Vitamin C (Ref. 6) data set using the ANISN code.⁷

The neutron energy spectra measured and calculated behind the 0.30-m-thick lithium hydride slab and behind the 0.05-m-thick lead plus 0.30-m-thick lithium hydride slab assembly are compared in Fig. 5.7.1. The measured data are indicated by the solid lines, and the points are the results of the calculation. The calculated data are obtained by smoothing the neutron flux per unit energy in each multigroup energy interval

Fig. 5.7.1. Measured and Calculated Neutron Energy Spectra as a Function of Neutron Energy for the Lithium Hydride and Lead Plus Lithium Hydride Slabs.

with an energy-dependent Gaussian function that characterizes the neutron response of the detector. For these data, the neutron detector is on the deuteron beam-target axis at a distance from the source of 1.81 m. The test slabs are interposed between the source and the detector at a distance from the source of ~ 1 m. The calculated and measured data are in fairly good agreement at all neutron energies between 850 keV and 15.1 MeV, but the calculated data are systematically lower than the measured spectra by 15 to 50%, depending on neutron energy. The kinematics of the DT reaction for 250-keV deuterons limits the energy of the emitted neutrons to 15.1 MeV, and the appearance of neutron flux above this energy is a manifestation of the Gaussian response of the detector. The difference between the calculated and measured spectra is the same for each slab composition and configuration. The presence of the lead results in a reduction in the magnitude of the spectrum but not the shape. The disagreement may be attributed to the transport cross-section data for lithium and/or hydrogen that were used in the calculations, but further investigation is required for confirmation.

Differential and integral neutron and gamma-ray energy spectra at detector locations both on and off the axis of symmetry will also be presented. The neutron energy spectra measured and calculated behind the 0.80-m-thick stainless-steel-borated-polyethylene assembly are in very good agreement at neutron energies between 850 keV and 15.1 MeV. These data were obtained to test the calculational procedures for the treatment of deep penetration of 14-MeV neutrons. The measured and calculated gamma-ray energy deposition rates as a function of depth in the thick stainless-steel-borated-polyethylene are also in good agreement.

*Research sponsored by U.S. DOE Office of Fusion Energy.

[†]UCC-ND Computer Sciences Division.

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Nuclear Analyses Supporting Fusion Reactor Designs

5.8. MONTE CARLO SIMULATION OF MOLECULAR FLOW IN A NEUTRAL BEAM INJECTOR AND COMPARISON WITH EXPERIMENT*

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R. G. Alsmiller, Jr. R. T. Santoro

[Abstract of ORNL/TM-7911, September 1981; also *J. of Fusion Energy* (submitted)]

Monte Carlo calculations have been performed to obtain estimates of the background gas pressure and molecular number density as a function of position in the PDX-prototype neutral beam injector which has undergone testing at the Oak Ridge National Laboratory. Estimates of these quantities together with the transient and steady-state energy deposition and molecular capture rates on the cryopanel of the cryocondensation pumps and the molecular escape rate from the injector were obtained utilizing a detailed geometric model of the neutral beam injector. The molecular flow calculations were performed using an existing Monte Carlo radiation transport code which was modified slightly to monitor the energy of the background gas molecules. The credibility of these calculations is demonstrated by the excellent agreement between the calculated and experimentally measured background gas pressure in front of the beamline calorimeter located in the downstream drift region of the injector. The usefulness of the calculational

method as a design tool is illustrated by a comparison of the integrated beamline molecular density over the drift region of the injector for three modes of cryopump operation.

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[†]Fusion Energy Division.

5.9. NEUTRON AND GAMMA-RAY STREAMING CALCULATIONS FOR THE ETF NEUTRAL BEAM INJECTORS*

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[Reprint of *Trans. Am. Nucl. Soc.* 38, 577-578, 1980; also ORNL/TM-7705 (February 1981); and *Nucl. Technol.* 2, 325 (1982)]

The tritium plasma of the Engineering Test Facility (ETF) fusion reactor will be heated and ignited by the injection of neutral deuterium. Since the deuterons must be injected through straight ducts into the plasma, the neutron and secondary gamma radiation produced as a result of the deuterium-tritium (D-T) reactions will stream directly into the neutral beam injectors and lead to adverse effects in vital components. The radiation leaking through the injection ports will be comprised of ~14-MeV neutrons (from the D-T reactions), plus a low-energy neutron and secondary gamma-ray distribution that results from the interactions of the energetic neutrons with the plasma liner and the primary shielding

about the torus. In this paper, two-dimensional radiation transport calculations carried out to estimate the effects on the injector components of radiation streaming through the injection duct are described, and the results of these calculations are presented and discussed.

The analysis was performed using the current ETF plasma dimensions, shielding configuration and composition, and operating scenario.¹ The neutral beam injectors were assumed to be the same as those proposed for use with the Tokamak Fusion Test Reactor (TFTR),² but appropriately modified to couple with the injection duct configuration of the ETF. The components of the injector, i.e., bending magnet, calorimeter, baffles, and crycondensation pumping panels, remained the same as those described in Ref. 2.

The radiation environment within the injector and duct shielding depends on the spatial, angular, and spectral distribution of the radiation entering the injector duct. This radiation is comprised of particles that enter the injector duct at the first wall and either stream directly into the injector or undergo collisions in the duct shield and then scatter into the injector. To approximate the radiation source, the plasma was divided into two regions. The first region was taken to be the volume of the plasma that is viewed from a point on the duct centerline at the injector entrance. Particles that originate in this volume can stream directly into the injector without undergoing collisions in the injector duct shield. The remaining plasma volume was taken to be the second source region. Particles originating in this volume undergo collisions in the primary torus shield or the injector duct shield before entering the injector.

The radiation originating in the viewed volume of the plasma was approximated using a point source located at the intersection of the plasma and injector duct centerlines. The radiation originating from the remainder of the plasma volume was approximated using a ring source located at a radial distance of 1.03 m from the intersection of the plasma and injector duct centerlines. For large duct length to duct diameter ratios (L/D), the point source gives rise to radiation levels at the injector entrance that

closely approximate the levels due to the viewed plasma volume, provided the magnitude and spectral shape of the point source are properly chosen. The validity of using a ring source depends on the proper choice of its location as well as on its magnitude and spectral shape. Locating the ring source at 1.03 m from the duct centerline represents a conservative location. Since most of the unviewed volume of the plasma is at radial distances >1.03 m, the selection of this location enhances the probability that particles originating in the unviewed plasma region will enter the injector.

The magnitude of the uncollided 14-MeV neutron component of the point source was obtained by considering the ratio of the viewed plasma volume to the total plasma volume. The magnitude of the uncollided 14-MeV neutron component of the ring source was obtained by conserving the total neutron wall loading of 2.4 MW/m^2 at the center of the entrance to the injector duct. To obtain the spectral shape of both the point and ring sources, a one-dimensional discrete ordinates calculation was performed using the ANISN code,³ with the 14-MeV D-T neutron source (i.e., the plasma region) and the primary shield surrounding the plasma modeled as infinite cylinders about the toroidal axis of the ETF. The details of obtaining these distributions are discussed.

The radiation transport calculations were performed using the DOT IV two-dimensional discrete ordinates code.⁴ These calculations were carried out using a relatively low order S_4 angular quadrature set. Since this quadrature set is not capable of treating radiation streaming, particularly through ducts with large L/D s, the DOT code was used in conjunction with the GRTUNCL first flight estimation code.⁵

In both the one- and two-dimensional procedures, the radiation transport was performed using a 35-neutron and 21-gamma-ray energy group transport library. These data were obtained by collapsing the 100-neutron, 21-gamma-ray energy group VITAMIN-C (ENDF/B-IV) data set.⁶ The angular dependence of the group-to-group transfer coefficients was treated using P_3 Legendre expansions. The

neutron and gamma-ray kerma factors were generated using the MACK⁷ and SMUG⁸ codes, respectively.

The calculated instantaneous nuclear heating rates as a function of distance along the cryocondensation pumping panel in the ETF neutral beam injector are shown in Fig. 5.9.1. Two pairs

Fig. 5.9.1. Energy Deposition Rate (MW/m^2) as a Function of Position Along the Neutral Beam Injector Cryopanel.

of curves are plotted. The upper pair of curves shows the total heating rate (due to neutrons plus gamma rays), and the lower pair of curves shows the heating rate due to neutrons only. For each pair of curves, the upper curve shows the response due to the combined contributions of radiation from the point and ring sources, and the lower curve is the heating rate due to the point source only. The heating rates are normalized to a neutron loading on the first wall of $2.4 \text{ MW}/\text{m}^2$, and are obtained assuming the cryopanel is composed of Type 316 stainless steel.

Calculated results of the instantaneous dose rate in the ion gun insulators have also been obtained and will be presented.

*Research sponsored by the U.S. DOE Office of Fusion Energy.

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5.10. ISO-RESPONSE CONTOURS OF THE ETF NEUTRAL BEAM INJECTORS*

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(Abstract of ORNL/TM-7783, May 1981)

Iso-response contours in the Engineering Test Facility (ETF) neutral beam injector have been calculated for the neutron flux, nuclear heating in SS-316 and G10 insulation, biological dose rate, and the $^{59}\text{Ni}(n,p)^{59}\text{Co}$ and $^{59}\text{Co}(n,\gamma)^{60}\text{Co}$ reactions. The spatial dependence of these responses in the injector, in the shielding that surrounds it, and in the injection duct is presented.

*Work sponsored by U.S. DOE Office of Fusion Energy.
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5.11. DOSE RATES FROM THE INDUCED ACTIVITY IN THE ETF NEUTRAL BEAM INJECTOR*

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R. A. Lillie R. G. Alsmiller, Jr.

(Abstract of ORNL/TM-7999, October 1981; also *J. of Fusion Energy* (submitted))

The dose equivalent rates outside the Engineering Test Facility neutral beam injector shield from the induced radioactivity have been

calculated for the reactor operating at 1140 MW for 1, 30, and 365 days. The dose rates at one day after shutdown are large even after one day of operation. Depending on the location and operating time, cooling times from 30 d to ~5 years are required before the dose rates are sufficiently low to allow routine maintenance work in the vicinity of the NBI shield.

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†UCC-ND Computer Sciences Division.

5.12. SPECTRAL TAILORING FOR FUSION RADIATION DAMAGE STUDIES: WHERE DO WE STAND?*

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[Abstract of J. Nucl. Phys. (in press); also Second Topical Meeting on Fusion Reactor Materials, Seattle, Washington, August 9-12, 1981]

The Oak Ridge Research Reactor is currently being used to irradiate an experiment (MFE-4A) to simulate in stainless steel-316 the He to DPA ratio of 14 appm He/DPA expected at the first wall of a fusion reactor. The experimental design criterion is to remain within $\pm 20\%$ of this ratio after an initial irradiation period of ~1.2 years. At present (March 31, 1981) the experiment has been in the reactor for 19 cycles corresponding to 131,543 Mwh, 4.05×10^{25} n/m² in thermal fluence, 1.21×10^{26} n/m² in total fluence, 3.1 DPAs and 21.0 appm He. The current He to DPA ratio is 6.8. This paper summarizes some of the analysis calculations for this experiment and presents the current status of this work. The analysis calculations utilize the codes VEN-TURE, ANISN, GAS, UNFOLD, etc. and the obtained results include gas production, DPA, and heating rates, and neutron spectral modifications resulting from different reactor fuel loadings and core piece compositions.

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†Engineering Technology Division.

‡UCC-ND Computer Sciences Division.

5.13. THE PRODUCTION RATE OF HELIUM DURING IRRADIATION OF NICKEL IN THERMAL SPECTRUM FISSION REACTORS*

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(Abstract of J. Nucl. Materials, in press)

For alloys with nickel as a major component, irradiation in mixed spectrum fission reactors produces a range of displacement damage and helium levels that can overlap a large part of the range calculated for fusion reactor service conditions. Irradiations in the High Flux Isotope Reactor (HFIR) have produced a data set adequate to define equations that give the total helium production. The data are for total neutron fluences of 0.09 to 3.09×10^{27} neutrons/m², yielding helium contents of 34 to 6108 at. ppm. The primary source of helium is the reaction sequence with thermal neutrons:

Examination of the available cross sections for these reactions and for other reactions that compete for ⁵⁸Ni and ⁵⁹Ni showed that the data are best fit when the 2200 m/s cross section for the second reaction is 11.60 ± 1.08 barns. The resulting helium production equation is:

$$\frac{N(\text{He})}{N(\text{Ni}-58)} = 0.12608 - 0.13272 \exp(-4.6\Phi_1) + 0.00664 \exp 5.9(-92\Phi_1),$$

where Φ_1 is the conventional 2200 m/s fluence of thermal neutrons, in units of neutrons per barn. This equation also gives a good fit to available experimental data from the literature.

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5.14. NEUTRONIC CALCULATIONS FOR THE CONCEPTUAL DESIGN OF AN IN-REACTOR SOLID BREEDER EXPERIMENT, TRIO-01*

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[Abstract of ORNL/TM-7758, March 1981; also
Nucl. Tech., Vol. 2, 73 (January 1982)]

Neutronics calculations have been performed to obtain tritium production and heat generation rates for the irradiation of solid tritium breeding materials in the Oak Ridge Research Reactor (ORR). Two breeder materials, Li_2O and LiAlO_2 , were considered. Burnup calculations were performed to estimate the amount of ^6Li present as a function of time.

*Research sponsored by U.S. DOE Office of Fusion Energy.

[†]UCC-ND Computer Sciences Division.

5.15. RADIATION STREAMING CALCULATIONS FOR INTOR-J*

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(Reprint of Trans. Am. Nucl. Soc. 38, 555, 1981)

The radiation streaming through the various penetrations in the blanket and shield of a tokamak fusion reactor will introduce significant problems in the operation and maintenance of the reactor and critical components. The radiation streaming through neutral-beam-injector (NBI) ports,¹⁻³ divertor throats,^{4,5} and diagnostic ports⁶ has been studied by several researchers. This paper describes the effects of radiation streaming through the divertor throat and NBI port of the INTOR-J reactor design that was proposed by the Japan Atomic Energy Research Institute (JAERI) to the International Atomic Energy Agency INTOR Workshop⁷ in 1979.

A horizontal view of the INTOR-J reactor is shown in Fig. 5.15.1. The 60-deg sector, which includes an NBI and two of the toroidal field coils (TFCs), was taken as the calculational model. A portion of the NBI that was modeled for use in three-dimensional Monte Carlo calculations to determine the radiation streaming effects is also shown in Fig. 5.15.1. The NBI and its

Fig. 5.15.1. Horizontal View of INTOR-J and Its Calculational Model.

components were represented as accurately as possible within the constraints of the general geometry package of the MORSE-GG code⁸ and the limitations of reasonable computation time.

Neutron and gamma-ray fluxes were calculated at several locations in the blanket and shield, superconducting TFCs, NBI, and in the reactor hall using the sequence shown in Fig. 5.15.2. The sequence consists of two phases: (a) the determination of the fluxes due to radiation streaming through the NBI port and (b) the calculation of the fluxes due to radiation leaking through the bulk blanket and shield. The radiation streaming through the injector port was calculated using a revised version of the MORSE-GG code that is capable of treating toroidal surfaces and that utilizes point detector estimator techniques compatible with reflective boundary conditions.⁹ This version of MORSE also

Fig. 5.15.2. Calculational Flow Chart.

includes subroutines for score point selection techniques that reduce the computational time required without a significant increase of the statistical error.¹⁰

The radiation streaming through the injection port was estimated by first calculating the uncollided neutron and scattered neutron and secondary gamma-ray distributions at the mouth of the NBI port. This was accomplished using the albedo Monte Carlo code of Yamauchi et al.¹¹ The spatial, angular, and energy dependence of the neutron and gamma-ray currents at the mouth of the port was stored on magnetic tape for subsequent use in the calculation of fluxes using Monte Carlo techniques and a three-dimensional representation of the NBI. The neutron and gamma-ray fluxes in the reactor hall due to radiations leaking through the blanket and shield were calculated using a code sequence consisting of DOT (to represent the reactor)¹² and DOMINO-MORSE (to represent the NBI).¹³

The fluxes and other relevant data at the various points of interest were obtained by summing the contributions from the calculational sequences. The GICX40 coupled 42-group neutron and 21-group gamma-ray cross-section set,¹⁴ based on ENDF/B-III and -IV for neutron data¹⁵ and the POPOP-4 library for gamma-ray production data,¹⁶ was used in the calculations. For the albedo Monte Carlo calculations, an 18-group neutron, 7-group gamma-ray set collapsed from the GICX40 set was employed.

A preliminary run for the flux calculations in the NBI with 2500 particles generated at the mouth of the port suggests sufficient statistical accuracy to be obtained at most of the NBI components within reasonable computation time.

*Work performed at Japan Atomic Energy Research Institute, Japan.

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High-Energy Particle Transport Calculations

5.16. HIGH- AND MEDIUM-ENERGY ACCELERATOR SHIELDING*

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(Abstract of Invited paper presented at the Mid-Year Symposium on Accelerator Health Physics of the Florida Health Physics Society, Orlando, Florida, February 1982)

When high-energy nucleons and mesons pass through matter a variety of nuclear reactions take place that produce additional nucleons and mesons, and a nucleon-meson cascade develops. Determining the shielding requirements of high- and medium-energy accelerators is, to a considerable extent, synonymous with determining the properties of such cascades for various incident particle types and energies, and for various geometries and materials.

A Monte Carlo transport code, HETC, for carrying out nucleon-meson transport calculations for shield design purposes has been available from the Oak Ridge National Laboratory for several years and this code will be discussed. The accuracy of results obtained with this code has been verified by comparisons with a variety of experimental data and some of these comparisons will be presented.

At low (<400-MeV) energies, few mesons are formed, and particle production by protons may, in many instances, be neglected, so shielding calculations at these energies involve only the transport of neutrons and photons. In this case, deterministic methods such as the method of discrete ordinates may be used. This is signifi-

cant because the method of discrete ordinates is usually more efficient than stochastic methods, and because with the method of discrete ordinates it is often possible to simplify calculations considerably by solving the adjoint rather than the "forward" transport equation. As an illustration of these concepts, importance functions for "neutron skyshine" at energies <400 MeV will be presented and discussed.

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5.17. EFFECTS OF FERMI MOTION AND SECONDARY PARTICLE COLLISIONS FOR PROTON DECAY IN NUCLEI*

T. A. Gabriel M. S. Goodman[†]

(Abstract of ORNL/TM-8068, February 1982; and Phys. Rev. Lett., in press)

The effects of Fermi motion and secondary particle collisions on the $\pi^0 e^+$ produced from proton decay in nuclei are investigated using a Monte Carlo program based on the intranuclear cascade model. The calculations indicate probabilities of 39%, 23%, and 15% for ^{16}O , ^{56}Fe , and ^{207}Pb , respectively for π^0 and e^+ to emerge simultaneously with angles $180^\circ \pm 13^\circ$. These probabilities are insensitive to total energy cuts provided $E_T \leftarrow 0.95 M_p$.

*Research sponsored by U.S. DOE Office of Basic Energy Sciences.

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5.18. A SENSITIVE SEARCH FOR NEUTRON-ANTINEUTRON MIXING*

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E. L. Hart[§]

(Abstract of paper presented at the Second Workshop on Grand Unification, Ann Arbor, Michigan, April 24-25, 1981)

We propose to perform a sensitive search for the phenomenon of baryon number violation ($\Delta B = 2$). We propose to search for neutron $\rightarrow$ antineutron transitions in a free neutron beam. The fact that matter appears to be stable sets a lower limit of about 10^5 seconds on the neutron-antineutron transition time. Recent theoretical predictions give mixing lifetimes on order of $10^{7 \pm 1}$ seconds. In a run of 120 days we will observe the transitions if the period is less than 10^8 seconds. The experiment will be performed at the Oak Ridge Research Reactor at the Oak Ridge National Laboratory. This site has the largest figure of merit (Nt^2) of all presently proposed sources of neutrons for this measurement. Neutrons will be transported 20 meters from the reactor core, under vacuum in a magnetic field free region. Antineutrons present in the beam as it exits the field free region will annihilate on a thin CD_2 target. These events will be identified by their characteristic signature of $2M_n c^2$ energy release in the form of emission of several pions.

*Research sponsored by U.S. DOE Division of High-Energy Physics.

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5.19. NEUTRINO PHYSICS AT THE SPALLATION NEUTRON SOURCE. II. NEUTRINO FLUXES, SHIELDING CONSIDERATIONS AND DETECTOR ANALYSIS*

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J. Wilczynski[‡] B. Zeitnitz[‡]

(Abstract of KFK 3174, June 1981)

The shielding and detector analysis associated with a contemplated low energy (~ 10 to 50

MeV) neutrino experiment at a spallation neutron source is presented and discussed. This analysis includes neutrino production and interaction rates, time dependence of the neutrino pulse, shielding considerations for neutrons coming directly from the spallation source and those which are scattered from other experimental areas, shielding considerations for galactic sources especially muons and finally detector responses to neutrino and background radiations. In general for a 1 mA (200 ns/pulse, 100 Hz), 1.1 GeV proton beam incident on a lead target surrounded by a moderator system, approximately 8 m of iron are required to reduce the background so that the event rate in the detector system is ≤ 1 event/day. If the detector system is not shadowed by the primary shield, then ~ 3 m of iron above the detector are required to reduce the background from scattered neutrons from experimental areas and galactic muons to an acceptable level of ≤ 1 event/day. For a 50 ton mineral oil segmented detector system and a volume distributed source, resolutions of approximately 10% (FWHM) can be obtained for gamma rays and electrons in the 10 to 20 MeV range.

*Work sponsored by Institut für Kern- und Teilchenphysik, Kernforschungszentrum Karlsruhe.

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5.20. A MONTE CARLO SIMULATION OF AN ACTUAL SEGMENTED CALORIMETER: A STUDY OF CALORIMETER PERFORMANCE AT HIGH ENERGIES*

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S. C. Wright[†] R. D. Kephart[§]

(Abstract of ORNL/TM-7123, January 1981; also Nucl. Instrum. Methods 195, 461 (1982))

The program CALOR simulated the performance of a real segmented Fe-liquid argon calorimeter at energies ranging from 10 GeV to 250 GeV. The energy resolution, angular resolution and energy deposition of these Monte Carlo results are in accord with the experimental results available at 10-38 GeV. $\sigma(E)/E = (33.4/\sqrt{E+5.07})\%$ and $\sigma_\theta(E) = 15.7 + 458/E$ mrad describe the resolutions.

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5.21. CALORIMETRY DESIGN FOR HIGH-ENERGY PARTICLE DETECTION*

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(Reprint of *Trans. Am. Nucl. Soc.* 41, 478, 1982)

INTRODUCTION

Calorimeters¹ are an important tool in high-energy experimental physics and are used primarily to determine the energy and type of incident particles: protons, neutrons, charged pions, leptons, gamma rays, etc. There are two basic types of calorimeters. The total absorption type (for example, NaI blocks, Pb glass blocks, etc.) in which the deposited energy is viewed in one form or another (Cerenkov light, scintillation light, electrical charge, etc.) throughout the system and the sandwich type in which only part of the energy is viewed (for example, a stack of Fe scintillation plastic sheets). One's ability to design a calorimeter to perform a certain task can have a strong influence upon the validity of the experimental results. The purpose of this paper is to describe a program package, CALOR,² that has been developed to assist experimentalists in designing and analyzing different calorimeter systems, thereby resulting in a savings of time, money, and effort. The CALOR package utilizes many of the current state-of-the-art Monte Carlo radiation transport codes, i.e., HETC,³ EGS,⁴ and MORSE.⁵ The validity of the results obtained with CALOR as well as with the individual codes has been verified many times by comparison with experimental data.⁶ The codes in the CALOR system are quite generalized and detailed enough so that any experimental calorimeter setup can be

studied. Due to this generalization, some software development is necessary because of the wide diversity of calorimeter designs. However, the calorimeter analysis presented illustrates many of the facets of the system.

CALOR: GENERAL DESCRIPTION

The three-dimensional, multimedia high-energy nucleon-meson transport code, HETC, is the heart of the CALOR system for incident hadrons and is used with slight modification to obtain a detailed description of the nucleon-meson cascade produced in energy absorbing systems. This Monte Carlo takes into account the slowing down of the charged particles via the continuous slowing-down approximation; the decay of charged pions and muons, inelastic nucleon-nucleus and charged-pion-nucleus (excluding hydrogen) collisions through the use of the intermediate energy intranuclear-cascade-evaporation (MECC)⁷ model ($E \lesssim 3$ GeV), and inelastic nucleon-hydrogen and charged-pion-hydrogen collisions via the isobar model ($E \lesssim 3$ GeV) and phenomenological fits to experimental data ($E \gtrsim 3$ GeV). Also accounted for are elastic neutron-nucleus collisions ($E < 100$ MeV); and inelastic nucleon- and charged-pion collisions with hydrogen. In past applications, high energy ($E \gtrsim 3$ GeV) inelastic charged-pion-nucleus and nucleon-nucleus collision data have been obtained by the use of a scaling model that takes data generated at ~ 3 GeV by MECC and scales it to all higher energies. Since then, the high energy intranuclear-cascade code (HECC)⁸ has become available and can now be used to generate this inelastic collision data. A complete incorporation of this model (HECC) into HETC will not be completed until a much later date. Until then, data generated by HECC for selected nuclides at energies of 5-, 10-, 20-, 40-, 80-, and 160-GeV for incident neutrons, protons, and charged pions have been placed in a data base for use by HETC. The scaling model is used to fill in all intermediate energies above ~ 3 GeV.

In most applications using HETC, neutrons are transported above energies of ~ 15 MeV and charged pions, muons, and protons and transported to ~ 1 MeV, with negative pions being

captured when they slow down to their cutoff energy. In most calorimeter applications, neutrons produced below their cutoff are assumed to deposit their energy at their point of origin or their energy is prorated on a gram-thickness basis between the active detector area and absorber area or totally ignored depending upon the application. In applications where the transport of low-energy neutrons is important (for example, in uranium-loaded calorimeters), the three-dimensional multi-group neutron and gamma-ray Monte Carlo transport code, MORSE is used.

The source distribution for the electromagnetic cascade calculation, i.e., photons from neutral pion decay, electrons, and positrons from muon decay, and deexcitation gamma rays from the decay of excited nuclei are provided by HETC. The transport of these particles is carried out by using a modified version of multimedia, 3-D, Monte Carlo code, EGS, developed by Ford and Nelson.⁴ This code takes into account all of the significant electron, positron-, and photon-interaction processes; i.e., Compton scattering, pair production, bremsstrahlung, photoelectric effect, annihilation, etc. Obviously, the EGS code is used exclusively for incident electrons, positrons, or gamma rays. The coupling of the codes mentioned above as well as for peripheral support codes is detailed in Fig. 5.21.1.

AN APPLICATION USING CALOR

The calculated results presented here are for a uranium-Fe-liquid argon calorimetry system. To analyze this calorimeter requires the full CALOR system and checks all facets of this system. The pulse height distribution obtained for 2- and 5-GeV protons incident are presented in Fig. 5.21.2. The results of including uranium in the calorimeter system are obvious. The resolution has been improved due to the fast fissioning of ²³⁸U. The experimental data obtained with this system are in excellent agreement with these calculated results. Additional data for this calorimeter will be presented as well as data for other calorimeter systems.

*Research supported by the Office of High Energy and Nuclear Physics.

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Fig. 5.21.1. Generalized Flow Chart of CALOR.

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2. T. A. Gabriel, J. D. Amburgey, and B. L. Bishop, "CALOR: A Monte Carlo Program Package for the Design and Analysis of Calorimeter Systems," ORNL/TM-5619, Oak Ridge National Laboratory Report, 1977.
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4. R. L. Ford and W. R. Nelson, "The EGS Code System: Computer Programs for the Monte Carlo Simulation of Electromagnetic Cascade Showers," SLAC-210 (June 1978).
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6. See Ref. 3-10 in Ref. 2.
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8. H. W. Bertini *et al.*, *Phys. Rev. C*17, 1382 (1978).

Fig. 5.21.2. Pulse Height and Sampling Distributions Expected from a U-Fe-A Calorimeter When 2- and 5-GeV Protons Are Incident.

5.22. CALCULATED DIFFERENTIAL π^+ PRODUCTION SPECTRA AT LARGE ANGLES FROM 28-GeV PROTONS ON A THICK IRIIDIUM TARGET AND COMPARISONS WITH EXPERIMENTAL DATA*

R. G. Alsmiller, Jr. T. A. Gabriel
B. L. Bishop†

[Abstract of ORNL/TM-7647, January 1981; also Nucl. Instrum. Methods 185, 253 (1981)]

Calculated results of the low-energy (<100 MeV) π^+ production spectra from 28 GeV protons incident on a thick iridium target are

presented and compared with experimental data. Experimental results are available at angles of 66° , 90° , and 133° with respect to the incident proton beam and comparisons with calculations are given at each of these angles. The calculated and experimental results are in moderately good agreement (factor of 2 to 4) over the energy range considered. Calculated results of the neutron, proton, π^+ and π^- energy spectra, averaged over the angular interval 45° to 75° , are also presented for energies ≤ 500 MeV.

*Research sponsored by U.S. DOE Office of High Energy and Nuclear Physics.

†UCC-NC Computer Sciences Division.

5.23. USE OF MONTE CARLO GENERATED PENCIL BEAM DATA IN ELECTRON DOSIMETRY*

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M. Goodman[§] A. Karpovsky[§] T. A. Gabriel

[Abstract of paper presented at the 23rd Annual Meeting of
the American Association of Physicists in Medicine, Boston,
MA, August 9-13, 1981]

EGS is a Monte Carlo program for simulation of electromagnetic cascade showers. It is presently being used to generate dose distributions of electron pencil beams for subsequent use in therapy dosimetry. EGS includes bremsstrahlung as well as secondary electrons in the computation of these distributions. Integration of these pencil beam profiles can produce broad beam distributions, and methods for including effects of inhomogeneities are being developed. Comparison of EGS output with data from Siemens Mevatron-12 and AECL Therac-20 accelerators has shown good agreement.

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5.24. A PHENOMENOLOGICAL MODEL FOR PARTICLE PRODUCTION FROM THE COLLISIONS OF NUCLEONS AND PIONS WITH FISSILE ELEMENTS AT MEDIUM ENERGIES*

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J. Barish[†]

[Abstract of ORNL/TM-7528, March 1981; also Nucl. Sci.
Eng. 79, 147 (1981)]

A fission channel has been added to the intranuclear-cascade-evaporation model of nuclear reactions so that this model can be used to obtain the differential particle production data that are needed to study the transport of medium-energy nucleons and pions through fissionable material. The earlier work of Hahn and Bertini on the incorporation of fission evaporation competition into the intranuclear-cascade-evaporation model has been retained and the statistical model of fis-

sion has been utilized to predict particle production from the fission process. Approximate empirically derived kinetic energies and deformation energies are used in the statistical model. The calculated number of emitted neutrons and residual nuclei distributions is in reasonable agreement with experimental data, but the number of emitted neutrons at the higher incident nucleon energies $\gtrsim 500$ MeV is sensitive to the level density parameter used.

*Research supported by U.S. DOE Office of Basic Energy Sciences.

[†]UCC-ND Computer Sciences Division.

5.25. NEUTRON PRODUCTION BY MEDIUM-ENERGY (≈ 1.5 GeV) PROTONS IN THICK URANIUM TARGETS*

R. G. Alsmiller, Jr. T. A. Gabriel
J. Barish[†] F. S. Alsmiller

[Abstract of ORNL/TM-7527, January 1981; also Nucl. Sci.
Eng. 79, 162 (1981)]

A model, that includes fission, for predicting particle production spectra from medium-energy nucleon and pion collisions with uranium nuclei has been incorporated into the nucleon-meson transport code HETC. A variety of calculated results obtained with this revised code for protons incident on uranium targets have been obtained and are compared with experimental data and with the calculations of other investigators. For incident proton energies $\gtrsim 1$ GeV the calculated results are in good agreement with several, but not all, of the available experiments.

*Research sponsored by U.S. DOE Office of Basic Energy Sciences.

[†]UCC-ND Computer Sciences Division.

5.26. NEUTRON PRODUCTION BY PROTONS (190-268 MeV) IN A WATER-COOLED TANTALUM TARGET*

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J. M. Barnes[†] R. T. Santoro

[Abstract of ORNL/TM-7949, September 1981; also Nucl.
Sci. Eng. 80, 452 (1982)]

Calculated results are presented for the number of neutrons produced, the neutron pro-

duction spectra, and energy deposition as a function of depth in the target when protons in the energy range 190 to 268 MeV are incident on a water-cooled tantalum target.

*Work Sponsored by U.S. DOE Office of Basic Energy Sciences.

†UCC-ND Computer Sciences Division.

5.27. NEUTRON-PHOTON MULTIGROUP CROSS SECTIONS FOR NEUTRON ENERGIES ≤ 400 MeV*

R. G. Alsmiller, Jr. J. Barish†

[Abstract of ORNL/TM-7818, June 1981; also Nucl. Sci. Eng. 80, 448 (1982)]

Multigroup cross sections (66 neutron groups and 21 gamma ray groups) are described for neu-

tron energies from thermal to 400 MeV. The elements considered are hydrogen, ^{10}B , ^{11}B , carbon, nitrogen, oxygen, sodium, magnesium, aluminum, silicon, sulfur, potassium, calcium, chromium, iron, nickel, tungsten and lead. These cross sections are available from the Radiation Shielding Information Center of the Oak Ridge National Laboratory.

*Work sponsored by U.S. DOE Office of High Energy and Nuclear Physics.

†UCC-ND Computer Sciences Division.

Integral Experiments Supporting Gas-Cooled Fast Breeder Reactor Designs

5.28. GCFR PLENUM SHIELD DESIGN - EXIT SHIELD EXPERIMENT*

F. J. Muckenthaler J. L. Hull† J. J. Manning

(Abstract of ORNL/TM-7715, May 1981)

This report describes the integral flux, energy spectra, and dose rate measurements made for the Exit Shield Experiment at the Oak Ridge National Laboratory Tower Shielding Facility as part of the Gas Cooled Fast Breeder Reactor Program. The source was the same mockup of fuel pins used in the previous Grid Plate Shield Experiment. Two mockups of the upper axial shield were studied: one with seven subassemblies prototypic of that portion of the Exit Shield without a control rod, and another that was representative of the shield region with a control rod.

The experiment was performed to provide verification of: the shield design methods, the shield effectiveness of a prototypic mockup, the analytic ability to calculate streaming effects in the presence of a control rod, and the source term

bias factors for the upper plenum. A series of measurements were made behind each configuration; for those configurations containing the control rod, measurements were made as a function of control rod position with and without B_4C in the control rod's subassembly.

*Work sponsored by U.S. DOE Office of Advanced Nuclear Systems and Projects.

†Operations Division.

5.29. FINAL ANALYSIS OF THE GCFR EXIT SHIELD INTEGRAL EXPERIMENT*

D. T. Ingersoll S. N. Cramer

(Abstract of ORNL/TM-7839, July 1981)

An integral experiment has been performed for verification of radiation transport methods used in the design of the exit shield for the proposed gas-cooled fast breeder reactor demonstration plant. The experiment was conducted at the ORNL Tower Shielding Facility and consisted of integral and spectral measurements of the neutron flux transmitted through prototypic mockups

of exit shield assemblies and a control rod assembly. The experiment was intended to confirm the predicted effectiveness of the proposed shield design, and to investigate neutron streaming characteristics in the vicinity of a control rod assembly.

The analysis was performed using discrete-ordinates and Monte Carlo transport methods and cross section data which are currently used for fast reactor shield design. The magnitude and spatial distribution of the neutron flux behind the central exit shield assembly and the control rod assembly were predicted well. The neutron flux behind the outer exit shield assemblies was significantly overpredicted due to modeling limitations. Significant changes in the neutron leakage from the control rod assembly and the adjacent exit shield assemblies were observed when the control rod mockup was moved from a fully inserted position to a fully withdrawn position. Finally, the presence of a control rod assembly shield was found to reduce the neutron leakage from the assembly by a factor of two.

*This work sponsored by U.S. DOE Office of Advanced Nuclear Systems and Projects.

5.30. GCFR PLENUM SHIELD DESIGN - SHIELD HETEROGENEITY EXPERIMENT*

F. J. Muckenthaler J. L. Hull[†] J. J. Manning

(Abstract of ORNL/TM-7714, May 1981)

This report presents measurements of neutron fluxes made in the Shield Heterogeneity Study at the Tower Shielding Facility (TSF) for the Gas Cooled Fast Reactor (GCFR) shielding program. The study consisted of mockups of selected neutron streaming paths in the outer radial shield surrounding the 300 MW(e) reactor. The experimental configurations included four basic segments: a straight planar slit through a graphite slab; a straight planar slit through an iron spacer between graphite walls; straight planar slits through the iron spacer and between the iron and graphite; and an offset slit through graphite.

A series of different slit widths were included in the study for each of the segments in order to

bracket anticipated conditions in the actual shield design. The TSR-II neutron flux was modified to simulate the spectrum incident on the GCFR radial shield by placing a spectrum modifier between the reactor and the graphite. Iron pieces were substituted for the stainless steel to model the envelopes containing the graphite. The neutrons streaming through the slits were mapped at several locations behind each mockup.

The integral experiment was performed to provide verification of methods used in calculating the neutron flux transmitted through the outer radial shield, thus reducing uncertainties in the calculated neutron flux incident on the prestressed concrete reactor vessel (PCRVR).

*This work sponsored by U.S. DOE Office of Reactor Research and Technology.

[†]Operations Division.

5.31. FINAL ANALYSIS OF THE GCFR RADIAL BLANKET AND SHIELD INTEGRAL EXPERIMENT*

D. T. Ingersoll L. R. Williams

(Abstract of ORNL-5756, April 1981)

An integral experiment has been performed for verification of radiation transport methods and nuclear data used in the design of the radial shield for the proposed gas-cooled fast breeder reactor demonstration plant. The experiment was conducted at the ORNL Tower Shielding Facility and consisted of integral and spectral measurements of the neutron and gamma-ray flux transmitted through slabs of materials which modeled a GCFR-type radial blanket and radial shield. Both UO_2 and ThO_2 blankets were investigated as well as several shield designs comprising stainless steel, graphite, and boronated graphite.

The analysis was performed using existing discrete-ordinates methods and cross section data which are currently used for fast reactor shield design. Neutron transmission through the blankets was predicted within 20-30% while gamma-ray heating between the blanket rows was predicted within 10%. ENDF/B-V thorium data appeared better than ENDF/B-IV for predicting

neutron transmission and gamma-ray heating. Neutron attenuation in the shield assemblies was generally predicted within 15% which corresponds to the estimated experimental and analytical uncertainties. Several other conclusions regarding the performance of laminated shields were obtained.

*Research sponsored by U.S. DOE Office of Reactor Research and Technology.

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

L. S. Abbott D. T. Ingersoll
F. J. Muckenthaler C. O. Slater

(Abstract of ORNL-5805, January 1982)

During the period between 1975 and 1980 a series of experiments were performed at the ORNL Tower Shielding Facility in support of the shield design for a 300-MW(e) Gas Cooled Fast Breeder Demonstration Plant. Although the GCFR program was cancelled in 1980, much of the information gained from the experiments and their analyses is generic and therefore applicable to other systems. In each case the approach was

to mock up a region of the GCFR reactor core and/or shield and to measure the intensities (and energies when possible) of reactor radiations penetrating through the region. In addition, calculations of the penetrating radiations were performed, using the same methods that were to be applied or had been applied to the GCFR system itself. When required, new calculational techniques were developed. Agreement between the calculated and measured results validated the method for application in the analysis of the GCFR shielding design. This report reviews the experiments and calculations, which included studies of: (1) neutron streaming in the helium coolant passageways in the GCFR core; (2) the effectiveness of the shield designed to protect the reactor grid plate from radiation damage; (3) the adequacy of the radial shield in protecting the PCRV (prestressed concrete reactor vessel) from radiation damage; (4) neutron streaming between abutting sections of the radial shield; and (5) the effectiveness of the exit shield in reducing the neutron fluxes in the upper plenum region of the reactor. In general, good agreement between the calculated and measured fluxes was obtained for all experiments.

*Research sponsored by U.S. DOE Office of Reactor Research and Technology.

Nuclear Analyser Supporting Gas-Cooled Reactor Designs

5.33. RADIAL SHIELDING ANALYSIS OF THE GAS-COOLED FAST REACTOR*

D. A. Reed¹ C. O. Slater H. L. Dodds²

(*Reprint of Trans. Am. Nucl. Soc. 38, 601, 1981*)

The purpose of this work is to evaluate the shielding effectiveness of the radial shield of the gas-cooled fast reactor (GCFR). This reactor,¹ currently being designed by General Atomic Company, poses a unique problem in shielding design. For safety purposes relating to loss of forced circulation of coolant accidents, the radial shield must allow flow-through natural circulation capability for the helium coolant. The coolant pathways through the shield must be designed so

as to prevent direct streaming of radiation from the plenum cavity to the reactor vessel liner. Important shielding criteria include a 30-yr neutron fluence limit for the vessel liner (10^{19} n/cm²) and a 30-yr dose limit for tendon grease in the concrete containment² (10^9 rad).

To determine the radiation exposure to the vessel liner and coolant duct material, a series of two-dimensional DOT-IV calculations was performed.³ Source descriptions for the plenum sides of the radial shields were obtained from upper and lower plena fluence calculations. Four radial shield geometries were chosen for analysis. Figure 5.33.1 shows a horizontal cut through the plenum, with an expanded view of one of the shield geometries. All geometries were similar in

that the coolant pathways were spaced at 7.5-deg intervals. Calculations for one 7.5-deg segment of each shield design were performed. Boundary conditions included a boundary source for the plenum side, a vacuum boundary for the liner side, and periodic boundaries on either side of the segment, to simulate all 48 parts of the shield. The transport calculation was done in X-Y geometry (i.e., the 7.5-deg curvilinear shield segment was

Based on comparisons of maximum calculated exposures and the shielding limits, it is concluded

Table 5.33.1. Comparison of Calculated 30-yr Exposures with the Maximum Allowed Exposures

Material	Maximum Calculated Exposure	Maximum Allowable Exposure	Quantity Compared
Shielding Graphite	$6.3 \times 10^{19} \text{ cm}^{-2}$	$1.4 \times 10^{22} \text{ cm}^{-2}$	total neutron fluence
Shielding SS 316B	$1.2 \times 10^{20} \text{ cm}^{-2}$	$4.0 \times 10^{22} \text{ cm}^{-2}$	total neutron fluence
Vessel Liner Iron	$3.2 \times 10^{18} \text{ cm}^{-2}$	$1.0 \times 10^{19} \text{ cm}^{-2}$	total neutron fluence
Tendon Grease (within PCRV)	$1.3 \times 10^8 \text{ rad}$	$1.0 \times 10^9 \text{ rad}$	total dose

that the two radial shield designs proposed for use in the GCFR adequately meet shielding requirements.

*Research sponsored by U.S. DOE Office of Reactor Research and Technology.

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3. W. A. Rhoades, D. B. Simpson, R. L. Childs, and W. W. Engle, Jr., "The DOT-IV Two-Dimensional Discrete Ordinates Transport Code with Space-Dependent Mesh and Quadrature," ORNL/TM-6529 (January 1979).
4. C. O. Slater, D. A. Reed, S. N. Cramer, M. B. Emmett, E. T. Tomlinson, "Analysis of the Conceptual Shielding Design for the Upflow Gas-Cooled Fast Breeder Reactor," ORNL-5669 (January 1981).
5. *Proc. 1980 Conf. Advances in Reactor Physics and Shielding*, Sun Valley, Idaho, September 14-19, 1980.

Primary emphasis was placed on calculating thermal neutron flux levels in the HTGR lower regions after giving due considerations to (1) neutron streaming through the large coolant holes in the bottom reflectors and support blocks and (2) thermal neutron streaming among boron pins within the graphite reflector blocks. The final aim was to determine the effect of this composite neutron streaming on the thermal neutron flux in the cover plates for the sidewall insulation. A design limit thermal neutron fluence of 10^{17} cm^{-2} was placed on the cover plates. Calculated results show this limit being exceeded by a factor of 10.5. However, large uncertainties in the calculations indicate a need for a definitive comparison of calculated and experimental results for a prototypic geometry. Other radiation levels were calculated to be well within the design limits.

*Research sponsored by U.S. DOE Office of Advanced Nuclear Systems and Projects.

5.34. SURVEY ANALYSIS OF RADIATION LEVELS IN THE HTGR LOWER REGIONS*

C. O. Slater S. N. Cramer

(Abstract of ORNL/TM-8141, February 1982)

A survey analysis of radiation levels within the General Atomic conceptual 900-MW(e) [$\sim 2000\text{-MW(t)}$] steam-cycle High-Temperature Gas-Cooled Reactor (HTGR) was performed.

5.35. STABILITY ANALYSIS OF THE HIGH TEMPERATURE THERMAL PEBBLE BED NUCLEAR REACTOR CONCEPT*

D. R. Vondy

(Abstract of ORNL/TM-7542, February 1981)

A study was made of the stability of the high temperature gas-cooled pebble bed core against

xenon-driven oscillation. This generic study indicated that a core as large as 3,000 MW_{th} could be stable. Several aspects present a challenge to analysis including the void space above the pebble bed, the effects of possible control rod configurations, and the temperature feedback contribution.

Special methods of analysis were developed in this effort. Of considerable utility was the scheme of including an azimuthal buckling loss term in the neutron balance equations admitting direct solution of the first azimuthal harmonic for

a core having azimuthal symmetry. This technique allows the linear stability analysis to be done solving two-dimensional (RZ) problems instead of three-dimensional problems. A scheme for removing the fundamental source contribution was also implemented to allow direct iteration toward the dominant harmonic solution, treating up to three dimensions with diffusion theory.

*Research sponsored by U.S. DOE Office of Nuclear Energy Programs.

Nuclear Analyses Supporting Utilization of Light-Water Reactors

5.36. THERMAL-HYDRAULIC ANALYSIS OF THE DUAL-FUNCTION GAMMA THERMOMETER*

J. O. Johnson[†] T. J. Burns

(Abstract of ORNL/TM-8089, December 1982; also ANS Eastern Regional Student Conference, Georgia Institute of Technology, Atlanta, Georgia, April 16-18, 1982)

A computational study was performed to further investigate the potential of a modified gamma thermometer as a monitor of both the localized power level and the adequacy of core cooling for a Pressurized Water Reactor.

The basic gamma thermometer has been proposed as an instrument for measuring the local heat generation rate within a reactor core. More specifically, the gamma thermometer can be viewed as a means for measuring the localized (e.g. within a particular assembly) power level. Thus, a set of GTs can be envisioned as providing the localized data that defines the overall power distribution within the core--which can be summed to yield the global power level of the reactor.

The volumetric heat source within the thermometer was updated to include heat deposited by neutrons as well as heat deposited by decay gammas and neutron-induced gammas. Utilizing this source, along with a more detailed thermal-hydraulics model, a series of thermal-hydraulic calculations were performed to simulate certain reactor transients of interest (i.e., reactor scram,

LOCA, etc.) in order to characterize the gamma thermometer response relative to power level monitoring and adequate core cooling indication.

The results of this study reaffirm the feasibility of utilizing the GT as a dual-function (power level and adequate core cooling capacity) measurement device, with each function accomplished virtually independent of the other. The study also indicates that there is a thermal-hydraulic regime for which the GT would no longer give easily interpretable signals. However, the pursuit of the GT as a viable nuclear instrument is still encouraged.

*Research sponsored by U.S. Department of Energy.

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5.37. APPLICATION OF STATIC AND TIME-DEPENDENT GENERALIZED PERTURBATION THEORY METHODS IN LWR PHYSICS ANALYSES*

J. R. White[†] M. L. Williams

(Summary of paper to be presented at ANS Topical Meeting on Advances in Reactor Physics and Core Thermal Hydraulics, Kiamesha Lake, NY, September 22-24, 1982)

Generalized perturbation theory (GPT) is a technique used to estimate changes in various reactor "response" parameters (e.g. relative power ratios, reactivity, etc.) arising from changes in the core configuration. Depletion perturbation theory (DPT) represents an extension of GPT which was developed to address time-dependent problems

encountered during reactor burnup calculations.¹⁻³ A computer code called DEPTH-CHARGE⁴ has been written to perform DPT calculations for multidimensional, multi-group burnup analyses and has been successfully applied in previous studies.⁵⁻⁷

Although there is a fair amount of applications experience with DPT, most of this knowledge has been directed toward the analysis of fast reactor systems.⁵⁻⁷ The primary purpose of the present study is to examine the capability of time-dependent perturbation theory as a tool for thermal reactor analysis. It is shown that DPT allows one to easily relate the beginning and

end-of-cycle states in a rigorous, quantitative manner. Such a capability can be useful in reload studies where many burnup calculations are performed to determine an acceptable loading pattern.

METHODOLOGY

The 2-D LWR model used in the sensitivity studies corresponds to the midplane of a 1000-MW PWR based on the B&W TMI-1 design.⁸ The beginning-of-cycle (BOC) model has three different enrichment assemblies and four different burnable poison loadings, arranged with 1/8-core symmetry as shown in Fig. 5.37.1.

W = Position Number

X = Assembly type

A = 2.06 w/o

B = 2.747 w/o

C = 2.747 w/o; 0.047 g B/in.

D = 2.747 w/o; 0.054 g B/in.

E = 2.747 w/o; 0.062 g B/in.

F = 3.05 w/o; 0.054 g P₁/in.

G = 3.05 w/o

Y = BOC power fraction

Z = EOC power fraction

R = Region Number

Region Power Fractions

Region	BOC	EOC
1	1.103	1.079
2	1.200	1.065
3	1.139	1.107
4	0.873	0.910
5	1.074	1.098
6	0.727	0.852

Fig. 5.37.1. Core Map Showing Assembly Locations and Zone/Region Power Distributions.

Representative four-group cross sections were generated with EPRI-CELL, and these data were assumed to be constant in all subsequent burnup calculations.

The 18-month cycle-1 operating history for this TMI-1 model was simulated in five burnup steps (75 full power days/step). A four-step soluble boron letdown curve was imposed and the nuclide transmutation during each step was calculated using assembly-averaged fluxes. This depletion calculation served as the reference case for the sensitivity study and provided the "unperturbed" BOC and EOC power distributions presented in Fig. 5.37.1.

For the purpose of this study, the power distribution was also characterized by the six coarse regions shown in Fig. 5.37.1. This was done to reduce the number of integral parameters used to represent the overall power profile. However, it should be noted that the choice of these particular regions was arbitrary and represents only one of many alternate methods of describing the power distribution. In fact, if more detail is desired, each assembly power could be treated as a response-region (at a somewhat increased cost since each response requires a separate adjoint calculation).

GPT RESULTS

Using data from the reference burnup calculation, a comprehensive set of adjoint computations were performed with the DEPTH-CHARGE/VENTURE⁹ system. The responses of interest included both BOC and EOC k_{eff} and region-averaged power ratios for the six regions defined in Fig. 5.37.1 (total of 14 cases, 7 static and 7 depletion runs). These responses should allow a detailed analysis and characterization of the BOC and EOC for this reactor model.

Also, it should be noted that once a reference calculation setup has been performed, the actual preparation of the adjoint computations is quite trivial and that each adjoint case has a comparable computer cost to the reference static and/or depletion problem. The reader is referred to Ref. 4 for details of the adjoint calculation procedure.

The numerical output of the adjoint solutions is a set of time-dependent derivatives that describe how the system response will react to a specified nuclide of cross-section perturbation. The nuclide derivatives,

$$N_z^*(t) = \frac{dR(t_f)}{dN_z(t)},$$

are nuclide-, space-, and time-dependent functions describing the rate of change of the response at time t_f with respect to the nuclide density vector N in zone z at time t . The components of the nuclide vector correspond to the atom densities of the various nuclides in the reactor, and the elements of the N^* vector are the respective nuclide "importances."

If we let $t_f = \text{BOC}$ and $t = \text{BOC}$, we have the familiar static derivative at BOC. However, if we set $t_f = \text{EOC}$ and $t = \text{BOC}$, then we have a time-dependent derivative that relates the response at the EOC to variations in the nuclide distribution at BOC (assembly shuffling, burnable or soluble boron adjustments, etc.). Thus, we have a quantitative measure that links the EOC state to the BOC state (or at anytime in between).

Tables 5.37.1 and 5.37.2 compare some GPT results with direct calculations for the BOC and EOC regional power responses. These data show results predicted by first-order perturbation theory versus exact results for three perturbations made to the reference model at BOC (see footnote at bottom of tables). There are at least three important observations to be made here.

First, we see that good agreement for the BOC responses is uniformly achieved only for Case 1. Although only a few assemblies were interchanged in Cases 2 and 3, these perturbations resulted in large variations in the power profile that were not well-predicted with first-order GPT. Thus, it appears that static first-order perturbation theory is not generally adequate for the analysis of PWR power distributions that result from "realistic" assembly interchanges, a fact which has been observed in other studies.¹⁰

In contrast, Table 5.37.2 indicates that changes in the EOC power distribution due to

Table 5.37.1. Comparison of GPT Results With Direct Calculation for the BOC Power Ratio Responses

n, Region	Technique	% $\Delta R_n/R_n$ ($R_n = PD_n/PD_{avg}$ at EOC)		
		Case 1	Case 2	Case 3
1	Direct	12.33	-40.80	49.14
	GPT	13.33	-32.00	29.14
2	Direct	8.58	-30.00	10.56
	GPT	9.47	-22.35	8.37
3	Direct	-1.32	-9.31	-6.23
	GPT	-1.42	-5.06	-4.00
4	Direct	-17.87	13.86	-4.47
	GPT	-20.13	13.46	-0.28
5	Direct	6.33	10.06	-11.55
	GPT	7.13	6.27	-8.92
6	Direct	3.44	56.12	-16.51
	GPT	3.96	35.73	-11.92

Note: Case 1 in zone 7, replace assy. B with assy. A.

Case 2 in zones 20 and 27, replace assy. F with assy. G.

Case 3 in zones 1, 2, 10, and 14, replace assy. E with assy. A.

Table 5.37.2. Comparison of DPT Results With Direct Calculation for the EOC Power Ratio Responses

n, Region	Technique	% $\Delta R_n/R_n$ ($R_n = PD_n/PD_{avg}$ at EOC)		
		Case 1	Case 2	Case 3
1	Direct	-2.41	9.55	18.26
	DPT	-2.59	9.63	-17.73
2	Direct	-1.22	6.57	-2.44
	DPT	-1.30	6.05	-2.91
3	Direct	0.81	0.63	3.07
	DPT	0.79	0.46	2.59
4	Direct	0.99	-2.09	-4.40
	DPT	1.15	-1.97	-3.45
5	Direct	-0.64	-3.46	6.28
	DPT	-0.73	-3.56	6.21
6	Direct	0.35	-7.75	6.10
	DPT	0.25	-6.32	6.09

Note: Case 1 in zone 7, replace assy. B with assy. A.

Case 2 in zones 20 and 27, replace assy. F with assy. G.

Case 3 in zones 1, 2, 10, and 14, replace assy. E with assy. A.

EOC perturbations are well-predicted with first-order time-dependent GPT! Thus, as indicated earlier, depletion perturbation theory does indeed provide an accurate quantitative means of relating parameters at the EOC to variations made at earlier times in the cycle, even for perturbations for which the static method breaks down.

The third and probably most important observation to be made from Tables 5.37.1 and 5.37.2 is that the magnitude of the deviation in the response due to a given BOC variation is considerably less at the EOC than at the BOC. This is due to the self-correcting nature of the depletion process. For example, highly nonuniform flux distributions tend to burn the peaked locations at a faster rate, thereby smoothing the overall distribution. In the present case, the reference power profile at EOC was quite uniform and even though large perturbations were introduced at the BOC for Cases 1-3, they had nearly died away by the EOC, again leaving fairly uniform distributions. Thus, the sensitivity of the EOC distribution to variations made at the BOC is relatively small and linear in nature. It is this linearity that permits the use of the time-dependent method in a wider range of applications relative to static GPT.

Although the above arguments are well-known and easily understood in a qualitative sense, the ability to quantify these sensitivities with depletion perturbation theory is a new and potentially useful capability.

SUMMARY

The present work has sought to demonstrate the validity and potential usefulness of depletion perturbation theory as a tool in the physics analysis of LWR systems. It has been shown that (1) time-dependent GPT methods can be utilized to relate the EOC state of a realistic PWR model to variations made at earlier times in the cycle and that (2) the first-order time-

dependent sensitivity function has a wider range of validity than that of the static case, due mainly to the smoothing nature of the depletion process. In addition, a considerable data base of nuclide and cross-section sensitivity information has been generated that characterizes both the BOC and EOC states for the PWR model utilized in this study.

The complete paper will discuss more fully the meaning and significance of these data and will also examine briefly several potential areas of application for time-dependent perturbation methods within the LWR industry.

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 †UCC-ND Computer Sciences Division.

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Integral Experiment Analyses Supporting Surveillance Dosimetry Improvement Program

5.38. COMPARISON OF CALCULATIONS WITH NEUTRON DOSIMETRY MEASUREMENTS PERFORMED AT THE OAK RIDGE POOLSIDE FACILITY*

R. E. Maerker M. L. Williams

(Reprint of Trans. Am. Nucl. 39, 812, 1981)

The Oak Ridge Poolside Facility (PSF), like the Pool Critical Assembly¹ (PCA), is used for benchmark dosimetry measurements that can serve to validate the transport methods used in calculating the high-energy neutron fluences (>0.1 MeV) in light water reactor pressure vessels required to estimate the neutron damage to the pressure vessels in the form of embrittlement. The PSF consists of an arrangement of two water gaps, 4- and 12-cm thick, separated by a simulated thermal shield and followed by a simulated pressure vessel wall and then a void box to represent a reactor cavity. The PSF is driven by the 30-MW Oak Ridge Reactor, whereas the geometrically similar core of the PCA has a maximum power of only 10 kW.

Previous papers^{2,3} have described comparisons of calculated and measured saturated activities in the PCA. These calculations were based on a fixed source provided for the PCA by Oak Ridge National Laboratory and the Studie Centrum Kernenergie/Centre d'Etude de l'Energie Nucleaire (MOL), in Mol, Belgium, which was obtained from both core criticality calculations and in-core fission rate measurements. The PCA reaction rates were limited to the $^{103}\text{Rh}(n,n')$, $^{115}\text{In}(n,n')$, $^{237}\text{Np}(n,f)$, $^{58}\text{Ni}(n,p)$, $^{238}\text{U}(n,f)$, and $^{27}\text{Al}(n,\alpha)$ reactions.

This paper reports the results of some calculated activities and compares them with published PSF measurements⁴ performed by Hanford Engineering Development Laboratory and other laboratories on the so-called Westinghouse surveillance capsule perturbation experiment. This experiment consisted of activation measurements performed within a pair of vertical 1-in.² surveillance capsules similar to those used by Westinghouse in its pressure vessel

surveillance program. One capsule was located immediately behind the thermal shield and the other immediately in front of the pressure vessel; they were positioned on opposite sides of the midplane centerline to eliminate any effect of one on the other. In addition, measurements were made by MOL inside very thin microtubes at different locations within the water gap separating the thermal shield and the pressure vessel. The purpose of this complete experiment was to provide an experimental determination of the perturbation on the reaction rates caused by the presence of the capsule. Our calculations not only will serve to verify these measured perturbations when they become available, but also will provide absolute comparisons with the measured reaction rates in the capsules.

In contrast to the PCA calculations, which utilized some in-core measurements, the PSF calculations were initiated by determining a fission source distribution based entirely on the results of a three-dimensional k-calculation of the core using diffusion theory.⁵ This distribution was obtained for near middle-of-cycle burnup and control rod conditions, which should adequately represent the time average of the entire 17-day cycle. The fission distribution was then integrated over the vertical (Z) direction and used as the fixed source in a discrete ordinates⁶ XY midplane calculation. To correct this midplane flux calculation for effects of finite core height and flux asymmetry due to control rod positioning, the VENTURE-calculated source distribution was also integrated over the width of the core (X) and used as the fixed source in another discrete ordinates YZ calculation. Three-dimensional fluxes throughout the water gaps and simulated thermal shield and pressure vessel wall were then synthesized according to the following prescription:

$$\phi(x,y,z) = \phi(x,y)\phi(y,z) / \int \phi(y,z)dz$$

Monte Carlo calculations have shown that the error introduced by the use of this equation is

small for those regions relatively near the mid-plane and the horizontal centerline of the reactor, a condition adequately satisfied by the measurement locations.⁷

The activities were calculated in units of becquerels per milligram of the natural element for the case of the $^{54}\text{Fe}(n,p)$, $^{63}\text{Cu}(n,\alpha)$, $^{46}\text{Ti}(n,p)$, and $^{58}\text{Ni}(n,p)$ reactions, and becquerels of fission product per milligram of fissioning isotope for the $^{237}\text{Np}(n,f)$ and $^{238}\text{U}(n,f)$ reactions using ^{140}Ba , ^{103}Ru , and ^{95}Zr as fission products. All calculations were corrected for an exposure time of 216 h and are compared with the measurements at each of the two positions in Table 5.38.1. The dosimetry cross sections, fission yields, and decay constants used in these calculations were obtained from ENDF/B-V.

From an inspection of Table 5.38.1, it can be concluded that the calculations agree with the measurements to within $\sim 15\%$, thus establishing the PSF for the first time as being accurately calculable starting from a core calculation and

extending the fluxes into the water by means of a transport calculation.

*Research sponsored by Nuclear Regulatory Commission Office of Light Water Pressure Vessel Irradiation Program.

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Table 5.38.1. Comparison of Calculated* and Measured* Activities

Reaction	Detector Next to Thermal Shield			Detector Next to Pressure Vessel		
	Calc. ^a	Meas. ^a	C/E	Calc. ^a	Meas. ^a	C/E
$^{54}\text{Fe}(n,p)$	6.02(3) ^b	6.59(3)	0.91	1.09(3)	1.215(3)	0.90
$^{63}\text{Cu}(n,\alpha)$	7.38(1)	7.83(1)	0.94	1.76(1)	1.85(1)	0.95
$^{46}\text{Ti}(n,p)$	4.63(3)	5.45(3)	0.85	1.00(3)	1.16(3)	0.86
$^{58}\text{Ni}(n,p)$	3.80(5)	4.31(5)	0.88	6.72(4)	7.72(4)	0.87
$^{237}\text{Np}(n,f)$ ^{140}Ba	9.20(5)	1.10(6)	0.84	1.08(5)	1.32(5)	0.82
$^{237}\text{Np}(n,f)$ ^{103}Ru	3.61(5)	4.20(5)	0.86	4.25(4)	5.00(4)	0.85
$^{237}\text{Np}(n,f)$ ^{95}Zr	2.26(5)	2.71(5)	0.83	2.67(4)	3.27(4)	0.82
$^{238}\text{U}(n,f)$ ^{140}Ba	1.47(5)	1.67(5)	0.88	2.20(4)	2.68(4)	0.82
$^{238}\text{U}(n,f)$ ^{103}Ru	5.96(4)	6.67(4)	0.89	8.92(3)	1.05(4)	0.85
$^{238}\text{U}(n,f)$ ^{95}Zr	3.04(4)	3.49(4)	0.87	4.57(3)	5.70(3)	0.80

*Averaged over all vertical locations except the +6.1-mm location in the thermal shield capsule and the +11.8-mm location in the pressure vessel capsule for the ^{238}U fission measurements only.

^aFor units, see text.

^bRead 6.02×10^3 , etc.

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5.39. SUPPLEMENTARY NEUTRON FLUX CALCULATIONS FOR THE ORNL POOL CRITICAL ASSEMBLY PRESSURE VESSEL FACILITY*

R. E. Maerker P. J. Maudlin†

[Abstract of ORNL/TM-7602, February 1981; also paper presented at the Fourth ASTM-EURATOM Symposium on Reactor Dosimetry at Gaithersburg, MD, March 22-26, 1982]

A three-dimensional Monte Carlo calculation was performed to estimate the neutron flux in the 8/7 configuration of the ORNL Pool Critical Assembly Pressure Vessel Facility. The calculational tool was the multigroup transport code MORSE operated in the adjoint mode. The MORSE flux results compared well with those using a previously adopted procedure for constructing a three-dimensional flux from one- and two-dimensional discrete ordinates calculations using the DOT-IV code. This study concluded that use of these discrete ordinates constructions in previous calculations is sufficiently accurate and does not account for the existing discrepancies between calculation and experiment.

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5.40. CALCULATIONS OF THE STARTUP EXPERIMENTS AT THE POOLSIDE FACILITY*

M. L. Williams R. E. Maerker

[Abstract of a paper presented at the Fourth ASTM-EURATOM Symposium on Reactor Dosimetry at Gaithersburg, MD, March 22-26, 1982]

Discrete ordinate calculations are made and the results compared with measurements per-

formed in the startup experiment at the Poolside Facility. Because of the physical size of the simulated surveillance capsule used in this experiment, the analytic procedure is more complicated than one adopted in earlier calculations of the PCA-PVF and PSF. The comparisons indicate the pressure vessel fluences in the long-term irradiation experiments still presently going on at the PSF, and which are geometrically identical to the startup experiment, can only be predicted to within about 20%.

*Research sponsored by U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research.

5.41. CALCULATIONS OF THE WESTINGHOUSE PERTURBATION EXPERIMENT AT THE POOLSIDE FACILITY*

R. E. Maerker M. L. Williams

[Abstract of a paper presented at the Fourth ASTM-EURATOM Symposium on Reactor Dosimetry at Gaithersburg, MD, March 22-26, 1982]

Discrete ordinates calculations are made and the results compared with measurements performed in the Poolside Facility for the purpose of validating various procedures adopted for the analysis of this facility. In addition, these calculations can be specifically used to verify the interpretation of measurements made to infer the perturbation effect of a Westinghouse surveillance capsule in a typical radiation environment. Comparisons indicate agreement on an absolute scale between measured and calculated reaction rates to within about 10% and agreement of the perturbation effect to within about 2%.

*Research sponsored by U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research, and U.S. Department of Energy.

5.42. CALCULATIONS OF TWO SERIES OF EXPERIMENTS PERFORMED AT THE POOLSIDE FACILITY USING THE OAK RIDGE RESEARCH REACTOR*

R. E. Maerker M. L. Williams

[Abstract of ORNL/TM-8326 (NUREG/CR-2690), May 1982]

This report contains two papers that were presented at the Fourth ASTM-EURATOM

Symposium on Reactor Dosimetry in Washington, D. C. on March 22-26, 1982 and serves as documentation of the analytical work performed by the Engineering Physics Division. These papers describe discrete ordinates calculations of two series of experiments that were performed at the Poolside Facility as part of the Surveillance Dosimetry Improvement Program and are very similar in scope.

*Work performed for U.S. Nuclear Regulatory Commission.

5.43. EVALUATION OF THE NEW ISNF ONE-DIMENSIONAL MODEL*

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(Reprint of *Trans. Am. Nucl. Soc.* 39, 887, 1981)

Last year, the National Bureau of Standards (NBS) reviewed¹ its one-dimensional model of the Intermediate Energy Standard Neutron Field² (ISNF). This standard field is being considered for a Cross Section Evaluation Working Group dosimetry benchmark problem.³ It is also used as a clean standard field in two Oak Ridge National Laboratory (ORNL) projects: in the development of ORACLE, the adjusted multigroup cross-section library,⁴ and in the development of an advanced methodology for light water reactor dosimetry applications.⁵

The measured values of fission rate ratios in the ISNF, as well as the corresponding calculated values, based on the old one-dimensional model,

and corresponding uncertainties have been reported earlier⁶ and are reproduced here in Table 5.43.1 for completeness. It can be seen that although all calculations, with the exception of the neptunium ratio, are not within one standard deviation of the corresponding experimental values, the discrepancy is not alarming. If one considers also the uncertainty in the calculated values, all measurements and calculations agree within about one joint standard deviation. We had thus no motivation to solicit a review of the ISNF model.

The NBS review of the ISNF one-dimensional model⁷ included additional auxiliary measurements of physical properties of the field's setup. The boron density was deduced from 2-keV neutron transmission measurements of the ISNF shell. The graphite density was determined from carbon samples in the block surrounding the cavity and in the reactor's thermal column. The moisture content of the graphite was also considered. This led to a change in the one-dimensional model.

The old and new one-dimensional models of the ISNF are given in Fig. 5.43.1. The main changes consist of

1. an increase of 2% in the ^{10}B density
2. a two-zone specification of the graphite reflector
3. addition of some hydrogen to the graphite to account for the moisture

Table 5.43.1. Measured and Calculated Central Reaction Rate Ratios and Their Relative Standard Deviations (S.D.)

Response	Measured		Calculated		Joint Relative S. D. %
	Value	Relative S. D. %	Value	Relative S. D. %	
28f/25f	0.0919	0.68	0.0885	4.28	4.33
	0.0926	1.39			4.50
37f/25f	0.510	2.03	0.502	10.30	10.50
49f/25f	1.155	1.28	1.136	2.64	2.93

Fig. 5.43.1. Physical Parameters for (a) Old and (b) New One-Dimensional Models of the ISNF.

4. a slight increase in the weight of the aluminum protective shell
5. a slightly different source radius.

A neutronics analysis of the ISNF using the new "best one-dimensional model" was performed at ORNL. The 174-group VITAMIN-E cross section library⁸ was used to calculate the 174-group central flux using XSDRNPM (Ref. 9) and the S_8 -linear/weighted numerical scheme.¹⁰ The calculations were performed once using the NBS ^{235}U fission spectrum¹¹ and once using the ENDF/B-V Watt fission spectrum.¹² Reaction rate ratios obtained using these calculations are given in Table 5.43.2. It is obvious that the Watt spectrum gives ratios closer to experimental values than the NBS fission spectrum, and that using the new one-dimensional model narrows the gap between the measured and calculated values.

In conclusion we can say that the revised one-dimensional model of the ISNF improves the agreement between measured and calculated values of central fission rate ratios.

*Research sponsored by Electric Power Research Institute.

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**Table 5.43.2. Central Reaction Rate Ratio Values in the ISNF
Using the New One-Dimensional Model**

Response	NBS - χ_{25}	ENDF-B/V - χ_{25}
28f/25f	.0871	.0910
37f/25f	.498	.513
49f/25f	1.136	1.143

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Miscellaneous Nuclear Analyses

5.44. MONTE CARLO ANALYSIS OF A TIME-DEPENDENT NEUTRON AND SECONDARY GAMMA RAY INTEGRAL EXPERIMENT ON A THICK CONCRETE AND STEEL SHIELD*

S. N. Cramer R. W. Roussin

(Abstract of ORNL/TM-7972, November 1981)

A Monte Carlo analysis of a time-dependent neutron and secondary gamma-ray integral experiment on a thick concrete and steel shield is presented. The energy range covered in the analysis is 15-2 MeV for neutron source energies. The multigroup MORSE code was used with the VITAMIN C 171-36 neutron-gamma-ray cross-section data set. Both neutron and gamma-ray count rates and unfolded energy spectra are presented and compared, with good general agreement, with experimental results.

*Research sponsored by Defense Nuclear Agency.

5.45. SENSITIVITY STUDY OF NEUTRON TRANSPORT THROUGH STANDARD AND REBAR CONCRETE*

S. I. Bkuiyan¹ R. W. Roussin J. L. Lucius²

(Reprint of *Trans. Am. Nucl. Soc.* 41, 493, 1982)

An investigation is under way at ORNL to (1) develop a data base pertinent to the transport of neutrons through thick concrete shields, (2) use the data in an energy group boundary selection and collapsing scheme, and (3) develop a simple methodology to access the data base to provide rapid solutions to practical shielding problems. This paper describes work carried out to fulfill objective (1), the work consisting of calculations of the transport of fission neutrons through 1- and 2-m-thick slabs of standard concrete¹ and rebar (steel-reinforced) concrete,² together with calculations of the sensitivities of the results to total, absorption, and elastic cross sections.

The transport calculations were performed with the one-dimensional discrete ordinates code ANISN³ in both forward and adjoint modes. The DLC-4IC/VITAMIN-C cross-section library⁴ (171 neutron, 36 gamma groups) was employed, with a P_3 cross-section expansion and an S_{16} angular quadrature. In all cases the fission source was assumed to be distributed within the first 1-cm thickness of the slab and the detector was assumed to occupy the last 1-cm thickness of the slab. For the rebar concrete the slab constituents were homogenized, with the horizontal and vertical No. 11 reinforcing steel rods comprising 7.6 vol.% of the slab. The quantity calculated was the absorbed dose rate, and care was taken in the mesh interval selection and source description to ensure agreement between the forward and adjoint results to within 0.02%.

The sensitivity calculations were performed with the SWANLAKE⁵ and FORSS⁶ codes. The JULIET module of FORSS based on the SWANLAKE algorithm for calculating sensitivity coefficients was used to aid in cross-section handling and plotting.

For all the cases considered, the sensitivity of the total dose rate at the exit face of the slab to the total, absorption and elastic scattering cross sections for the total mix of concrete constituents was determined. For the 1-m standard concrete, the sensitivity to the cross sections for the individual constituents was also determined, as well as the sensitivity to the cross sections corresponding to the water content of the slab. For the remaining cases, only the sensitivity to the cross sections for the total mix was calculated, except for the 1-m rebar slab, in which case the sensitivity to the cross sections for steel was also determined.

Some of the results of the sensitivity calculations are presented in Table 5.45.1. An examination of the 1-m standard concrete case shows that 90% of the total cross-section sensitivity of the dose rate is due to neutrons. Elastic scattering is the dominant partial cross section, accounting for 74% of the total dose rate sensitivity. Among the constituents, oxygen, silicon, and hydrogen contribute 38%, 24%, and 22% respectively to the total cross-section sensitivity, which means that these three constituents account for 85% of the total

sensitivity of the dose rate. The water in the concrete contributes 26% of the sensitivity. On the other hand, there are some elements that make insignificant contributions. This indicates that it would be possible to reduce the number of elements used to represent concrete in a calculation by considering their importance to the total response.

In the case of the 1-m rebar slab, the steel bars increase the magnitude of the sensitivity by 17%. The sensitivity for the 1-m rebar case is 56% of that for the 2-m rebar case. In the 2-m rebar case the gamma sensitivity increases to 42% of the total sensitivity compared to only 10% in the 1-m standard concrete case. The presence of the steel in the 2-m rebar case results in a 20% increase in the magnitude of the sensitivity. Comparing the two 2-m cases shows that elastic scattering contributes 47% and 44% of the total sensitivity for standard and rebar concrete respectively.

Other interesting results can be deduced from the sensitivity profiles, that is, plots of the energy dependence of the sensitivity coefficients for a given medium. Sensitivity profiles have a variety of uses. We are planning to use them to choose a new group structure for a few-group library. It is also possible to develop an automatic group-collapsing scheme on the basis of these coefficients.

All the sensitivities for the total cross section are negative, which means that as the cross section increases the calculated dose rate decreases. It is also observed that the sensitivities are large in the high energy region of the energy spectrum. Fine structure of the sensitivity spectrum caused by the resonances and antiresonance of the constituent elements is apparent.

From this study a data base for concrete shielding will be developed and will be followed by the development of an efficient methodology for predicting neutron responses.

*Research sponsored by U.S. Department of Energy.

⁴IAEA, Bangladesh Atomic Energy Commission.

⁵UCC-ND Computer Sciences Division.

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Table 5.45.1. Cross-Section Sensitivity of Total Dose Rate Transported Through Concrete Slabs (Fission Source)

Cross Section ^b	S. Sensitivity ^a		
	Gamma Groups	Neutron Groups	All Groups
1-m-thick Standard Concrete			
$\Sigma_t(\text{mix})$	-7.18487-01 ^c	-7.2481+00	-7.96654+00
$\Sigma_a(\text{mix})$	-2.91101-03	-7.3207-01	-7.34982-01
$\Sigma_s(\text{mix})$	0.0	-5.8875+00	-5.8875+00
$\Sigma_t(\text{H}^2\text{O})$	-3.73450-02	-2.0324+00	-2.06973+00
$\Sigma_a(\text{H}^2\text{O})$	-7.73712-02	-1.2218-01	-1.22195-01
$\Sigma_s(\text{H}^2\text{O})$	0.0	-1.9224+00	-1.9224+00
2-m-thick Standard Concrete			
$\Sigma_t(\text{mix})$	-5.61354+00	-8.3919+00	-1.40055+01
$\Sigma_a(\text{mix})$	-7.63756-03	-1.5209+00	-1.52850+00
$\Sigma_s(\text{mix})$	0.0	-6.65681+00	-6.65681+00
1-m-thick Rebar Concrete			
$\Sigma_t(\text{mix})$	-1.01620+00	-8.3633+00	-9.37947+00
$\Sigma_a(\text{mix})$	-4.29377-03	-6.6165-01	-6.65948-01
$\Sigma_s(\text{mix})$	0.0	-6.5426+00	-6.5426+00
$\Sigma_t(\text{steel})$	-2.26255-01	-1.0646+00	-1.29088+00
$\Sigma_a(\text{steel})$	-3.06586-03	-2.4869-01	-2.51753-01
$\Sigma_s(\text{steel})$	0.0	-4.4376-01	-4.4376-01
2-m-thick Rebar Concrete			
$\Sigma_t(\text{mix})$	-7.15619+00	-9.7171+00	-1.68732+01
$\Sigma_a(\text{mix})$	-1.11216-02	-1.27830+00	-1.28946+00
$\Sigma_s(\text{mix})$	0.0	-7.52635+00	-7.52635+00

^a $S_{\Sigma}(r) = (dR/R)/(d\Sigma/\Sigma)$, where r is a point at phase space, R is the total response (dose rate), and Σ is the macroscopic cross section.

^b Σ_t , Σ_a , Σ_s are the total, absorption, and elastic scattering macroscopic cross sections.

^cRead: -7.18487×10^{-1} .

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5.46. CROSS-SECTION ADJUSTMENT WITH MONTE CARLO SENSITIVITIES: APPLICATION TO THE WINFRITH IRON BENCHMARK*

M. C. G. Hall[†]

(Abstract of Nucl. Sci. Eng., in press)

A major obstacle in obtaining adjusted cross sections from integral experiments is the expensive and time-consuming evaluation of sensitivities and modeling corrections. The principal contribution of this paper is the development of a state-

of-the-art Monte Carlo method that evaluates sensitivities particularly efficiently, and that uses "point" nuclear data and 3-D combinatorial geometry to eliminate modeling errors. This method enables adjustment procedures to be applied more reliably and generally than previously possible. Theoretical advances include the way the sensitivity estimator is chosen and evaluated. Also the adjustment procedure takes into account all the Monte Carlo statistical errors, and iteration is used to cope with non-linearities. The methods developed are successfully applied to an analysis of the Winfrith Iron Benchmark Experiment.

*Work performed under the auspices of the UK Atomic Energy Authority.

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5.47. ADJOINT GAMMA RAY ESTIMATION TO THE SURFACE OF A CYLINDER - ANALYSIS OF A REMOTE REPROCESSING FACILITY*

S. N. Cramer

[Abstract of ORNL/TM-7686, July 1981; also Nucl. Sci. Eng. 79, 417 (1981)]

The next-event estimator in the MORSE multigroup Monte Carlo code has been extended to include Klein-Nishina scattering and annihilation radiation from pair production for both the forward and adjoint modes of calculation. A formulation for the solid angle subtended at a point by a cylinder has also been used in the estimator. These procedures have been included in the investigation of the gamma ray environment in the design of a remote fuel reprocessing facility. Calculational results are presented which indicate the validity and efficiency of the developed methods as compared to those in standard use.

*Research sponsored by U.S. DOE Office of Nuclear Fuel Cycle.

5.48. SHIELDING CONSIDERATIONS FOR SPACE POWER REACTORS*

D. E. Bartine W. W. Eagle, Jr.

(Abstract of a paper and viewgraphs presented at a Special Conference on Prime Power for High-Energy Space Systems, Norfolk, VA, February 22-24, 1982)

Shielding design considerations are important for space power reactor systems. Shielding is required to meet radiation level constraints for payload and reactor electronic components and perhaps for other materials. Material selection and shield weight and size are important design considerations, requiring careful analysis and verification, but shield design is complicated by radiation streaming paths caused by shield penetrations for coolant and control systems. Resources are available to meet these requirements as a result of previous shielding programs, most notably the NASA SNAP program, the DOD weapons effects program, and the DOE LMFBR program. Available analytic techniques include 1-D and 2-D discrete ordinates and 3-D Monte Carlo for radiation transport, and associated size and shape optimization capability. Facilities available for experimental measurements at Oak Ridge National Laboratory include the Tower Shielding Facility (TSF) reactor and a modified SNAP-2 reactor. The TSF reactor is a 1-MW(th) research facility used to verify cross-section data and calculational techniques for dose attenuation, which could also be used to verify radiation streaming through penetrations, and shield weight and shape optimization. The modified SNAP reactor is a 10-KW(th) system with high-enriched ZrH fuel, Be reflector, NaK coolant, and a LiH shield. The core and reflector assembly are SNAP-10A design, with a SNAP-2 shield which could be replaced with other experimental shields to verify prototypic shield designs.

In summary, a shielding technology program is necessary to support space power reactor development. Payload and reactor electronic

components must be protected; and shield composition, weight, and shape are important design considerations. Techniques are available for design analysis and optimization, and experimental facilities are available for design verification. Shielding analysis should proceed concurrently with reactor design, and experimental verification should follow the analysis closely.

*Research sponsored by U.S. Department of Energy.

5.49. AIR-OVER-GROUND CALCULATIONS OF THE NEUTRON, PROMPT, AND SECONDARY GAMMA FREE-IN-AIR TISSUE KERMA FROM THE HIROSHIMA AND NAGASAKI DEVICES*

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

(Reprint of *Trans. Am. Nucl. Soc.* 41, 471 (1982))

This paper reports preliminary results of the two-dimensional discrete-ordinates calculations for the air-over-ground transport of radiation from the Hiroshima and Nagasaki weapon devices. It was found that the gamma-ray kerma dominated the total kerma for both environments.

Considerable effort in the area of radiation shielding during the last two decades has been directed toward developing adequate transport methods and cross section data to accurately describe the numerous neutron and gamma-ray interactions within various materials. Once these methods and cross sections have been benchmarked against integral data, they can be used with confidence in calculating solutions to problems which have little or no experimental data such as the Hiroshima and Nagasaki bursts. For problems of this type that can be described in two-space dimensions, are not time-dependent, and require the particle fluence throughout the entire system, the discrete ordinates method may be used effectively. The results described herein were produced mainly with the two-dimensional, discrete ordinates code DOT (Ref. 1), run in R-Z geometry with p_3 scattering.

The neutron and gamma-ray leakage spectra from the Hiroshima and Nagasaki devices were obtained from Preeg.² This data was placed into the energy group structure of the Defense Nuclear Agency (DNA) Library,³ which contains the standard cross-section data and response factors used in this study. The atmospheric data for determining the air atomic number densities was obtained from Malik.⁴ Since the original devices were exploded at such high altitudes (570-580 m for Hiroshima and 503 m for Nagasaki) and preliminary calculations indicated that the ground contributed only fifteen percent of the prompt and secondary gamma-ray kerma, a simplified four element ground⁵ was used. An initial calculational effort in 1976 gave neutron kerma results for the Hiroshima device which were higher than those given later by Loewe and Mendelsohn.⁶ An investigation of this disagreement revealed that the 1/E weighted DNA cross sections did not represent properly the scattering phenomenon associated with the Hiroshima neutron leakage spectrum. Therefore, new cross-section sets for both the Hiroshima and Nagasaki air environments were produced by collapsing the VITAMIN-C 171 neutron/36 gamma-ray group cross sections⁷ into the DNA group structure using the ANISN one-dimensional transport code.⁸

An anomaly known as "ray effects" may occur in two-dimensional discrete-ordinates geometry if the source and detector are small, the scattering mean free path is large compared to the space mesh, and portions of the space mesh are not intersected by at least one of the discrete polar directions. If ray effects are dominant, then the particle fluences in the mesh intervals intersected by the discrete polar directions tend to be too large, and in mesh intervals not intersected, the fluences tend to be too small. This effect can be mitigated by 1) use of the first-collision source in each space mesh, or by 2) using a higher order angular quadrature (more polar angles). For most cases, 1) will insure accurate results. However, if the localized source which characterizes the device leakage in turn produces another localized source

(e.g., secondary gamma-ray production), then a combination of 1) and 2) must be used to obtain the proper particle fluences. For the neutron calculations, technique 1) was sufficient. But since hydrogen was present in the air, a large secondary gamma-ray source was produced near the device and technique 2) was used to mitigate these ray effects.

The total collided particle fluence was calculated with the DOT code using the first collision source as input. Any calculations which required a high-order (e.g. S_{20}) quadrature set for ray smoothing, were run with a low-order (S_8) set until converged, then restarted with the high-order set until convergence was once again obtained (usually no more than two iterations). The final scalar fluence was then available to convolute with any desired response factors.

Calculational results indicate that because of differences in the leakage spectra of the two devices and the moisture in the air, the neutron free-in-air tissue kerma relaxation lengths changed at different rates for the two bursts. The relaxation length for the Nagasaki device reached a constant value for a ground range much closer to ground zero than did the Hiroshima device. The relaxation lengths were ultimately determined by neutrons of energies >2 MeV. This explained the basic difference between the high Hiroshima neutron T65D kerma values and the lower calculational values.⁹ The T65D data assumed the neutron relaxation lengths to be equal for both the Hiroshima and Nagasaki bursts. The gamma-ray kerma for the T65D data and the recent calculations were in better agreement than were the neutron kerma. The gamma kerma tend to dominate the total kerma in the recent calculations for both Hiroshima and Nagasaki.

*Research sponsored by Defense Nuclear Agency.

[†]ICC-ND Computer Sciences Division.

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5.50. BACKGROUND OF DOSE ESTIMATES FOR HIROSHIMA AND NAGASAKI*

D. E. Bartine

(Reprint of *Trans. Am. Nucl. Soc.* 41, 469, 1982)

A comprehensive dosimetry study of Hiroshima and Nagasaki was directed by John Auxier¹ and designated as T65D (Tentative 1965 Doses.) The radiation dose estimates given in this study were obtained by combining tissue kerma vs distance (ground range) data developed by Auxier² and house-shielding factors developed by Cheka.³ The dose vs distance data and energy yield for the Nagasaki (Fat Man) device were obtained primarily from weapons tests of similar devices. No directly comparable weapons tests were performed for the Hiroshima (Little Boy) device, so energy yield and dose vs distance data were obtained primarily from weapons data for

the closest devices tested and from related cold configuration and reactor experiment measurements. These neutron leakage data sources included the Ichiban Critical Assembly, a 1-D experimental mockup of the Little Boy device, measurements taken using a duplicate device with a Californium source. The Oak Ridge National Laboratory Health Physics Research Reactor was used as a source for neutron and gamma measurements in an air-over-ground geometry. The tissue dose vs distance relationship developed by Auxier assumed a single constant neutron tissue kerma relaxation length for both Hiroshima and Nagasaki and constant but different gamma kerma relaxation lengths. The T65D results appeared to be corroborated by the studies of Hashizume⁴ and Ichikawa⁵ who estimated gamma-ray fluence from thermoluminescence of tiles and neutron fluence from ⁶⁰Co activity in reinforcing rods. The T65D results indicated that the dose to survivors at Nagasaki was almost completely due to gamma radiation [~ 10 Gy (γ) vs ~ 0.5 Gy (n) at 1000 m ground range], while the Hiroshima doses were more balanced, [~ 3 Gy (γ) and ~ 2 Gy (n) at 1000 m].

In 1976 Preeg,⁶ then at Los Alamos, performed a neutron leakage calculation, including both radiation transport and hydrodynamic effects, for an essentially 1-D mockup of the Hiroshima (Little Boy) device. Preeg reported that 1-D infinite air calculations utilizing this source spectra gave considerably lower neutron doses than those reported in T65D. This indication was confirmed by two-dimensional air-over-ground calculations performed by Pace⁷ at Oak Ridge in 1976 and by Kaul⁸ of SAI using the ATR-4 code which simulates air-over-ground transport.

In 1980, Loewe and Mendelsohn⁹ investigated the neutron and gamma doses for Hiroshima and Nagasaki using the Preeg source and including an estimate of the delayed gamma radiation due to fission products. Their use of a cross section set especially collapsed to represent the low-energy neutron source at Hiroshima resulted in neutron doses even lower than those of Pace. Later calculations by Pace using a revised cross section set verified these results. Both studies also indicated an increased gamma dose for Hiroshima and the

lower gamma dose for Nagasaki compared to the T65D values.

Also in 1980, Kerr¹⁰ initiated a review of the dosimetry for the atomic-bomb survivors for the National Council on Radiation Protection. As part of this review, experimental data, including sulfur activation by high-energy neutrons at Hiroshima and Nagasaki, was compared with the recent calculations of Loewe and Pace. The investigation indicated the need for a more thorough evaluation based on a revised neutron leakage (angular and energy spectrum) for Hiroshima, especially. This session will present some current results for a Defense Nuclear Agency-directed study to examine radiation transport aspects, primarily for the Hiroshima device. It is important to note that this session represents only part of a larger continuing effort to upgrade the dosimetry and the effects correlation for Hiroshima and Nagasaki.

*Research sponsored by Defense Nuclear Agency.

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Section 6

ENERGY ECONOMICS MODELING AND ANALYSIS

6.0. INTRODUCTION

In recent years the division has entered into the area of energy economics modeling and analysis. The initial, relatively short-term project was directed towards the evaluation of the validity of the large energy modeling system (LEAP) that was used by the DOE Energy Information Administration to predict the energy portion of the U.S. economy for the period 1995-2020 in its 1978 annual report to congress. The LEAP evaluation was completed during 1981, with more recent efforts for EIA of this type being devoted to long-term projections for the 1981 annual report to congress and to an evaluation of the National Utilities Financial Statement (NUFS) Model. In an related effort, also for EIA, a brief survey of nuclear fuel-cycle models was performed.

Under the sponsorship of the DOE Division of Policy and Strategic Planning (Office of Fossil Energy), another program was initiated in late 1980 which has as its aim the development of strategic planning capabilities for U.S. liquid and gaseous fuel supplies. As part of this program, two computer models are being developed, a Liquid Fuels Supply (LFS) model and a Gaseous Fuels Supply (GFS) model, both of which will be incorporated in a World Oil Model that includes a detailed representation of the OPEC cartel. This World Oil Model, sponsored by the DOE Division of Energy Security, is being developed by the Engineering Physics Division, together with its sub-contractors, Decision Focus, Inc., and Lewin Associates.

Evaluation of LEAP: Long-Term Energy Analysis Program. -- Several reports have resulted from an integrated set of investigations carried out by the Engineering Physics Division, together with the ORNL Energy Division, to assess the validity of EIA's energy forecasting model LEAP as it was used for the 1978 EIA Administrator's Report to Congress. The evaluation results were documented during the calendar year 1981. These included the final evaluation report (Paper 6.1), as well as individual technical reports on (1) the development and application of an adjoint sensitivity theory to the model, (2) the mathematical structure of the model, (3) its major underlying assumptions, and (4) its input data. Also, the team's operating experience with the computer code was reported, and "predictions" generated with the model were compared with "real" results experienced during the latter half of the 1970s. Using this project as a focus, a general approach to the evaluation of models of the energy economy was developed, documented, and employed in guiding the study. Finally, an analysis of "screening" methodology in the context of this LEAP evaluation showed that it is possible to focus attention on the most important of thousands of parameters using a smaller number of model runs only when sufficient prior information is available about the behavior of model outputs under input parameter changes.

Altogether, the LEAP-78 evaluation effort was one of the most thorough evaluations of an energy model ever performed, and the documented results should be of great interest to other evaluators and to users and developers of similar models.

Evaluation of NUFS. -- During the reporting period an evaluation of the National Utilities Financial Statement (NUFS) Model was conducted for EIA. NUFS is a financial regulatory

model of the U.S. electrical utilities industry. It is used primarily to assess the financial feasibility of projected capacity expansion programs of either publicly owned or privately owned utilities. The evaluation of NUFS was conducted using the comprehensive evaluation methodology that had been developed at ORNL.

Survey of Nuclear Fuel-Cycle Codes. -- At the request of EIA, the division also undertook a brief survey of available nuclear fuel-cycle models and identified those codes that are potentially useful in fulfilling the task of the Nuclear Energy Analysis Division. Subsequently the EUREKA uranium mining and milling code was evaluated in detail and the results forwarded to EIA.

Development of Strategic Planning Capabilities for Liquid and Gaseous Fuel Supplies. -- The objective of this new program sponsored by DOE/FE is to enable quantitative assessment of competitive supply technologies. This should contribute to the development of the knowledge base required for reducing U.S. energy supply vulnerability to the economic and security implications of reliance upon scarce and increasingly expensive conventional petroleum resources. As noted above, this program includes the development of the computerized LFS and GFS models. It also includes the development of an automated capability for sensitivity and uncertainty analysis, as well as a decision-making methodology based on sensitivity theory and planning capabilities for contingencies in the event of supply interruptions.

The Liquid Fuels Supply (LFS) model has been developed in detail and has been benchmarked. A prototype Gaseous Fuels Supply (GFS) model is currently operational and is being used as a test bed for the development of an expanded GFS model.

Development of a World Oil Model. -- Since the OPEC cartel is an important factor in setting world hydrocarbon prices that strongly influence U.S. technological and policy decisions in energy economics, the forces governing the OPEC decision-making process must be understood. Key components of this understanding must include a behavioral model of OPEC and a model of the world energy market in which it operates. The World Oil Model being developed for DOE/ES is an attempt to respond to these objectives.

As designed, the World Oil Model distinguishes between five demand regions (the U.S., Japan/Asia/Oceania, other OECD countries, lesser developed countries, and centrally planned economies) and eight regions competing to provide the supplies (U.S., Canada, Asia/Oceania, Europe, lesser developed countries, centrally planned economies, OPEC, and a backstop source). The U.S. is represented by integrating the U.S. LFS model (see above) into the World Oil Model.

Design of the World Oil Model has been completed by Decision Focus, Inc., and its testing will soon be started at ORNL.

**6.1. AN EVALUATION OF THE
LONG-TERM ENERGY ANALYSIS
PROGRAM USED FOR THE 1978
EIA ADMINISTRATOR'S REPORT
TO CONGRESS***

R. W. Peelle C. R. Weisbin

R. G. Alsmiller, Jr.

[Abstract of ORNL-5741, October 1981, issued by Energy Information Administration, U.S. DOE as DOE/NBB-0009, October 1981; also abstract of Energy 7(2), 155 (1982)]†

This report presents an evaluation of the Long-Term Energy Analyses Program (LEAP), a computer model of the energy portion of the U.S. economy that was used by the Energy Information Administration (EIA) of the U.S. Department of Energy for the 1995-2020 projections in its 1978 Annual Report to Congress. An overview of the 1978 version, LEAP Model 22C, is followed by an analysis of the important results needed by its users. The model is then evaluated on the basis of (1) the adequacy of its documentation, (2) the local experience in operating the model, (3) the adequacy of the numerical techniques used, (4) the soundness of the economic and technical foundations of the model equations, and (5) the degree to which the computer program has been verified. To show which parameters strongly influence the results and to approach the question of whether the model can project important results with sufficient accuracy to support qualitative conclusions, the numerical sensitivities of some important results to model input parameters are described. The input data are categorized and discussed, and uncertainties are given for some parameters as examples. From this background and from the relation of LEAP to other available approaches for long-term energy modeling, an overall evaluation is given of the model's suitability for use by the EIA.

Following publication of reports from the present evaluation efforts, most aspects of the documentation of the LEAP Model 22C meet DOE standards; documentation of the model parameters does not. Members of the evaluation team were able to learn to operate and make minor modifications to the model resident on the EIA computer system, but the model was not readily transportable to other computer sites.

Published LEAP results were replicated. The numerical techniques used in the LEAP system were adequate for obtaining convergence of Model 22C, but they do not have a firm mathematical basis.

The LEAP modeling system is flexible and potentially useful for studying alternative energy supply patterns under (nearly) exogenously determined final demand. The usefulness of this system is conditional upon (1) the validity of future cost and economic behavior data estimated now; (2) the broad realism of the equations used to simulate market interaction and investment; and (3) the ability to represent in the model the alternative conditions and policies important to an energy agency. Even if questionable input parameters are accepted, the LEAP system for ARC-78 did not fully meet conditions (2) or (3); for example, it did not permit the study of retrofit of capital plant for the purpose of achieving improved efficiency, and not all regulatory policy options could readily be handled in the system's framework. The model equations were obtained by modifying a partial equilibrium analysis on the basis of engineering judgment. They were not derived from any firm economic theory describing plant construction and operation decisions with competition among alternative suppliers in the complex energy network. In the specific LEAP model used for the 1978 Administrator's report, long-term regional issues other than those related to coal development could not have been addressed.

The summary of sensitivity studies, performed on the LEAP model for ARC-78 through a novel application of adjoint sensitivity theory, shows relatively high sensitivities of total energy cost to parameters used in many activities and to others that affect demand and the economic efficiency of important activities. Observed nonlinear effects were sufficiently severe in model reruns with altered parameters that second-order sensitivity coefficients needed to be considered. The input data appear representative of data previously used in similar antecedent models, but they were not well enough documented to allow assessment of their appropriateness by the public.

Proper input to the model at EIA was apparently inhibited in a few cases by the imprecise data definitions available to the agency.

Long-term energy analysis in some form is required by the statutes establishing the Department of Energy. The LEAP system should be appropriate to help meet a portion of these analysis needs, particularly those involving alternative energy supply options for the early part of the next century. The LEAP modeling effort for the 1978 Administrator's report was a reasonable first attempt to obtain useful projections from the LEAP system. A number of recommendations are included for improvements in the LEAP system and the way it was used.

*Research sponsored by the U.S. Department of Energy Office of Energy Information Validation.

†"An Assessment of the Long-Term Energy Analysis Program Used for the EIA 1978 Report to Congress," by C. R. Weisbin, R. W. Peelle, and R. G. Alsmiller, Jr.

6.2. AN APPROACH TO EVALUATION OF ENERGY-ECONOMY MODELS*

C. R. Weisbin R. W. Peelle A. S. Loeb†

[Abstract of ORNL-5742, March 1981; also *Energy* 6(10), 999 (1981); also *Technol. App. Bull.*, in press]

The increasing importance of models that stress the energy portion of the economy has created the demand for evaluation, by other than the model developer, to determine the strengths and weaknesses of a model with respect to the needs of the intended user. The literature describing how evaluations should be performed shows agreement on several basic components of the evaluation process, but at the same time is confusing because, although the same words may be used to describe these components, they often have different meanings. Although the models of interest are quantitative and the corresponding computer programs yield numerical results, published guidelines on model evaluation have lacked operationally defined checkpoints that can be used to summarize concisely and quantitatively the status of knowledge at a given time. Information related to desirable endpoints such as user requirements in terms of model outputs and associated maximum uncertainties, data sensitivities,

etc., have customarily been expressed in more qualitative terms that are difficult to measure.

A set of guidelines is developed for evaluating energy-economy models; the components of a thorough model evaluation are defined in an operational manner. Quantitative measures based on sensitivity and uncertainty analysis of results important to the user are included. Examples of the various components are drawn from the literature and from evaluation by Oak Ridge National Laboratory of the Long-Term Energy Analysis Program (LEAP) used by the Energy Information Administration for making projections to year 2020 included in that agency's 1978 Annual Report to Congress. Possible paths for future development of evaluation techniques are indicated, particularly those for which evaluation components must be more strictly defined after further practical experience is gained.

*Research sponsored by U.S. DOE Office of Energy Information Validation.

†Energy Division.

6.3. ADJOINT SENSITIVITY THEORY AND ITS APPLICATION TO LEAP MODEL 22C*

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J. D. Drischler J. E. Horwedel†
J. L. Lucius† J. W. McAdoo†

[Abstract of ORNL/TM-7789, June 1981; also *Operations Research*, in press]

There has been available for some time a mathematical methodology, called adjoint sensitivity theory, for determining the sensitivity of results of interest to each of the data elements that enter into the calculations. In this paper adjoint sensitivity theory is discussed and its applicability to a large energy-economics model is demonstrated by applying it to a specific calculation carried out with the Long-Term Energy Analysis Program (LEAP). Numerical results for dR/dx , where R is the result of interest and x is any one of the data elements, are presented for all x for which dR/dx is appreciable and for several definitions of R . In a number of cases the accuracy of the dR/dx obtained by adjoint methods has been verified by direct calculations and these comparisons are also presented.

The application of the theory as presented here requires extensive development work in that a large amount of analytic differentiation must be carried out and a substantial effort is needed to evaluate these derivatives. In the course of the work, a method was developed that would allow all of the required derivatives to be obtained numerically using LEAP and this method is presented and discussed. This numerical method is applicable only to codes with the very special modular structure of LEAP.

*Research sponsored by U.S. DOE Office of Energy Information Validation.

†UCC-ND Computer Sciences Division.

6.4. MATHEMATICAL FOUNDATIONS OF LEAP*

E. M. Oblow

(Abstract of ORNL-5754, April 1981)

As part of the ORNL energy model validation program an analysis was made of the mathematical structure of the LEAP energy modeling code. Emphasis was placed on understanding the equations which the code attempts to solve and the iterative numerical algorithm used to achieve a converged solution. The equations were found to be highly nonlinear, similar in form to those found in equilibrium boundary value problems. A combination of a standard nonlinear Gauss-Seidel approach and a modification of Newton's method was used as the basis of an iterative algorithm to find local solutions to the equilibrium equations. Several weaknesses in this approach are discussed, mainly those dealing with the computation of relaxation coefficients in the Newton's method segment of the algorithm. The chief failing of the method as employed in LEAP is the use of a diagonal Jacobian matrix for Newton's method with an *ad hoc* scheme for calculating the diagonal matrix elements. Most problems encountered had large off-diagonal elements leading to a characteristic oscillatory breakdown in convergence. It was concluded that the LEAP algorithm cannot be used to achieve convergence in an automated fashion for any general problem. In practice hand intervention is needed in most

cases. Recommendations are made to approximate certain key elements of the Jacobian matrix to improve the general usefulness of the algorithm. Analytic expressions for the approximate elements are derived so as to facilitate implementation in LEAP. Other aspects of the numerical procedures are discussed, and recommendations for their improvement are also made.

*Research sponsored by U.S. DOE Office of Energy Information Validation.

6.5. THE ECONOMIC FOUNDATIONS OF LEAP MODEL 22C*

J. A. Hansen[†] M. Becker[‡] J. L. Trimble[§]

(Abstract of ORNL-5748, April 1981)

The Department of Energy's Energy Information Administration (DOE/EIA) annually publishes energy future projections in its *Annual Report to Congress*. These projections, which are divided into short- mid- and long-term, are based on DOE/EIA energy-economy models. The model for long-term projections is the DOE/EIA's Long Term Energy Analysis Program (LEAP). This report identifies and presents an initial assessment of the major underlying economic assumptions of LEAP Model 22C, the version of LEAP used to prepare the 1978 *Annual Report to Congress*. The major assumptions of Model 22C are: competitive markets; constant input proportions in production processes; constant returns to scale in production processes; unlimited supplies of capital and variable inputs at fixed prices; continuous (infinitely divisible) units of input; investment based on perfect foresight; technical change limited to exogenous forecasts; minimal regional differences (except for coal production); unlimited supplies of foreign oil at exogenously specified prices; very little government regulation; and final demand quantities which are essentially exogenously specified.

*Research sponsored by U.S. DOE Office of Energy Information Validation.

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6.6. OPERATING EXPERIENCE WITH LEAP FROM THE PERSPECTIVE OF THE COMPUTING APPLICATIONS ANALYST*

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B. C. Toney[†]

(Abstract of ORNL/CSD/TM-141, May 1981)

In conjunction with the Energy Information Administration's interest in forecasting the long-range impact of new technologies, the Long-Term Energy Analysis Program (LEAP), which was used for the energy price-quantity projections in the 1978 Annual Report to Congress (ARC '78), was used in an ORNL research program to develop and demonstrate a procedure for evaluating energy-economic modeling computer codes and the important results derived therefrom. The LEAP system used in the ORNL research, the mechanics of executing LEAP, and the personnel skills required to execute the system are described. In addition, a LEAP sample problem, subroutine hierarchical flowcharts, and input tables for the ARC '78 energy-economic model are included. Results of a study to test the capability of the LEAP system used in the ORNL research to reproduce the ARC '78 results credited to LEAP are presented. This document is one report in a series of reports which describe the ORNL LEAP-related research program.

*Research sponsored by U.S. DOE Office of Energy Information Validation.

[†]UCC-ND Computer Sciences Division.

6.7. CONTRIBUTION TO SCREENING METHODOLOGY*

F. G. Perey

(Abstract of ORNL-5744, March 1981)

The screening method for determining objective approximate values for the sensitivity coefficients of the output of a computer code to the input data is analyzed. An "estimator" proposed to solve such problems is used to investigate one of the outputs of a simplified model of the Long-Term Energy Analysis Program (LEAP) of the Energy Information Administration (EIA). The

results are presented and compared to the true values of the sensitivity coefficients. The estimator produces results which appear more arbitrary than approximate. The screening problem is analyzed from the point of view of a theory of logical inference, and it is argued that no objective, approximate, and useful solution exists for a code treated as a "black box" (that is to say that no assumptions, prior or otherwise, are made about the code, and we have only the ability to run the code) since no degree of confidence can be placed on the solution. Requirements on additional information needed to produce a solution which could qualify as objective, approximate, and useful are investigated. It is shown that such information can in principle be obtained, and the problem then admits a solution. The emphasis in this report is on the conceptual aspects rather than on the mathematical details of the problem and its solution.

*Research sponsored by U.S. DOE Office of Energy Information Validation.

6.8. INPUT PARAMETERS FOR LEAP AND ANALYSIS OF THE MODEL 22C DATA BASE*

L. Stewart[†] M. Goldstein[†]

(Abstract of ORNL-5746, May 1981)

The input data for the Long-Term Energy Analysis Program (LEAP) employed by the Energy Information Administration for projections of long-term energy supply and demand in the United States have been studied and additional documentation provided. Particular emphasis has been placed on the LEAP Model 22C input data base, which was used in obtaining the output projections which appear in the 1978 Annual Report to Congress. Definitions, units, associated model parameters, and translation equations are given in detail. Many parameters were set to null values in Model 22C so as to 'turn off' certain complexities in LEAP; these parameters are listed in Appendix B along with parameters having constant values across all activities. The values of the parameters for each activity are tabulated along with the source upon

which each parameter is based - and appropriate comments provided, where available. The structure of the data base is briefly outlined and an attempt made to categorize the parameters according to the methods employed for estimating the numerical values. Due to incomplete documentation and/or lack of specific parameter definitions, few of the input values could be traced and uniquely interpreted using the information provided in the primary and secondary sources. Input parameter choices were noted which led to output projections which are somewhat suspect. Other data problems encountered such as input errors and code overrides are summarized. Some of the input data were corrected and a "revised" base case was constructed. The output projections for this "revised" case are compared with the Model 22C output for the year 2020 for one of the ten sectors, the Transportation Sector. LEAP could be a very useful tool, especially so in the study of emerging technologies over long time frames. The results summarized in this report could serve to improve the data base and the LEAP code and therefore enhance further projections of long-term energy supply and demand.

*Research sponsored by U.S. DOE Office of Energy Information Validation.

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6.9. LEAP MODEL 22C RESULTS VERSUS EIA 1979 REPORT*

M. Goldstein[†]

(Abstract of ORNL/TM-7739, March 1981)

A comparison study of LEAP Model 22C predicted results for years 1975 and 1980 versus the compiled "real" data of EIA 1979 report for the years 1975, 1976, 1977, 1978, and 1979 has been performed. It is concluded that most of the quantities predicted by LEAP Model 22C for 1975 and 1980 are in reasonable agreement (within about 2%) with EIA 1978 data.

Many differences between LEAP Model 22C predicted results and EIA 1979 "real" data for 1975 and 1980 are also discussed.

*This work sponsored by DOE Office of Energy Information Validation.

[†]Nuclear Research Centre-Negev, Israel.

6.10. A SURVEY OF NUCLEAR FUEL-CYCLE CODES*

C. R. Thomas[†] G. de Saussure
J. H. Marable

(Abstract of ORNL/TM-7747, April 1981)

At the request of the Office of Energy Information Validation (OEIV), a two-month survey of nuclear fuel-cycle models was undertaken by the Model Evaluation Team of the Oak Ridge National Laboratory (ORNL). This report presents the information forthcoming from the survey. Of the nearly thirty codes reviewed in the survey, fifteen of these codes have been identified as potentially useful in fulfilling the tasks of the Nuclear Energy Analysis Division (NEAD) as defined in their FY 1981-1982 Program Plan. Six of the fifteen codes are given individual reviews. The individual reviews address such items as the funding agency, the author and organization, the date of completion of the code, adequacy of documentation, computer requirements, history of use, variables that are input and forecast, type of reactors considered, part of fuel cycle modeled and scope of the code (international or domestic, long-term or short-term, regional or national). The report recommends that the Model Evaluation Team perform an evaluation of the EUREKA uranium mining and milling code.

*Research sponsored by U.S. DOE Office of Energy Information Validation.

[†]Energy Division.

6.11. LFS, A STRATEGIC PLANNING CAPABILITY FOR U.S. LIQUID FUELS SUPPLY*

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(Abstract of ORNL/TM-8407, in press)

Under the sponsorship of the Division of Policy and Strategic Planning (DOE/FE), Oak Ridge National Laboratory is conducting a research program in the modeling of liquid and gaseous fuel supplies. The strategic goal is to enable quantitative assessment of competitive supply technologies and thus contribute to the

development of the knowledge base required for reducing the U.S. energy supply vulnerability to the economic and security implications of reliance upon scarce and increasingly expensive conventional petroleum resources. The specific objectives of the project are as follows:

1. Develop computerized models for U.S. liquid (LFS) and gaseous (GFS) fuels supply to enable quantitative assessment of competitive supply technologies, and provide DOE/FE with operational support in their use.
2. Develop an automated capability for LFS/GFS sensitivity and uncertainty analyses.
3. Develop a decision making methodology based on sensitivity theory to critically examine the potential benefits of various R&D programs.
4. Plan for contingencies in the event of supply interruptions (supply vulnerability analysis).
5. Provide DOE/FE with a "public sector home" for computer models and analyses.

Particular attention is being focused on representing mechanisms by which new technologies penetrate the oil market, and assuring reliable comparison of the relative economics of enhanced oil recovery, shale oil, synthetics, conventional onshore, and offshore production, in meeting United States needs for liquid fuels as functions of time.

The present document reports on the development of a comprehensive Liquid Fuels Supply (LFS) model. Particularly important features of this model include the disaggregation of the constituent liquid supply technologies, the extensive review and documentation of the data associated with the model, and the capability to treat explicitly the dynamics of investment decisions and technological change.

The purpose of this paper is to call attention to the existence of this important new analytical tool, to provide a potential analyst with a reasonably complete description of its essential features, and to discuss the benchmarking completed and underway.

Detailed network diagrams, process equations and sources of data are given. Twelve depletable resource processes are represented. These include *in situ* and surface retorting of western oil shale, tar sands, western and eastern conventional onshore production, three types of enhanced oil recovery, eastern and western offshore recovery, and Alaskan onshore and offshore production. Details of the LFS model equations used to determine the equilibrium prices and quantities of liquid fuels are included. The fundamental data parameters which provide the computational basis for each supply technology are calculated from technology-specific relationships between marginal cost and supply. LFS results are compared to projections of the National Energy Policy Plan. It is found that the LFS results generally fall within stated NEPP uncertainty ranges. A comprehensive sensitivity analysis capability, developed in conjunction with the LFS formulation, has been indispensable in identifying the differences in calculated results.

Appendices provide additional details on the methodology used to project findings rates and on the computation of marginal costs.

The Liquid Fuels Supply model described herein is also being integrated into a World Oil Model sponsored by the DOE Division of Energy Security and is being considered for application by the Energy Information Administration. With such a potentially large and important user community, documentation and review in the open literature, as well as public availability, is considered essential.

*Research sponsored by U.S. DOE Office of Fossil Energy.

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6.12. DESIGN OF A LIQUID FUELS SUPPLY MODEL FOR U.S. POLICY ANALYSIS*

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(Abstract of Energy. The International Journal, in press)

A strategic goal of highest priority for DOE's Fossil Energy Program is to reduce U.S. energy supply vulnerability to the economic and security implications of reliance upon scarce and increasingly expensive conventional petroleum resources by developing the knowledge base for alternative supply technologies. To further the understanding of how explicit and implicit technological, economic and behavioral variables affect our capability to confidently develop options to deal with such fundamental policy issues which have long term consequences, the Division of Policy and Strategic Planning in the Office of Fossil Energy of the U.S. Department of Energy (DOE/FE) has sponsored the development of a comprehensive Liquid Fuels Supply (LFS) model. Particularly important features of this model include the disaggregation of the constituent liquid supply technologies, the extensive review and documentation of the data associated with the model, and the capability to treat explicitly the dynamics of investment decisions and technological change.

The purpose of this paper is to call attention to the existence of this important new analytical tool, to provide a potential analyst with a reasonably complete description of its essential features, and to discuss the benchmarking completed and underway.

Detailed network diagrams, process equations, and sources of data are given. Twelve depletable resource processes are represented. These include *in situ* and surface retorting of western oil shale, tar sands, western and eastern conventional onshore production, three types of enhanced oil recovery, eastern and western offshore recovery, and Alaskan onshore and offshore production. Details of the LFS model equations used to determine the equilibrium prices and quantities of liquid fuels are included. The fundamental data parameters which provide the computational basis for each supply technology are calculated from technology specific relationships between marginal cost and supply.

LFS results are compared to projections of the National Energy Policy Plan. It is found that the LFS results generally fall within stated NEPP uncertainty ranges. A comprehensive sensitivity analysis capability, developed in conjunction with the LFS formulation, has been indispensable in identifying the differences in the calculated results.

The Liquid Fuels Supply model described herein is also being integrated into a World Oil Model sponsored by the DOE Division of Energy Security and is being considered for application by the Energy Information Administration. With such a potentially large and important user community, documentation and review in the open literature, as well as public availability, is considered essential.

*Research sponsored by U.S. Department of Energy.

[†]Lewin and Associates, Inc., Washington, DC.

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Section 7

SAFETY AND RELIABILITY ASSESSMENTS FOR NUCLEAR POWER REACTORS

7.0. INTRODUCTION

The prediction in our last progress report that (reactor) systems reliability and operational research would expand into a major area of work within the division was well founded. The first efforts in this direction began in the mid-1970s when a reactor safety program was initiated and the two reactor data bases CREDO and SACRD were established (see Section 8). In 1978-1979 this work was expanded when the division, through several subcontractors, began providing risk analyses for the Nuclear Regulatory Commission and also initiated studies of human factors as they relate to reactor safety.

Human Factors Studies

The human factors studies began in late 1978 when during the course of our reactor safety studies we recognized that human beings (specifically control room operators) should be considered as an integral part of reactor safety systems. A literature search yielded little data on control room operator performance and indicated the need for a research program to study the impact of human factors in reactor operations. NRC at that time had a need for data on operator responses to plant emergencies in order to evaluate a proposed industry design standard addressing the acceptability of manual versus automatic safety-related actions, and the Commission supported the initial studies. The TMI-2 accident underscored for the entire industry the importance of the human element and the need for research into many areas of human factors in nuclear power plants.

Our most recent studies of safety-related operator actions are represented in this report by Papers 7.1-7.4. The general approach has been to observe the responses of licensed operators to safety-related events on a training simulator and to calibrate the simulator data to "field" data, i.e., to historical data on accident events that have occurred during actual reactor operations.

These studies of operator actions led us into studies of simulator response characteristics -- their adequacy for replicating reactor responses and the possibilities for improving them (see Papers 7.5-7.7). Simulator use by other types of industries was also examined. The overall conclusion was that simulators should be considered as only one element of a complete training system, and that simulator requirements should be defined as part of an integrated systematic approach to training program development. Such a comprehensive program has recently been initiated within the division to develop a systematic method for NRC to evaluate personnel selection and training programs. Thus the initial work on operator performance has led to another area of human factors research - personnel selection and training.

The study of human performance also has extended to plant personnel outside of the control room. A study was completed for the Division of Human Factors Safety, Office of Nuclear Research Regulation of NRC, on the feasibility of licensing nuclear utility managers and officers. A summary of that study is included here as Paper 7.8. More recently, a several-year program has been undertaken for the Office of Nuclear Regulatory Research of NRC to develop and validate a computer simulation model to predict the reliability of maintenance personnel in carrying out their tasks. Two papers on the scoping/feasibility study performed prior to the initiation of this program are included here.

It is the goal of the division staff to develop these individual projects into a comprehensive integrated research program that addresses all areas of human factors - human engineering, personnel selection and training, job performance aids, human performance/reliability, and personnel safety.

Risk Assessments

The division continues to be involved in NRC risk assessments, and, because of a lack of sufficient staffing, still largely in a managerial role overseeing the work of various subcontractors. One assessment in which the division is participating in directly is concerned with the current unresolved safety issue of pressurized thermal shock (PTS) to the reactor pressure vessel. PTS refers to a reactor transient condition that leads to rapid cooldown followed by repressurization (or simultaneous cooldown and repressurization), the safety issue being thermal shock (fracture) of the vessel. A preliminary estimate of the frequency of occurrence of the conditions that could lead to PTS has been made, and follow-on work is attempting to identify the detailed sequences that will lead to PTS, together with their probabilities of occurrence.

Another risk assessment study completed during this reporting period consisted in the development of a methodology (by subcontractor JBF Associates, Inc.) for performing flood risk analyses for nuclear power plants, once the flooding level versus probability is known. This work included developing two computer programs: ESP, which screens accident sequences to identify those that are potentially significant contributors in the event of a flood; and NOAH, which provides detailed qualitative analysis of the plant systems identified by ESP.

A third project recently completed (but not yet published) was performed by Torrey Pines Technology to develop qualitative and quantitative approaches for the evaluation of common cause failures and to apply them to the analysis of the auxiliary feedwater systems of several PWR plants.

Other on-going nuclear power plant risk studies are addressing (1) acceptable levels of risk, (2) utilization of risk analysis and risk criteria, (3) methods for evaluating risk analyses, (4) fire risk, (5) accident sequence precursor methodology programs, (6) the impact of truncation on interim reliability program sequences, and (7) in-plant reliability data.

Finally, several division members are participating in the new international Society for Risk Analysis as it relates to NRC risk assessments, one activity being the publication of the society's newsletter. In addition, it is to be noted that a report cited in our last progress report as part of this program has now been published in book form (*Acceptable Risk*, by Baruch Fischhoff, Sarah Lichtenstein, Paul Slovic, Stephen L. Derby, and Ralph L. Keeney, Cambridge University Press, New York, NY, 1981).

7.1. CRITERIA FOR SAFETY-RELATED NUCLEAR POWER PLANT OPERATION ACTIONS: INITIAL PRESSURIZED WATER REACTOR (PWR) SIMULATOR EXERCISES*

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C. Crowe[‡] P. M. Haas

[Abstract of ORNL/NUREG/TM-434 (NUREG-CR-1908),
September 1981]

Nuclear plant control room simulator exercises for seven pressurized water reactor (PWR) events were conducted with ten control room teams. Operator performance was recorded by an automatic Performance Measurement System (PMS) and by subjective evaluation. Response times and error probabilities were estimated for selected actions. The effect of some experimental variables on operator performance is discussed. Application of the response time data to ANSI Standard N660 is included. The data collected will later be compared to field data being collected for similar events in order to provide a basis for extrapolation of simulator data to actual operating conditions. Ultimately, a base of human performance data will be developed from simulator experiments which can be used to establish criteria and standards, evaluate effects of key performance-shaping factors, and support safety/risk assessment analyses.

*Research sponsored by U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research.

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7.2. CRITERIA FOR SAFETY-RELATED NUCLEAR POWER PLANT OPERATION ACTIONS: PRESSURIZED WATER REACTOR (PWR) FIELD DATA COLLECTION*

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D. K. Baer[‡] P. M. Haas

[Abstract of ORNL/Sub-7688/1, September 1981]

This report is one of a series covering ORNL's NRC-funded program on safety-related operator actions. The program is designed to collect and assess data on the performance of

nuclear power plant (NPP) operators during abnormal or emergency events in order to support development of criteria and standards for safety-related operator actions. The report summarizes the results and conclusions from the initial effort to collect field data on events in Pressurized Water Reactor (PWR) plants.

Techniques for data collection as well as the problems and limitations of field data are reported, along with the initial results. The quantitative data emphasized in this study was the time required for operators to correctly diagnose the abnormal or emergency event and initiate the first correct manual action. Plots of a distribution of response-time data are presented for three PWR events, and the data are fit reasonably well by a log-normal model. Subjective data on key performance-shaping factors during response to abnormal or emergency events were obtained by means of a structured survey form completed by a sample of approximately 40 operators. Results of that survey indicate that training, procedures, and clarity of diagnostic displays are viewed as more critical problems than control room design.

*Work sponsored by the Nuclear Regulatory Commission
Office of Nuclear Regulatory Research.

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7.3. SAFETY-RELATED OPERATOR ACTIONS IN NUCLEAR POWER PLANTS*

P. M. Haas

(Reprint of Trans. Am. Nucl. Soc. 38, 240, 1981)

Introduction

The Safety Related Operator Actions Program at Oak Ridge National Laboratory (ORNL) is intended to provide the U.S. Nuclear Regulatory Commission (NRC) with quantitative and qualitative data on the performance of nuclear power plant operators during accident events. The data are necessary to support licensing decisions, standards development, and research in a number of areas related to operational safety. The program, developed after a preliminary assessment of available historic data,¹ consists primarily of three tasks: (a) collection

and assessment of data from controlled simulator exercises. (b) collection and assessment of field data, and (c) calibration of simulator to field data.

Simulator Exercises

The simulator exercises are being conducted by General Physics Corporation (GPC) at the Tennessee Valley Authority Training Center near Soddy-Daisy, Tennessee. Licensed nuclear plant operators from different utilities perform a controlled series of exercises that involve a number of transient events (typically six to ten during an 8-h period), with periods of "normal operation" in between. Data are recorded automatically using the Performance Measurement System developed previously by GPC under contract to the Electric Power Research Institute. The operators are informed that data are being recorded as part of a study for ORNL sponsored by the NRC, and the instructor is not permitted to "interfere" with the exercise after it is initiated. Other than these two factors, the experiments are conducted as routine training exercises. There is no attempt to create a "more realistic environment" by introducing additional simulated "stress" or other artificial factors. The fact that some difference exists between the environment of a training simulator and an actual plant under accident conditions is recognized and is, in fact, the reason for the intended calibration of simulator data to field data.

Field Data Collection

The field data collection is being carried out by Memphis State University Center for Nuclear Studies. Licensee event report summaries provided by the Nuclear Safety Information Center at ORNL are used as an indicator to identify applicable events. Subsequently, site visits are conducted to compile all available records - control room logs, supervisor logs, computer output, plant upset records, etc. - that relate to the event. As much as possible, an attempt is made to review the details of the event with site personnel, especially any available who were involved with the event of interest. In addition, a written questionnaire is used to obtain subjective information from operators about significant accident events they have experienced.

Calibration of Simulator Data

The concept of "field calibrated simulation" was suggested by Swain² for application in a somewhat different context. Conceptually, the approach is similar to that which might be used for any engineering correlation. However, for the problem at hand, it is clear at the onset that it will be extremely difficult to quantify the many potentially significant variables, and that data available will be relatively few and often subjective. Therefore, the "mathematical model" can be expected to be simplistic, and the initial calibration somewhat crude. However, it does appear to be feasible to establish whether there are at least consistent relationships between simulator and field data (e.g. actions requiring more time than others in the simulator also require more time than the same actions in the field data). If such a consistency is demonstrated and key performance-shaping factors are identified in this early work, further analysis and experiments can provide the data to support a more comprehensive predictive model.

Initial Results

The initial series of ten "experimental runs" on the pressurized water reactor (PWR) (Sequoyah plant) simulator has been completed, and a report summarizing results will be published in the near future.³ The PWR field data collection is not yet complete. However, an example of the kind of quantitative data that are being extracted is provided in Fig. 7.3.1. The figure is a cumulative probability plot comparing results of simulator exercises to field data collected during the preliminary study reported in Ref. 1 for the "Inadvertent Safety Injection at Power" (ISI) event. The ordinate values are the time in seconds for the operating crew to perform the first required action (reset SI) correctly. Since a linear fit on a log scale approximates the data points reasonably well, a log-normal distribution is assumed to be reasonable for both samples. The log-mean values of response time are 1.9 and 4.8 min, respectively for the field data and simulator data. A similar analysis for the event "Dropped Rod" gave log-mean response times of 0.7 and 1.6 min, respectively, for the field data (sample size $N = 19$) and simulator data ($N = 8$). In both cases,

Fig. 7.3.1. Comparison of Simulator and Field Data: A Cumulative Probability Plot of Time Required for Correct Performance of the First Required Action for the PWR ISI Event.

the mean response time was greater for these initial simulator runs than for the field data. There are a number of obvious factors -- familiarity of the test subjects with the Sequoyah plant procedures and control room, current state of training, etc. -- that could dominate or distort the results, and the data are certainly too few to support general conclusions at this time.

It is encouraging, but equally inconclusive, that the data comparison for these two events tends to support the suggestion of "consistency" between the two data sets. That is, the event with a greater mean time for response in the simulator (ISI) also has a greater mean time in the field.

Conclusion

The current program of simulator experiments and field data collection should be viewed as one necessary element in the NRC's overall approach to improving operational safety of nuclear power plants. It will provide specific information on operator response times useful for the near-term licensing and standards needs, and will initiate development of a realistic data base on operator performance necessary for a broad range of

analysis and assessments currently in progress or planned. The results and analysis of data completed at this time are very preliminary and should not be used to form general conclusions. However, they do demonstrate that the approach to development of a data base from field-calibrated simulator results should be successful.

*Research sponsored by U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research.

1. P. M. Haas and T. F. Bott, "Criteria for Safety-Related Nuclear Plant Operator Actions: A Preliminary Assessment of Available Data," NUREG/CR-0901, ORNL/NUREG/TM-330 (1979).
2. A. D. Swain, "Field Calibrated Simulation," SC-R-67-1045, Sandia Corp. (1967).
3. "Safety Related Operator Actions: Initial PWR Simulator Exercises," General Physics Corporation Report prepared for Oak Ridge National Lab., to be published as ORNL/NUREG/TM-434.

7.4. NUCLEAR POWER PLANT CONTROL ROOM TASK ANALYSIS: PILOT STUDY FOR PRESSURIZED WATER REACTORS*

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S. Eckel[‡]

[Abstract of ORNL/Sub/79-40432/1 (NUREG/CR-2598), May 1982]

The purposes of this nuclear plant task analysis pilot study were:

1. To demonstrate the use of task analysis techniques on selected abnormal or emergency operation events in a nuclear power plant;
2. To evaluate the use of simulator data obtained from an automated Performance Measurement System to supplement and validate data obtained by traditional task analysis methods;
3. To demonstrate sample applications of task analysis data to address questions pertinent to nuclear power plant operational safety: control room layout, staffing and training requirements, operating procedures, interpersonal communications, and job performance aids.

Five data sources were investigated to provide information for a task analysis. These sources were (1) written operating procedures (event-based); (2) interviews with subject matter experts (the control room operators); (3) videotapes of the control room operators (senior reactor operators and reactor operators) while responding to each event in a simulator; (4) walk-talk-throughs conducted by control room operators for each event; and (5) simulator data from the PMS.

A Westinghouse pressurized water reactor nuclear power plant simulator was utilized in this study. A simulator gives researchers the ability to observe actual control manipulations during accident scenarios. Four abnormal or emergency events were studied: nuclear instrument failure; small break loss of coolant accident (LOCA); steam generator tube leak; and inadvertent safety injection at power. Upon completion of the task analyses, computerized data reduction was performed. A PRIME 1-1000 computer was used to manage the task analytic data base.

Videotapes were felt to be the richest single source of information, but they frequently left some gaps in the data accumulation, e.g. the exact switch manipulation, and the sequence of manipulations was sometimes difficult to discern. The PMS data source provided that data. This study demonstrated the usefulness of a task analysis methodology that combines traditional data collection by analysts with PMS data to provide a complete task analytic data set. The power of a machine-readable computer data base to support various applications of task analytic data was also demonstrated.

*Research sponsored by U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research.

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7.5. TRAINING SIMULATOR REQUIREMENTS - A COMPARISON AMONG SEVERAL INDUSTRIES*

P. M. Haas

(Summary of paper presented at the Ninth Water Safety Research Information Meeting, Gaithersburg, MD, October 27, 1981)

In the TMI Action Plan (NUREG-0660), the Nuclear Regulatory Commission listed various

actions planned to upgrade nuclear power plant (NPP) simulators and their use in operator training programs. One was a review and update of the existing standard ANSI/ANS-3.5, "Nuclear Power Plant Simulators for Use in Operator Training," followed by the issuance of a regulatory guide endorsing the revised standard. This paper summarizes results, conclusions and recommendations of a study^{1,2} conducted by Oak Ridge National Laboratory (ORNL) and consultants from General Physics Corporation, Franklin Research Center, Memphis State University, and Analysis and Measurements Services in support of NRC's efforts to upgrade the existing standards and the use of simulators for training.

The primary recommendation from the study is that the nuclear industry and NRC move toward a systems approach to training such as that used by several non-nuclear industries. A series of specific actions recommended are:

- a. A cooperative front-end planning effort by NRC and industry representatives leading to a planning document and, possibly, an "umbrella" standard for the systems approach.
- b. A clear delineation to the industry of NRC's planned role and level of regulatory involvement in selection, training, simulator certification, etc.
- c. Continuation and, if possible, acceleration of NRC and industry efforts to develop a job/task analytic data base for control room personnel.
- d. Continued improvement of ANSI/ANS-3.5, possibly consolidating with ANSI/ANS-3.1 (the standard for selection and training of nuclear power plant personnel).
- e. Implementation of data acquisition systems required by NUREG-0696, as well as procedures for feedback of plant data to the simulator design evaluation process.

*Research sponsored by U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research.

1. C. A. Belsterling et al., "Specification and Verification of Nuclear Power Plant Training Simulator Response

Characteristics, Part I: Summary of Current Practices for Nuclear and Non-Nuclear Simulators," ORNL/TM-7985, NUREG/CR-2353, Vol. 1 (December 1981).

2. T. W. Kerlin et al., "Specification and Verification of Nuclear Power Plant Training Simulator Response Characteristics, Part II: Conclusions and Recommendations," NUREG/CR-2353, Vol. 2, (ORNL/TM-7985/P2) (May 1982).

7.6. SPECIFICATION AND VERIFICATION OF NUCLEAR POWER PLANT TRAINING SIMULATOR RESPONSE CHARACTERISTICS - PART I. SUMMARY OF CURRENT PRACTICES FOR NUCLEAR AND NON-NUCLEAR SIMULATORS*

C. A. Belsterling[†] S. Pandey[†] J. C. Praskiewicz[‡]
E. J. Kozinsky[‡] L. Felkins[§] P. M. Haas

[Abstract of ORNL/TM-7985/1 (NUREG/CR-2353, Vol. 1), December 1981]

This report reviews the methods and practices of the nuclear industry for specifying and verifying the performance characteristics of simulators used to train nuclear power plant operators. It also reviews the training simulator methods and practices of selected non-nuclear industries (in particular, the civilian and military aircraft industries) and compares them with those of the nuclear industry. In addition, it identifies non-nuclear methods that might be profitably adopted by the nuclear industry and perhaps included in a simulator standard endorsed by the Nuclear Regulatory Commission. Final conclusions and recommendations are discussed in a companion report [NUREG/CR-2353, Vol. 2 (ORNL/TM-7985/P2)].

*Research sponsored by U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research.

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7.7. SPECIFICATION AND VERIFICATION OF NUCLEAR POWER PLANT TRAINING SIMULATOR RESPONSE CHARACTERISTICS - PART II. CONCLUSIONS AND RECOMMENDATIONS*

P. M. Haas D. L. Selby
T. W. Kerlin[†] L. Felkins[‡]

[Abstract of ORNL/TM-7985/2 (NUREG/CR-2353, Vol. 2), May 1982]

This report is the second volume of a two-volume report summarizing the findings, conclusions, and recommendations of a survey study conducted by Oak Ridge National Laboratory and its subcontractors for the U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research. The objectives of the study were: (1) to gather information on standards and practices of the nuclear industry and certain non-nuclear industries which might be used to specify and verify the response characteristics of nuclear power plant training simulators; (2) to compare findings from the nuclear and non-nuclear industries in order to identify which of the non-nuclear practices might be profitably adopted by the nuclear industry; and (3) to recommend actions that could be pursued, by NRC or by the nuclear industry, to improve standards and practices. The first volume of the report summarized the information gathered from the nuclear and non-nuclear industries. The fundamental conclusion was that simulators should be recognized as only one element of the complete training system, and simulator requirements should be defined as part of an integrated systematic approach to training program development. The Systems Approach to Training (SAT) or Instructional Systems Development (ISD) process developed and used extensively in the U.S. military was cited as having potential for application to the nuclear industry.

This report presents the conclusions and recommendations from the survey study. The general conclusions are:

(1) The nuclear industry should adopt, and NRC regulatory and research actions should support, the systems approach to training as a structured framework for development and validation of personnel training systems.

(2) The potential exists for significant improvement in the ability to obtain a sufficient data base to assess simulator fidelity. Systems Identification Technology offers a potential framework for model validation and should be investigated further for its utility and practicality in assessing NPP simulator fidelity. More specifically, installation of the data collection/recording equipment required by NRC in NUREG-0696 could provide a vastly improved source of data for simulator fidelity assessment.

(3) The NRC needs to continue its post-TMI actions to involve itself more rigorously and more formally in the entire process of NPP personnel training system development (including simulator performance specification and verification). However, this involvement should be a "participative" one with industry, in which the NRC determines guidelines, standards, or generic criteria to be used, and reviews industry programs for conformance but does not dictate the details of each program.

(4) The existing simulator standards and guidelines should be examined and reorganized as necessary to support the use of systems approach to training. The standards should require and support a holistic approach to training system development that recognizes simulators and simulator training as only parts of the complete training program and "full-scope, high-fidelity, site-specific" simulators as only one useful training device.

Some of the specific recommendations for adopting the SAT/ISD process to the nuclear industry are:

(1) Formation of an NRC/industry planning/coordination group to define respective roles, assess resource requirements, and set a schedule for implementation.

(2) A program planning study to develop a programmatic plan for the adaptation and use of SAT/ISD methodology by NRC and the nuclear industry, and for the research necessary to support NRC's use of the methodology.

(3) Development of a user's guide and NRC/industry workshops to establish common terminology and practice.

(4) A pilot study demonstrating application of the adopted SAT/ISD methodology to an actual nuclear industry training program.

*Research sponsored by U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research.

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7.8. REPORT ON THE FEASIBILITY OF THE LICENSING/CERTIFICATION OF NUCLEAR UTILITY MANAGERS AND OFFICERS*

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M. Colella[†] P. Haas D. Messer[†]
J. Sheehan[†] S. Thomas[†]

(Abstract of a letter report to NRC, December 18, 1981; to be published in a future NRC report)

Section 307(b) of Public Law 96-295 directed the Nuclear Regulatory Commission (NRC)

to undertake a study of the feasibility and value of licensing... plant managers of utilization facilities and senior licensee officers responsible for operation of such facilities.

The Licensee Qualifications Branch of the Division of Human Factors Safety in turn contracted with the Oak Ridge National Laboratory (ORNL) to provide technical assistance in completing this study. This report presents the results of ORNL's efforts, including those of its subcontractor, Science Management Corporation (SMC).

The required information was collected through interviews. Analysis of responses proceeded throughout the interview phase, and team members frequently met informally to dis-

cuss results as the study progressed. A number of specific findings began to emerge and, as a result, it was possible to further sharpen the focus of particular questions. A more formal analysis was undertaken after most of the interviews had been completed.

The consensus was that the licensing of management personnel as a legal requirement administered in a process similar to that used for the SRO license is undesirable and would probably be counterproductive in terms of both public health and safety and the effective operation of nuclear power plants. However, a need was identified for assuring that line management personnel possess the managerial and technical education, training, and experience necessary to perform their jobs with a high level of effectiveness.

*Research sponsored by U.S. Nuclear Regulatory Commission Office of Nuclear Reactor Regulation.

†Science Management Corp., Washington, DC.

7.9. THE NUCLEAR POWER PLANT MAINTENANCE PERSONNEL RELIABILITY PREDICTION (NPP/MPRP) EFFORT AT OAK RIDGE NATIONAL LABORATORY*

H. E. Knee P. M. Haas A. I. Siegel†

(Abstract of paper presented at the International ANS/ENS Topical Meeting on Probabilistic Risk Assessment, September 20-24, 1981, Port Chester, New York)

Human errors committed during maintenance activities are potentially a major contribution to the overall risk associated with the operation of a nuclear power plant (NPP). An NRC-sponsored program at Oak Ridge National Laboratory is attempting to develop a quantitative predictive technique to evaluate the contribution of maintenance errors to the overall NPP risk. The current work includes a survey of the requirements of potential users to ascertain the need for and content of the proposed quantitative model, plus an initial job/task analysis to determine the scope and applicability of various maintenance tasks. In addition, existing human reliability prediction models are being reviewed and assessed with respect to their applicability to NPP maintenance tasks. This paper discusses the status of the program and summarizes the results to date.

*Research sponsored by U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research.

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7.10. JOB ANALYSIS OF MAINTENANCE MECHANIC POSITION FOR THE NUCLEAR POWER PLANT MAINTENANCE PERSONNEL RELIABILITY MODEL*

**A. I. Siegel† W. D. Bartter†
F. F. Kopstein†**

[Executive Summary of ORNL/TM-8301 (NUREG/CR-2670), June 1982]

An analysis was performed of the job of maintenance mechanics in nuclear power plants in order to provide a part of the information required for modeling nuclear plant maintenance activities. It is believed that such a model would provide substantial insights into the various human, equipment, and environmental factors that are likely to affect reliability of maintenance personnel, and thereby suggest and allow evaluation of standards, design changes or other modifications to improve reliability and minimize public risk.

Method

The task list method of job survey was selected because the approach minimizes data acquisition costs, interferes minimally with the work of job incumbents, is comprehensive and objective, provides quantitative data, and seemed highly appropriate for achieving the goals of the work. In collaboration with supervisory personnel at BWP and PWR nuclear power plants, a comprehensive list of 107 tasks performed by maintenance mechanics was developed. The tasks were grouped within six generic work functions: remove and install, test and repair, inspect and perform preventive maintenance, miscellaneous, communication, and report preparation.

The lists were assembled into appropriate questionnaire form. For each listed task, the form inquired into: frequency of performance, task completion time, safety consequences of improper performance, and the amount of training required to perform the task proficiently. Additionally, scaled information was requested about the amount of each of seven abilities: (1) visual speed, accuracy, and recognition, (2) gross motor coordination, (3) fine manual dexterity, (4) strength and stamina, (5) cognition, (6) memory, and (7) problem solving required for function completion. Survey forms were distributed to a

sample of 27 nuclear power plants for completion and return. Thirty-one maintenance mechanics representing 17 of these nuclear power plants returned completed forms.

Results

The work yielded a diversity of information relative to the job of the maintenance mechanic in nuclear power plants. The information obtained will be used to develop the logic and content of a model for quantitatively assessing the activities of maintenance personnel, especially from the point of view of public safety.

The frequency of performing various maintenance mechanic tasks was bimodally distributed with tasks falling largely into one of two classes: (1) between once a year and once every six months, and (2) about once a week.

More than half of the tasks were identified as having potential risk consequences if improperly performed. Considering the "consequences" of improper task performance data, the five tasks with the greatest risk implications in the case of inadequate performance were: (1) remove and install reactor and dry-well heads, (2) test and repair reactor system components, (3) remove and install pressurizer mechanical relief valves, (4) test and repair pressurizer relief valves, (5) remove and install core spray pumps, seals, and valves. Hierarchically, the public risk associated with the various functions was: (1) remove and install, (2) test and repair, (3) preventive maintenance, (4) miscellaneous tasks, (5) communication, (6) report preparation.

Investigation of the reliability of the data indicated that considerable trust can be placed in the results.

Implications

From the point of view of model development, the results suggested that any model which considers maintenance mechanics' performance as related to public risk/safety must possess a sophisticated logic, provide information on training and refresher training and skill retention, consider tasks which are as long as or greater than one day in duration, allow parametric manipulation of various abilities, and the broad spectrum in

nature. Moreover, the model must consider these and other effects (situational and constraining factors) interactively and combinatorially.

Complete and cross validated task analyses are recommended for those frequently performed tasks possessing the highest public risk consequences.

Similar job surveys are suggested in other maintenance areas, i.e., instrument and control technicians, electricians, and supervisors.

*Research sponsored by U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research.

†Applied Psychological Services, Inc.

7.11. METHODOLOGY FOR FLOOD RISK ANALYSIS FOR NUCLEAR POWER PLANTS*

D. P. Wagner[†] M. L. Casada[†]

[Summary of paper presented at the Workshop on High Consequence/Low Probability Risk Analysis, Crystal City, VA, June 15-18, 1982]

The Reactor Safety Study¹ and the Lewis Committee Report² identify floods as external hazards that warrant further investigation for the assessment of nuclear power plant risks. The principal importance of floods arises from their potential to produce essentially simultaneous failure and/or degradation of several components, resulting in increased system failure probability. Such flood-induced failures can be characterized as common cause failures.

The purpose of the methodology for flood risk analysis is to provide a rational and generally applicable basis for assessing the impact of floods on nuclear power plant risk. The procedures are consistent with accepted risk assessment methods and are based on techniques of fault tree³ and common cause failure analysis.⁴

The analysis input consists of system fault trees, component "vulnerability" location information, flood pathways and component failure probabilities. The component vulnerability location describes the minimum flood level that will fail or degrade the component either by directly causing the component degradation or by a command fault. For example, a component that is protected by a retaining wall is not affected until the

flood reaches a level sufficient to overflow the barrier. Similarly, a component that depends on controls or electrical circuitry at a lower elevation can be commanded to malfunction by a flood at this lower elevation.

The component's response to submersion is described by changes in the component failure rate. In many cases, a component will fail with probability one upon submersion. Many components experience a substantial increase in failure rate upon submersion. Still other components may remain unaffected by submersion, with no change in their failure rates.

The analysis is performed by first incrementing flood level and identifying the components submerged at each increment. The system minimal cut sets are examined to determine if all members of a minimal cut set are submerged. Standard reliability techniques⁵ calculate system failure probabilities using modified failure rates for the submerged components. Component "importance" as a function of flood level can also be determined.

Useful results are obtained from both the qualitative and the quantitative analyses. The qualitative analysis generates two lists of information: (1) the components that are failed or degraded by each incremental flood level; and (2) the minimal cut sets that are failed or degraded by each incremental flood level. These lists are valuable in determining structurally critical components that can be provided extra protection against flooding by barriers and/or relocation. The quantitative analysis produces the probability of system failure versus flood level, giving the increase in system failure probability as the flood depth increases. These changes in system failure probability allow calculation of the increased in accident sequence frequencies as a measure of the increased risk due to floods. Figure 7.11.1 shows the general shape of the accident sequence frequency function as a function of flood probability.

*Research sponsored by U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research.

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1. "Reactor Safety Study," WASH-1400 (NUREG 75/014), USNRC, October 1975.

Fig. 7.11.1. General Shape of the Accident Sequence Frequency Function as a Function of Flood Probability.

2. H. W. Lewis et al., "Risk Assessment Review Group Report to the U.S. Nuclear Regulatory Commission," NUREG/CR-0400, September 1978.
3. J. B. Fussell, "Fault Tree Analysis - Concepts and Techniques," *Generic Techniques in Systems Reliability Assessment*, Noordhoff, Lyden, 133-162 (1976).
4. D. P. Wagner, "A Procedure for Common Cause Failure Analysis of Complex Systems," Master's Thesis, The University of Tennessee, Knoxville, TN, August 1977.
5. J. B. Fussell, "How to Hand Calculate System Reliability and Safety Characteristics," *IEEE Transactions on Reliability*, Vol. 24, 169-174, August 24, 1975.

7.12. FLOOD RISK ANALYSIS METHODOLOGY DEVELOPMENT PROJECT - FINAL REPORT*

D. P. Wagner[†] M. L. Casada[†] J. B. Fussell[†]

[Abstract of ORNL/TM-8314 (NUREG/CR-2678), June 1982]

This document presents the concepts and methodology necessary to perform flood risk analysis for nuclear power plants, once the flooding level versus probability is known. The methodology is consistent with accepted probabilistic risk assessment (PRA) techniques and is usable either during the normal course of a PRA or as

an "add on" analysis for an existing PRA. The methodology fulfills many of the requirements for flood analysis suggested by the Nuclear Regulatory Commission's recent PRA Procedures Guide (NUREG/CR-2300).

The basic inputs to the methodology are:

- accident sequences and their consequence categories,
- fault trees for the events that comprise the accident sequences,
- occurrence probabilities for the accident sequences and the events that comprise the accident sequences,
- basic event failure data,
- basic event vulnerability elevations for flood events, and
- flood occurrence probabilities.

The flood analysis procedure allows screening of the accident sequences to determine the accident sequences that are potentially significant contributors to risk due to flooded effects. A qualitative flood simulation identifies flooded minimal cut sets and critical flood levels for system failures using the system fault trees. The quantitative analysis uses these results to calculate the flood's contribution to:

- system failure probability,
- accident sequence occurrence frequency, and
- consequence category occurrence frequency.

The appendices to this report describe two example applications of the flood risk methodology to systems of the Surry Power Station, the pressurized water reactor (PWR) used in the Reactor Safety Study (WASH-1400). The first application considers an external flooding scenario and the second application considers an internally generated flood. The results include the contributions of these floods to the Reactor Safety Study results as a function of flood probability.

Additional results of the project are two computer programs that aid in the accident sequence

screening and qualitative flood simulation segments of the flood risk analysis. These programs are described in a separate report.

*Research sponsored by U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research.

[†]JBF Associates, Inc.

7.13. ESP AND NOAH - COMPUTER PROGRAMS FOR FLOOD RISK ANALYSIS OF NUCLEAR POWER PLANTS*

D. P. Wagner[†] D. F. Montague[†] J. J. Rooney[†]
J. B. Fussell[†] L. S. Baker[†]

[Abstract of ORNL/TM-8313 (NUREG/CR-2677), June 1982]

This report describes a computer program package that aids in assessing the impact of floods on risk from nuclear power plants. The package consists of two distinct computer programs: ESP and NOAH.

The ESP program improves the efficiency of a flood analysis by screening accident sequences and identifying accident sequences that are potentially significant contributors to risk in the event of a flood. Input to ESP includes accident sequences from an existing risk assessment and flood screening criteria. The output from ESP includes:

- accident sequences that are potentially significant contributors to risk,
- specific plant systems contained in these accident sequences that may require detailed analysis, and
- a quantitative estimate of the flood contribution to risk.

The NOAH program provides detailed qualitative analysis of the plant systems identified by ESP. Input to NOAH includes:

- the system fault tree from the existing risk assessment,
- vulnerability elevations for each component represented in the fault tree, and
- a detailed flood level profile for the plant.

NOAH performs a qualitative flood simulation of the fault tree to determine:

- flooded minimal cut sets; that is, minimal cut sets that have all their components submerged by the flood,
- partially flooded minimal cut sets when no flooded minimal cut sets are found,
- flood protection sets; that is, groups of components that can be protected to mitigate the flood effects on the system, and
- the critical flood level; that is, the lowest flood level where at least one minimal cut set is submerged.

The output from NOAH is directly usable in quantitative evaluations of flood effects on plant risk.

This report contains detailed user's manuals for both the ESP and NOAH computer programs. Input and output of both programs is described and illustrated with example problems. Detailed programmer's guides are also provided in the appendices.

This report assumes that the reader has experience in the areas of fault tree analysis and risk assessment methodology.

*Research sponsored by U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research.

[†]JBF Associates, Inc.

7.14. EVALUATION OF PRESSURIZED THERMAL SHOCK*

R. C. Kryter[†] R. D. Cheverton[‡] F. B. K. Kam[§]
T. J. Burns R. A. Hedrick[‡] C. W. Mayo[§]

[Abstract of ORNL/TM-8072 (NUREG/CR-2083), October 1981]

This report evaluates the threat to reactor vessel integrity posed by pressurized thermal shock (overcooling) events. The study focuses on pressurized-water reactors manufactured by the Babcock and Wilcox Company, as typified by the Oconee Unit 1 Plant. The thermal-hydraulic transient data used as input to the linear elastic fracture-mechanics analysis of vessel flaw propagation were calculated by others, using the IRT and TRAC computer codes. In some hypothetical transients, vessel failure due to pressurized thermal shock is predicted early in reactor life, but identified calculational approximations, modeling deficiencies, and uncertainties in the probabilities of occurrence for the overcooling transients studied suggest that the quantitative results obtained should be regarded as preliminary estimates having substantial associated uncertainties. These areas of uncertainty are evaluated and recommendations are made for further work.

*Research sponsored by U.S. Nuclear Regulatory Commission.

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[¶]SAI Consultants.

Section 8

INFORMATION ANALYSIS AND DISTRIBUTION

8.0. INTRODUCTION

Information analysis and distribution, an important activity of the division ever since the Radiation Shielding Information Center was established in 1963, comprises an increasingly larger fraction of the division's program. Under the management umbrella of EPIC (Engineering Physics Information Centers), we have continued to operate specialized information centers in several areas. Throughout the years RSIC has served the international scientific community as a technical institute for shielding design and analysis, and, with its history and experience, RSIC has also served as a guide for establishing and operating other EPIC centers. One of these, the Technical Data Management Center (TDMC), is sponsored by the Nuclear Regulatory Commission (NRC) to support NRC programs in technical data management and computer technology.

During this reporting period a new center, the Center of Model Evaluation Technology (COMET), was established under EPIC to perform information analysis in support of an ORNL model evaluation program for the DOE Energy Information Administration (see Section 6 of this report). COMET's activities have now been phased out, with its computing technology transferred to the National Energy Software Center (NESC) at Argonne National Laboratory and other material archived within the division.

Similarly, the Reactor Safety Research Data Repository (RSRDR), operated under EPIC for NRC and described in our last progress report, has been phased out. The RSRDR data bases have been transferred to Idaho Nuclear Engineering Laboratory (INEL), where they have been integrated into an existing data bank operated for NRC by EG&G.

Under another branch of the division, two reactor data bases, both established around 1976, are maintained for the DOE Office of Reactor Research and Technology: CREDO (Centralized Reliability Data Organization), which provides advanced reactor component reliability/availability data; and SACRD (Safety Analysis Computerized Reactor Data), which collects data needed as input for computer codes used for fast reactor safety studies.

Finally, during this reporting period the division has participated in establishing the Carbon Dioxide Information Center (CDIC) under the auspices of Carbon Dioxide Research Division of DOE's Office of Basic Energy Sciences. CDIC is administered by the ORNL Information Division and staffed by individuals from the Information, Engineering Physics, Environmental Sciences, and Computer Sciences Divisions.

The Engineering Physics Information Centers (EPIC)

Radiation Shielding Information Center (RSIC). -- RSIC 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 requesters.

The RSIC scope includes the physics of interaction of radiation with matter; radiation production, protection, and transport; radiation detectors and measurements; engineering design techniques; shielding materials properties; computer codes useful in research and design; and nuclear data compilations. The goals of RSIC are to function as a technical institute to provide information (bibliographic and other data, computer codes, technical advice) upon request; collect, evaluate, enrich, distill, and repackage information to extend the state of the art, bringing into the public domain technology more usable and more valuable than the sum of the input; and to initiate and effect research and development in areas of need that can be managed by RSIC personnel.

During this reporting period work has continued in the development of fine- and broad-group cross-section libraries (see Section 2 of this report). The 174 neutron, 38 gamma-ray group VITAMIN-E general-purpose cross-section library based on ENDF/B-V now contains 60 materials. It has been used with success for fast reactor core and shielding applications and various other radiation transport calculations within the Division. Release for general distribution is planned for the summer of 1982. The generation of multigroup data for the ENDF/B-V Activation, Dosimetry, and Gas Production File has also been completed and the resulting data will be released as part of VITAMIN-E.

An effort is under way to update the ENDF/B photon interaction data with the latest available evaluated data from the National Bureau of Standards.

RSIC has continued to participate in the work of the Cross Section Evaluation Working Group (CSEWG), with the Shielding Subcommittee presently chaired by an RSIC staff member. Other staff members are actively involved with the American Nuclear Society in developing radiation protection and shielding standards and standards for computer programming and computer code documentation.

In the area of computing technology, RSIC now makes available >600 computer code packages and >95 data packages. The RSIC Newsletter is currently mailed monthly to 1433 people. During FY 1981, more than 3044 letters/telephone calls of requests were processed by RSIC, resulting in 8900 separate activities required to satisfy the requests. A total of 97 persons came for an orientation visit and/or to use the center's facilities.

The RSIC computer-based information system (SARIS), containing more than 12,000 literature references, is accessible on-line nationally through the DOE RECON system. The sixth indexed bibliography was printed from the data files using computer-based typesetting programs.

The computer-based statistical record system (ADES), operable on the EPIC minicomputer (ECLIPSE), is now used to compile information about the profile of the RSIC user community.

Technical Data Management Center (TDMC). -- TDMC was established in July 1976 to serve NRC programs with goals and priorities determined by an advisory committee of NRC staff members chaired by the contract monitor within the Division of Technical Information and Document Control of the NRC Office of Administration. Its primary purpose is to support NRC programs by assisting in verification, validation, documentation, and standardization of computing technology (including codes and other technical data) used in the licensing and regulatory processes.

Current work includes the following subtasks: (A) technical data management in "open code/data package" concept; (B) program enhancement, development of new options and extensions,

standardization, and documentation of selected computing technology; (C) technical data generation, validation, and documentation; and (D) publications in support of TDMC tasks and other NRC Technical Data Management Programs.

Subtask A treats nonshielding methods in the SCALE (Standard Computer Analysis for Licensing Evaluation) system. Work was performed in three areas: (1) testing, packaging, dissemination with a mechanism for feedback, and maintenance of stand-alone computer codes associated with SCALE; (2) the processing of Release 2 of the SCALE system; and (3) providing information resulting from applications using TDMC-treated codes and data for NRC technical monitors.

Subtask B has included treatment of meteorological (atmospheric) and other environmental transport, and radiological assessment computing technology. These include XOQDOQ (routine effluent releases at nuclear power stations); LPGS (liquid pathway generic study of the consequences of accidental radioactive releases to the hydrosphere from floating and land-based nuclear power plants) and its companion radiological information database system, GENDOS; and RABGAD (atmospheric dispersion modeling). Several reports are in preparation describing this work.

Subtask C includes calculations to generate data for the Handbook of Gamma-Ray Attenuation Coefficients and Buildup Factors for Engineering Materials produced in collaboration with the ANS-6 Standards Committee. Work continues on compiling and analyzing gamma-ray transport data for the proposed ANS-6.4.3 standard. Attention has been concentrated on verifying moments data for water, iron, and lead. Results of ASFIT and Monte Carlo calculations have been compared with the moments data and various features have been investigated. Work is in progress to obtain discrete ordinates calculations. Attention has been given to the effects of secondary radiation (fluorescence and bremsstrahlung), and a paper on this subject is soon to appear in *Nuclear Science and Engineering*. The behavior of the dose rate at a shield-tissue interface has also been investigated. A paper on this subject was given at the June 1982 ANS meeting. Absorption coefficient data have been obtained from the National Bureau of Standards. These data have been used to prepare the required tables for the standard. Work on this project is proceeding cooperatively at Oak Ridge, Los Alamos, General Atomic, and in India.

Center of Model Evaluation Technology (COMET). -- COMET was established as an information exchange medium between the various elements of the DOE Energy Information Administration (EIA) model and data evaluation program. There were two parts to the COMET operation: it supported the ORNL Model Evaluation Team and it supported the Institution's Model Testing Program.

COMET maintained a reference library consisting of literature relating to energy/econometric models; the entire collection was entered into a computerized data base maintained on the Engineering Physics Information Centers' minicomputer. COMET collected, tested, and packaged models and offered them to those working in its field. It also maintained a computerized address directory containing the names of individuals working in the energy/econometric area.

COMET's last major action, carried out at EIA request, was the creation of a road map through the documentation which related to LEAP, a model which had been extensively examined by the Model Evaluation Team. Much of the documentation for this model was written at ORNL. Rather than create a single document to be offered to the user of LEAP, a chart was created, listing what was thought should be included if a single document, or set of documents, were ever published.

When the COMET operation was phased out (end of FY 1981), there were nearly 900 items in the library, well over 1400 people listed in the directory, and 18 tested computer code (model) packages. This computing technology was transferred to the National Energy Software Center (NESC) at Argonne National Laboratory and the other material is currently archived within the Division.

Reactor Data Bases: CREDO and SACRD

As an advanced reactor reliability data base, CREDO's primary functions are to collect, evaluate, and disseminate detailed engineering, event, and operating data on advanced reactor components. In addition, information pertaining to component maintenance, plant availability and human interaction data is compiled. To date, its two primary data sources have been the Fast Flux Test Facility and the Experimental Breeder Reactor II, but it has and will continue to use other sources as they become available. CREDO's intent is to provide all the functions usually associated with data bases, including data analysis services upon request.

SACRD continues as a data base on materials properties and other handbook information needed for computer codes used in fast reactor safety studies. Several areas, such as thermodynamics, heat transfer, etc., are covered.

CDIC: Carbon Dioxide Information Center

CDIC's purpose is to support the nation's CO₂-climate research effort by providing a focal point for the compilation and distribution of CO₂-related information under systematic quality control. One of its primary functions is to serve as a networking mechanism between the user community and the large number of established national and international data centers containing data relevant to the CO₂-climate issue (such as atmospheric CO₂ measurements, fossil fuel utilization, forest conversion, characterization of ocean properties, historical records from ice cores, tree rings, etc.). CDIC will provide details on content, accessibility, format, and usefulness of the needed data sets.

While this locator function will serve users' needs for experimental data, a major activity of CDIC is to provide a clearinghouse for evaluated information. The intent of the evaluation process is to provide a peer-reviewed documented reference data set to which others may be compared. The evaluated file should ultimately improve with time as better data are found. Final data sets should have a quantitative estimate of the uncertainty in the data and traceability to the primary sources of the information. The data sets will be stored in a computer-searchable format and will be disseminated with supporting documentation.

A specialized bibliographic data base on CO₂ and climate, in development for several years and currently containing over 3000 citations, is maintained. Emphasis has been placed on studies relating to the carbon cycle, atmospheric modeling, agriculture and forestry, energy technologies, and historical records relevant to reconstruction of past climates. This data base is used to support CDIC internal programs and request/response activities.

In addition to literature and numeric data, CDIC will collect, organize, evaluate, package and disseminate computing technology (algorithms, codes, models) as required for a fully integrated information analysis center program needed in support of a global assessment of the CO₂ problem.

8.1. USER EXPERIENCE AND COMPATIBILITY IN DOCUMENTATION STANDARDS*

B. F. Maskewitz

(Abstract of a paper presented at the National Bureau of Standards FIPS Software Documentation Workshop, Gaithersburg, MD, March 3, 1982)

This paper reviews existing guidelines for documentation of scientific computer programs or data libraries and outlines the essential elements for facilitating exchange of the software. Selected case studies will be made in which accepted standards were followed from the programming stage through documentation, and an analysis of user experience.

*Work sponsored jointly by U.S. Nuclear Regulatory Commission, Defense Nuclear Agency, and U.S. DOE Office of Energy Technology.

8.2. RECOMMENDED PROGRAMMING PRACTICES TO FACILITATE THE PORTABILITY OF SCIENTIFIC COMPUTER PROGRAMS*

Working Group ANS-10.2[†]

L. I. Kopp L. L. Barinka M. L. Couchman
B. W. Kirschner S. J. Natanson J. E. Olhoeft
E. J. Pastor G. R. Poetschat D. R. Vondy

(Abstract of ANSI/ANS-10.2-1982, published by American Nuclear Society)

This standard recommends guidelines to help the programmer create a program that is easily portable. These guidelines also provide good programming practices which are useful even when writing programs that are not expected to be transmitted out of the author's organizational unit. Such practices can reduce the effort required for the inevitable, and not infrequent, change to a different computer system or transfer of personnel responsibilities within the same organization. For related guidelines on documen-

tation and user needs, refer to the following American National Standards: Guidelines for the Documentation of Digital Computer Programs, N413-1974 (ANS-10.3) (Revision of ANS Std. 2), [2]; and Guidelines for Considering User Needs in Computer Program Development, ANSI/ANS-10.5-1979.[3].

*ORNL participation supported by U.S. Department of Energy.

[†]Members of Working Group represent the following organizations: U.S. Nuclear Regulatory Commission; Interactive Computer Modelling, Inc.; Systems Control, Inc.; Yankee Atomic Electric Company; NUS Corporation; Westinghouse Electric Corporation; Digital Equipment Corporation; G. R. P. Consulting, Inc.; and Oak Ridge National Laboratory.

8.3. ANS SHIELDING STANDARDS FOR LIGHT WATER REACTORS*

D. K. Trubey

(Abstract of paper presented at the Fourth ASTM - EURATOM Symposium on Reactor Dosimetry, National Bureau of Standards, Washington, D. C., March 22-26, 1982)

The purpose of the American Nuclear Society Standards Subcommittee ANS-6, Radiation Protection and Shielding, is to develop standards for radiation protection and shield design, to provide shielding information to other standards-writing groups, and to develop standard reference shielding data and test problems. A total of seven published ANS-6 standards are now current. Additional projects of the subcommittee, now composed of nine working groups, include: standard reference data for multigroup cross sections, gamma-ray absorption coefficients and buildup factors, additional benchmark problems for shielding problems and energy spectrum unfolding, power plant zoning design for normal and accident conditions, process radiation monitors, and design for postaccident radiological conditions.

*Work jointly supported by U.S. Nuclear Regulatory Commission and U.S. Department of Energy.

8.4. SPECIFIC GAMMA-RAY DOSE CONSTANTS FOR NUCLIDES IMPORTANT TO DOSIMETRY AND RADIOLOGICAL ASSESSMENT*

D. K. Trubey L. M. Unger[†]

(Abstract of paper presented at the Health Physics Society Annual Meeting, Louisville, KY, June 21-26, 1981; also abstract of ORNL/RSIC-45, September 1981, and of ORNL/RSIC-45/RI, May 1982)

Tables of specific gamma-ray dose constants (the unshielded gamma-ray dose equivalent rate at 1 m from a point source) in SI units have been computed for approximately 500 nuclides important to dosimetry and radiological assessment. The half lives, gamma-ray energies and emission probabilities, mean attenuation coefficients, and thicknesses for a lead shield providing 95% dose equivalent attenuation are also listed. The gamma-ray energy and emission probability data were taken from a new compilation (D. C. Kocher, "Radioactive Decay Data Tables: A Handbook of Decay Data for Application to Radiation Dosimetry and Radiological Assessments," U.S. Department of Energy [DOE/TIC-11026]) and are available in computer-readable form from the ORNL Radiation Shielding Information Center as DRALIST/DLC-80. The computation of dose rate was based on flux-to-dose-rate factors given in an American National Standard ("Neutron and Gamma-Ray Flux-to-Dose-Rate Factors," ANSI/ANS-6.1.1, American Nuclear Society, La Grange Park, IL 60525, 1977). The attenuation by a lead shield was computed from published coefficients (J. H. Hubbell, "Photon Cross Sections, Attenuation Coefficients, and Energy Absorption Coefficients from 10 keV to 100 GeV," NSRDS-NBS 29, National Bureau of Standards, 1969). The full results are published elsewhere (Laurie M. Unger and D. K. Trubey, "Specific Gamma-Ray Constants for Nuclides Important to Dosimetry and Radiological Assessments," ORNL/RSIC-45, 1981).

*Research sponsored by U.S. Nuclear Regulatory Commission Office of Administration.

[†]Heidelberg College, Tiffin, Ohio.

8.5. GAMMA-RAY DOSE AT SHIELD-TISSUE INTERFACES AND BUILDUP FACTOR IMPLICATIONS*

D. K. Trubey D. V. Gopinath[†] K. V. Subbaiah[†]

(Reprint of Trans. Am. Nucl. Soc. 41, 494, 1982)

Buildup factors continue to be widely used for gamma-ray shield design, but there are several problems in application. Shure¹ has pointed out the need for buildup factors to be consistent with the ANSI standard flux-to-dose factors.² However, consistent buildup factor data cannot be obtained by simply integrating the ANSI response function over infinite-medium spectra, because this function gives the response at various depths in a tissue phantom for a gamma ray of a certain energy entering the phantom. The ANSI response function can only be applied to a spectrum emerging from a shield as determined from a transport calculation.

In order to evaluate the dose in tissue from gamma rays which have penetrated a shield, a detailed transport study was undertaken at the Safety Research Laboratory at Kalpakkam in collaboration with the Radiation Shielding Information Center. Computations were made using the one-dimensional transport code ASFIT³ which has recently been extended to treat the secondary sources of fluorescence and bremsstrahlung.⁴ The code uses a nodal structure in space and energy and is well suited for the study of interface phenomena. Shield materials studied were water, concrete, iron, and lead with thicknesses of 1-20 mfp. In all cases, computations were made for these shields and the shields followed by a 30-cm-thick standard tissue layer as in Ref. 2. The energy range studied was 0.05 to 5 MeV.

In all shield-tissue cases, the low energy part of the spectrum at the interface is very much enhanced compared to the emergent spectrum from the base shield, and in the case of water, the interface spectrum is close to that of an infinite water medium. Similarly, the tissue-response buildup factors are also close to the infinite-medium buildup factors. This phenomenon is

observed generally for all materials with low atomic number. However, with increasing atomic number of the shield material, significant changes appear. The spectra at the interface are quite different from the transmitted or infinite-medium spectra as shown in Fig. 8.5.1. For energies up to 0.2 MeV, the maximum dose in the tissue does not occur at the interface but at depths varying from 0.5 to 1.5 cm inside the tissue. The interface buildup factors are also significantly higher compared to finite or infinite-medium buildup factors.

Fig. 8.5.1. Energy Spectrum at Lead-Tissue Interface (10-mfp-Thick Lead, 0.2-MeV Source).

In the cases where the maximum dose in the tissue is not at the interface, the interface buildup factors need to be corrected by the ratio of the maximum dose in the tissue to the interface dose to obtain the correct tissue-response buildup factors, and this correction further enhances the difference from the infinite-medium case. For instance, in the case of 5-mfp-thick iron and 0.1 MeV source energy, the infinite medium, finite medium, interface, and corrected buildup factors are 1.74, 1.61, 2.75, and 2.90. Similarly, for 10 mfp of lead and 0.2 MeV incident gamma rays, they are 1.51, 1.41, 2.14, and 2.16. Additional results are given in Table 8.5.1.

Table 8.5.1. Plane Collimated Buildup Factors for Lead

E (MeV)	Thickness (mfp)	Buildup Factor ^a			
		1*	2	3	4
0.05	1	1.02	1.01	1.69	1.74
	2	1.02	1.01	1.70	1.75
	5	1.03	1.02	1.71	1.84
	10	1.03	1.03	1.71	1.77
	20	1.04	1.04	1.74	1.78
0.20	1	1.23	1.16	1.74	1.76
	2	1.26	1.20	1.80	1.83
	5	1.36	1.29	1.94	1.98
	10	1.51	1.41	2.14	2.16
	20	1.81	1.64	2.47	2.48
1.0	1	1.38	1.36	1.53	1.53
	2	1.65	1.63	1.84	1.84
	5	2.29	2.24	2.56	2.56
	10	3.06	3.01	3.46	3.46
	20	4.18	4.09	4.75	4.75
3.0	1	1.49	1.40	1.51	1.51
	2	1.85	1.76	1.89	1.89
	5	2.95	2.78	3.05	3.05
	10	4.80	4.56	5.03	5.03
	20	8.68	8.15	9.16	9.16

^a 1 infinite medium, 2 finite medium, 3 interface, 4 corrected.

It is concluded that, for the range of shields and parameters studied, a set of correction factors can be determined to convert infinite-medium, single-collision buildup factors to those consistent with the tissue phantom approach taken by the ANSI standard. The maximum correction is nearly a factor of 2 for a 20-mfp-thick lead shield at low energies, but is only about 5% for 3-5 MeV. Thus in some applications the correction is negligible, but sometimes low energy gamma rays are important, e.g. the gamma ray from ¹³³Xe is 80 keV, and this is an important nuclide in radiation assessment.

*Research sponsored by Office of Administration, Nuclear Regulatory Commission

†Safety Research Laboratory, Reactor Research Centre, Kalpakka, India.

1. K. Shure, *Nucl. Sci. Eng.* **69**, 432 (1979).
2. M. E. Battat, "Neutron and Gamma-Ray Flux-to-Dose-Rate Factors," ANSI/ANS-6.1.1-1977, American Nuclear Society (1977).
3. D. V. Gopinath and K. Santhanam, *Nucl. Sci. Eng.* **43**, 186 (1971).
4. K. V. Subhaiah, A. Natarajan, D. V. Gopinath, and D. K. Trubey, *Nucl. Sci. Eng.* **81**, 172 (1982).

8.6. EFFECT OF FLUORESCENCE, BREMSSTRAHLUNG, AND ANNIHILATION RADIATION ON THE SPECTRA AND ENERGY DEPOSITION OF GAMMA RAYS IN BULK MEDIA*

K. V. Subbaiah[†] A. Natarajan[†]
D. V. Gopinath[†] D. K. Trubey

[Abstract of Nucl. Sci. Eng. 81(2), 172 (1982)]

The contribution of bremsstrahlung, annihilation, and fluorescence radiation to the spectra gamma radiation and to the dose buildup factors has been investigated as a function of source energy, atomic number, and sample thickness over an energy range of 0.1 to 8 MeV. The computations were performed with the one-dimensional transport code ASFIT modified to take into account all the secondary radiations. The required mathematical formulation along with representative results obtained for uranium, lead, iron, and water, typifying materials of very high, high, medium, and low atomic number, are presented and discussed.

A noticeable effect of including bremsstrahlung sources is the general softening of the scattered radiation spectra inside the medium and at the exit. This effect is more pronounced in high atomic number. The bremsstrahlung contribution is seen most prominently in the reflection spectra above 0.511 MeV, where the contribution from other processes is insignificant. The effect of annihilation radiation is significant in the region between 0.511 MeV and the K edge, below which the effects of fluorescence radiation overshadow all others. Peaks and discontinuities characteristic of single-scatterings of these radiations are seen in the reflection spectra, gradually disappearing with depth in the medium.

The effect of fluorescence on the dose buildup factor is spectacular for source energies close to the K edge and falls off rapidly thereafter. The impact of bremsstrahlung, on the other hand, steadily rises with source energy. The influence of annihilation radiation is comparatively modest and is significant only for systems of intermediate atomic numbers.

*Research sponsored by U.S. Nuclear Regulatory Commission.

[†]Safety Research Laboratory, Kalpakkam, India.

8.7. BOOK REVIEW: STRUCTURE SHIELDING AGAINST FALLOUT GAMMA RAYS FROM NUCLEAR DETONATIONS*

D. K. Trubey

[Excerpts from Nucl. Sci. Eng. 79(4), 434 (1981)]

It is only occasionally that an exhaustive treatise on a shielding subject is published. We have one now in the nearly 1000-page book *Structure Shielding Against Fallout Gamma Rays From Nuclear Detonations* by L. V. Spencer, A. B. Chilton, and C. M. Eisenhauer.¹

All of the authors have been leading researchers in shielding, especially fallout shielding, for many years. Spencer and Eisenhauer, especially, are well known for their development and application of the moments method for gamma-ray penetration and, with N. FitzSimons, of the so called "Standard Method" for fallout shielding analysis developed in the 1960s. Although very few people are engaged these days in fallout shielding analysis, the motivation for it has not entirely disappeared, considering the nuclear arsenals in various parts of the world. There is the possibility, too, of some spinoff -- the technology can also be used for analysis of reactor accident fallout.

This massive work is far more than a state-of-the-art report; it is a textbook on gamma-ray transport, the moments method of solving the Boltzmann transport equation, fallout gamma-ray sources, fallout phenomena, shielding concepts, penetration data, field experiments, and shielding analysis (especially the "Standard Method"). This book was obviously prepared with great care. I noticed only one misspelled name, and there were very few typographical errors. The authors also were very careful to distinguish between flux density and current, thus avoiding a problem which is frequently encountered in such discussions. Each section has a long list of references, mostly from the 1950-1978 period. There is an author index and a subject index, both of which seem to be quite thorough.

This book is a must for any engineer in fallout shielding and would be a valuable reference to anyone interested in gamma-ray shielding applications.

*Work sponsored jointly by U.S. Nuclear Regulatory Commission and the Defense Nuclear Agency.

8.8. A SHORT COURSE IN THE OPERATION OF INTERACT*

M. B. Emmett[†] W. A. Rhoades

(Abstract of ORNL/CSD/TM-187, in press)

INTERACT, an online computer system for text entry, editing, and remote job entry, has a great deal of potential for rapid, simple and effective support of applications users. As is frequently the case with documentation, the descriptive material available is not adequate to explain the system to the beginner. In an effort to remedy this problem, this document is being published to acquaint users with a simple subset of INTERACT that will enable them to use the more important features in a relatively short time.

*Research sponsored by U.S. DOE Division of Reactor Research and Technology.

[†]UCC-ND Computer Sciences Division.

8.9. THE CENTRALIZED RELIABILITY DATA ORGANIZATION (CREDO): A DATA AND INFORMATION BANK FOR ADVANCED REACTORS*

H. E. Knee

(Reprint of Trans. Am. Nucl. Soc. 38, 492, 1981)

The Centralized Reliability Data Organization (CREDO) is an advanced reactor reliability/availability data base. It was established at Oak Ridge National Laboratory by the Reactor Research and Technology Division of the U.S. Department of Energy.

CREDO's primary functions are the collection, evaluation, and dissemination of detailed engineering, event, and operating data on advanced reactor components. In addition, information pertaining to component maintenance, plant availability, and human interaction is compiled. CREDO will provide all of the functions

usually associated with data bases. It will also provide data analysis services to its users on request.

CREDO's two primary data sources are the Fast Flux Test Facility in Richland, Wash., and the Experimental Breeder Reactor II (EBR-II) in Idaho Falls, Ida. To take advantage of all possible sources of advanced reactor component data, CREDO has and will collect applicable data at various existing U.S. test loops.

As an advanced reactor reliability data base, CREDO is the focal point for the assimilation of these data in the United States. In addition, CREDO is actively pursuing the establishment of a cooperative data exchange program with countries outside the United States, in particular, the Dounreay and Prototype Fast Reactors in the United Kingdom.

The heart of the CREDO system is its sophisticated and versatile data base management system (DBMS). The system consists of various searching and sorting subroutines from ORCHIS (the Oak Ridge Computerized Hierarchical Information System), plus a driving program that includes specialized software for the particular needs of CREDO operating under a DBMS developed from the JOSHUA system (developed at Savannah River Laboratory). The JOSHUA/ORCHIS combination provides a flexible and efficient data management system that provides the capability for performing the varied tasks of a reliability data center. These include cataloging, storage, searching, retrieval, routine statistical analysis, and routine output generation. The flexibility of CREDO's DBMS allows highly detailed, significantly different file structures to be efficiently related for searching purposes.

CREDO provides various types of standardized outputs. Qualitative information is provided through several types of data inventories and summaries. In addition, the raw data are available to the user if the reduced data in CREDO's standardized qualitative outputs do not contain the desired depth of detail. Quantitative information and data are provided on request to the user. Such data as a failure rate, mean time to repair, failure mode and failure cause trends, and population data are output from CREDO's failure rate

program. If additional in-depth statistical analysis is desired, the user may call on the services of CREDO's statistics code, CRESTCO. CRESTCO produces both analytic and graphical outputs. It allows the user to specify any or all of five different failure probability distribution functions and calculates the associated distribution parameters. If further data analysis is required by the user, the CREDO staff will provide data analysis services. Recent studies have included analysis of sodium valves, electromechanic pumps, and the hazards associated with sodium systems and EBR-II's experience with them.

The development and operation of the CREDO system has been done under the scrutiny of the CREDO steering committee. It is composed of nationally recognized experts in the various aspects of breeder reactor design, safety, and reliability. It provides general guidance for the

CREDO system and aids in the development of CREDO with respect to enhancing its user applicability.

The need for accurate component and system reliability/availability data and experience has long been realized by the nuclear industry. With the advanced reactor industry in the United States being relatively young, CREDO offers the opportunity to initiate a comprehensive program that can provide meaningful, accurate data for a developing technology. CREDO will supply a centralized source of evaluated reliability data and associated services necessary for the breeder reactor safety program and will meet the needs of research and development programs involving reliability/availability assessment.

*Research sponsored by U.S. DOE Office of Reactor Research and Technology.

APPENDIX

SCIENTIFIC AND PROFESSIONAL ACTIVITIES

November 1, 1980 - May 31, 1982

L. S. Abbott

Cochairman, American Nuclear Society Committee for 1982 World's Fair Exhibit (Exhibit Designer)
Exhibits Chairman, Oak Ridge/Knoxville Section of American Nuclear Society
Outstanding Member of the Year, Oak Ridge/Knoxville Section of American Nuclear Society (1981-1982)
Editor, *The Acorn*, newsletter published by Oak Ridge/Knoxville Section of American Nuclear Society
Associate Editor, *Risk Newsletter*, published by Society for Risk Analysis

R. G. Alsmiller, Jr.

Member, National Committee on Radiological Protection Scientific Committee 52 on "Conceptual Basis of the Calculations of Dose Distributions"
Member, Department of Energy Advisory Panel on Accelerator Radiation Safety
Chairman, Department of Energy Advisory Panel on Accelerator Safety for the Review of the 400-GeV Proton-Proton Colliding Beam Facility (ISABELLE) at Brookhaven National Laboratory
Invited lecture, "Adjoint Sensitivity Theory and Its Application to a Large Energy-Economics Model," Energy Information Administration Department of Energy, November 1981
Invited paper, "High- and Medium-Energy Accelerator Shielding," presented at the Mid-Year Symposium on Accelerator Health Physics of the Health Physics Society, Orlando, Florida, February 22-24, 1982
Invited paper, "Analysis of ORNL Radiation Streaming Integral Experiments," with R. T. Santoro et al., Annual Meeting of the American Nuclear Society, Miami Beach, Florida, June 7-12, 1981

J. Barhen

Member, Processing Codes Subcommittee, Cross-Section Evaluation Working Group (CSEWG)
Best Paper Award, American Nuclear Society, Reactor Physics Division, "On the ENDF/B Unresolved Resonance Region Formalism Representation," with R. B. Perez, G. de Saussure, J. L. Munoz-Cobos, and R. Q. Wright, November 1981
Invited paper, "Validation of Scientific and Technical Information," with R. G. Alsmiller, Jr. and C. R. Weisbin, American Society for Information Sciences Meeting, Knoxville, Tennessee, June 1982
Invited review article, "Resonance Shielding in Thermal Reactor Lattices," with W. Rothenstein, E. Taviv, and A. Aminpour, *Ann. Nucl. Eng.*, 9, 161 (1982)
Referee, Proposal Reviews, National Science Foundation
Referee, Proposal Reviews, DOE/Office of Basic Energy Sciences
Referee, *Nuclear Science and Engineering*

D. E. Bartine

Vice Chairman/Chairman-Elect, American Nuclear Society Radiation Protection and Shielding Division, June 1981-June 1982

Member, Defense Nuclear Agency INRAD Working Group Weapons and Environmental Subgroup (WESG)

Member, Technical Advisory Committee for Radiation Transport for the Ballistic Missile Defense (BMD)

Member, Evaluation Committee for Air Force Reactor Studies (MX)

Member, GCFR Reactor Physics and Shielding Working Group, US/FRG/Swiss Umbrella Agreement on Gas-Cooled Reactors (1977-1981)

Member, Election Inspection Committee for Knoxville-Oak Ridge Section of American Nuclear Society (1981-1982)

Member, American Nuclear Society Radiation Protection & Shielding Division *Ad Hoc* Committee for Promotion of the 6th ICRS

Certificate of Governance awarded by American Nuclear Society Board of Directors (1980)

Invited paper, "Shielding Considerations for Space Power Reactors," Special Conference on Prime Power for High-Energy Space Systems, Norfolk, Virginia, February 1982

Lecturer, Oak Ridge Associated Universities, course entitled, "Introduction to Reactors and Reactor Theory, July 1981

Local Section Representative, American Nuclear Society

Technical Coordinator, US/Japanese LMFBR Exchange Agreement (1982)

Participant, Industrial Research Institute Advanced Study Group - Topic III: R&D Interface with Marketing and Production (1982)

Participant, Carbide Management Training Program on Project Selection (Louis A. Allen Associates), May 1982

Participant, Fundamentals of Human Factors for Engineering and Design course, Anacapa Sciences, Inc., April 1981

Advisor, Nuclear Engineering Department, University of Missouri at Rolla, Evaluation of Long-Term Facility Requirements

T. J. Burns

Patent, "Gamma Thermometer Based Reactor Core Liquid Level Detector," July 1981 (Pending)

Technical Advisor to Assistant Secretary of State for Oceans, Environment, and Scientific Affairs, November 1980-June 1981, Off-site assignment

Participant, Application of Reliability and Risk Analysis, George Washington University, April 1981

Participant, Fundamentals of Human Factors for Engineering and Design course, Anacapa Sciences, Inc., April 1981

Member, Review Group for PRA Procedures Guide, October 1981

Participant, Systems Reliability and Risk Assessment, ORNL, January 1982

D. G. Cacuci

Invited lecturer, "Sensitivity Theory for Nonlinear Systems," Columbia University, 1982

Invited lecturer, "Sensitivity Analysis for Nonlinear Systems," Georgia Institute of Technology, 1982

Invited lecture, "Adjoint Method of Sensitivity Analysis: Illustrative Results for a Radiative-Convective Model," with M. C. G. Hall, National Science Foundation/National Academy of Sciences, 1982

Invited lecture, "Adjoint Operator Method of Sensitivity Analysis and Applications to a Radiative-Convective Climate Model," with M. C. G. Hall, Courant Institute of Mathematical Sciences, New York University, 1982

Invited lecture, "Adjoint-Operator Based Sensitivity Analysis of a Radiative-Convective Model of the Climate," with M. C. G. Hall, Institute for Energy Analysis 1981

Referee, *Journal of Mathematical Physics*

Referee, *Nuclear Science and Engineering*

S. N. Cramer

Participant in a program of data testing and code benchmarking at the Nuclear Energy

- Agency Data Bank at Saclay, France, as a NEA Data Bank Consultant (October 1981 - January 1983)

G. de Saussure

Chairman, International Meetings Subcommittee of National Program Committee, American Nuclear Society

Member, National Program Committee, American Nuclear Society

Member, Steering Subcommittee of National Program Committee, American Nuclear Society

Referee, *Nuclear Science and Engineering*

Referee, *The Physical Review*

Honorary Professor, Nuclear Engineering Department, U. of Tennessee

Member, Thesis Committee, Ph.D. Thesis, U. of Tennessee (C. Yang and B. Broadhead)

Invited papers, coauthor, at the ANS June 1981 National Meeting and at the IAEA Consultant Meeting on U and Pu Isotope Resonance Parameters

Chairman and Session Organizer, IAEA Consultants Meeting on Uranium and Plutonium Isotope Resonance Parameters, Vienna, September 28- October 2, 1981

Best Paper Award, American Nuclear Society, Reactor Physics Division "On the ENDF/B Unresolved Resonance Region Formalism Representation," with R. B. Perez, J. Barhen, J. L. Munoz-Cobos, and R. Q. Wright, 1981 Winter Meeting

J. K. Dickens

Member, American Nuclear Standards Committee 5; Working Group 5.1, Decay Heat Standard; Working Group 5.2, Fission Product Yields

Referee, *Nuclear Science and Engineering*

Referee, *The Physical Review*

Designated by IAEA Working Group on Fission Product Nuclear Data as the correspondent on world-wide experimental measurements of fission product decay heat (1977-continuing)

Colloquia: "Nuclear Power Reactor Safety," Duke University, Dept. of Physics and Astronomy (November 1981); "Nuclear Fission - New Ideas and New Experiments," U. of Kentucky, Dept. of Physics and Astronomy (March 1982)

Invited seminar, "Nuclear Power Reactor Safety," U. of Missouri, Dept. of Nuclear Engineering (February 1982)

Speaker's Bureau Presentations: "Nuclear Safety," Bnai Brith, Clarksdale, Mississippi (February 16, 1982); Kiwanis Club, Gatlinburg, Tennessee (October 20, 1981); Lions Club, Johnson City, Tennessee (October 29, 1981); Kiwanis Club, Maryville, Tennessee (December 8, 1981); Sertoma Club, Oak Ridge, Tennessee (February 16, 1982); Optimist Club, Carter Community, Knoxville, Tennessee (March 8, 1982)
Member, ORNL Graduate Fellow Selection Panel

G. F. Flanagan

Chairman, Reactor Safety Division Program Committee, American Nuclear Society
Member, Standards Committee 5.4, IEEE
Member, Standards Committee 19.3, American Nuclear Society
Member, Editorial Board, *Risk Analysis Journal*
Member, Technical Program Committee, ANS/ENS/CNS Thermal Reactor Safety Conference, Chicago, Illinois, August 1982
Registration Chairman, Low Probability/High Consequence Risk Analysis Workshop, Arlington, Virginia, June 15-17, 1982
Chairman, Oak Ridge/Knoxville Section American Nuclear Society
Editor, *Risk Newsletter*, published by Society for Risk Analysis
Referee, *Nuclear Safety Journal* and *Nuclear Technology*
Referee, *Risk Analysis Journal*
Member, Ph.D. Recruiting Program, ORNL
Member, ORNL Speaker's Bureau
Representative, ORNL Advisory Committee for Office of Risk Analysis
Member, Department of Energy Special Task Force to Solicit Industrial Response on Safety Research Requirements to Fulfill Public Law T6-567, "Nuclear Safety Research, Development and Demonstration Act of 1980"
Assistant Professor, University of Tennessee, "Fast Reactor Safety"

C. Y. Fu

Member, Evaluation Methods Subcommittee, Cross Section Evaluation Working Group (CSEWG)
Invited paper, "Experience in Using the Covariances of Some ENDF/B-V Cross Sections: Proposed Improvements and Addition of Cross-Reaction Covariances," Fourth ASTM-EURATOM Symposium on Reactor Dosimetry, Gaithersburg, Maryland, March 22-26, 1982
Invited lecture, "Consistent Calculation of Angular Distributions of Secondary Neutrons from Compound and Precompound Reactions," Lawrence Livermore National Laboratory, February 1981
Referee, *Nuclear Science and Engineering*

T. A. Gabriel

Member, Four-Laboratory Group (SLAC, DESY, CERN, ORNL) which is developing a calculational and experimental data base for shielding electron accelerators
Advisor for SLAC, ANL, and BNL on high-energy transport calculations for shielding
Consultant to the Nuclear Research Center, Karlsruhe, W. Germany
Invited lecture, "The Proposed ORNL, Harvard, U.T. Neutron-Antineutron Oscillation Experiment," T. A. Gabriel et al., presented at the Workshop on Neutron-Antineutron Oscillations, Harvard University, Cambridge, Massachusetts, April 30-May 1, 1982

Invited lecture, "Shielding and Detector Analysis in Support of Neutrino Physics at an Intense Spallation Neutron Source," T. A. Gabriel and R. A. Lillie, presented at the Nuclear Research Center, Jülich, W. Germany, April 2, 1981

L. H. Gray

Chairman, Nominations Committee, Intermountain Chapter, Human Factors Society, 1981

Invited seminar, "System Engineering Approach to Systems Design, and the Personnel Subsystem," Idaho State University, Idaho Falls, June 1981

R. Gwin

Member, ORNL Criticality Committee, 1981

P. M. Haas

Member, IEEE NPEC Subcommittee SC-7, Human Factors and Control Facilities

Member, IEEE Nuclear Power Engineering Committee, SC-5, Reliability

Member, IEEE Working Group on Human Factors, SC-5.5

Member, Journals, Periodicals, and Critical Reviews Committee, American Nuclear Society (December 1980 - June 1982)

Section Editor, *Nuclear Safety*, Accident Analysis Section

Invited paper, Ninth Water Reactor Safety Research Information Meeting, October 1981

M. C. G. Hall

Invited lecture, "Adjoint-Operator Based Sensitivity Analysis of a Radiative-Convective Model of the Climate," with D. G. Cacuci, Institute for Energy Analysis, 1981

Invited lecture, "Adjoint Method of Sensitivity Analysis; Illustrative Results for a Radiative Convective Model," with D. G. Cacuci, National Science Foundation/National Academy of Sciences, 1982

Invited lecture, "Adjoint Operator Method of Sensitivity Analysis and Applications to a Radiative-Convective Climate Model," with D. G. Cacuci, Courant Institute of Mathematical Sciences, New York University, 1982

H. R. Hendrickson

Data Coordinator for the Nuclear Data Section of IAEA, Vienna, Austria, August 1979 - December 1981

Invited official visitor to shielding research groups, Brno Technical University, Brno, Czechoslovakia (October 23, 1981) and Prague Technical University, Prague, Czechoslovakia (October 26, 1981)

D. T. Ingersoll

Member, Program Committee, Radiation Protection and Shielding Division, American Nuclear Society

Member, Shielding Data Testing Subcommittee, Cross-Section Evaluation Working Group (CSEWG)

Member, ORNL Task Group on Computer User's Priorities

Member, Subcommittee of the Reactor Operations Review Committee

H. C. Kae

Member, Centralized Reliability Data Organization (CREDO) Steering Committee
 Member, Program Committee, Nuclear Reactor Safety, American Nuclear Society
 Member, CREDO Data Evaluation Working Group and CKEDO Input/Output Working Group
 Section Editor, *Nuclear Safety*, Accident Analysis Section
 Invited paper, "The Nuclear Power Plant Maintenance Personnel Reliability Prediction (NPP/MPRP) Effort at Oak Ridge National Laboratory," presented at the September 20-24, 1981 International ANS/ENS Topical Meeting on Probabilistic Risk Assessment, Port Chester, NY

D. C. Larson

Cochairman, General Purpose Evaluations Subcommittee, Cross Section Evaluation Working Group (CSEWG)
 Member, Evaluation Methods Subcommittee, CSEWG
 Member, ORNL Proposal Review Committee (Seed Money)
 Member, ORNL Graduate Fellow Selection Panel
 Reviewer, *Nuclear Science and Engineering*
 Proposal reviewer, Office of Basic Energy Sciences
 Member, Office of Basic Energy Sciences (OBES) Committee on Nuclear Data for Fusion Reactor Applications

R. A. Little

Member, ORNL Reactor Experiments Review Committee
 Chairman, Periodic ORR and BSR Experiments Review Subcommittee
 Invited lecture, "Discrete Ordinates Methods and Applications," Tennessee Industrial Week course, U. of Tennessee, September 1981

R. W. Maerker

Cochairman, Neutron and Gamma-Ray Transport Workshop at 4th ASTM- EURATOM Reactor Dosimetry Conference, NBS, March 22-26, 1982
 Member, ASTM E10.05 Subcommittee, 1978-1982
 Member, Shielding Subcommittee, Cross Section Evaluation Working Group (CSEWG)
 Invited lecture, "The LEPRICON Methodology," presented at U. of Missouri, Rolla, April 22, 1982
 Invited papers, "Calculations of the Westinghouse Perturbation Experiment at the Poolside Facility," and "Calculations of the Startup Experiments at the Poolside Facility," both with M. L. Williams, and "Surveillance Dosimetry: Achievements and Disappointments," with "Surveillance Dosimetry: Achievements and Disappointments," with J. J. Wagschal and B. L. Broadhead, given at the 4th ASTM-EURATOM International Conference on Reactor Dosimetry at NBS, March 22-26, 1982
 Reviewer, *Nuclear Science and Engineering* and *Nuclear Engineering*

F. C. Malenach

Member, Nuclear Energy Agency Committee on Reactor Physics (NEACRP)
 Member, ORNL Awards Advisory Committee
 Chairman, ORNL Computing Steering Committee (OCSC)

Member, Department of Energy's LMFBR Core Working Group responsible for review and evaluation of the LMFBR core physics exchanges with foreign organizations, primarily with the United Kingdom, France, Federal Republic of Germany, and Japan

J. H. Marable

Member, ORNL Criticality Committee

Member, Data Testing Subcommittee, Cross Section Evaluation Working Group (CSEWG)

Member, Sensitivity and Uncertainty Analysis User's Group

B. F. Maskewitz

Member, American Nuclear Society Board of Directors (180-1983)

Member, American Nuclear Society Executive Committee (1981-82, 1982-83)

Member, American Nuclear Society Planning Committee (1979-82, 1982-84)

Fellow, American Nuclear Society, June 1982

Chairman, American Nuclear Society, Radiation Physics and Shielding Ad Hoc Committee for the Sixth International Conference on Radiation Shielding to be held in Tokyo, Japan, May 1983

Chairman, ANS 10.3 Subcommittee on Software Documentation Standards

Member, ANS-10 Subcommittee of the ANS Standards Committee

Member, Program Committee for 1982 Conference of the American Society for Information Science

Referee, Computer Code Abstracts, *Nuclear Science and Engineering*

Referee, Information Science Papers, 1981-1982, National Computer Conference

Speakers' Bureau Presentations: Energy Forum, Ontario Branch of Women's League for Conservative Judaism, Toronto, Canada, May 1982; Committee for Energy Awareness, Hershey, Pennsylvania, April 1982

Invited paper, "User Experience and Compatibility in Documentation Standards," Federal Information Processing Society (FIPS), National Bureau of Standards, March 3, 1982

Invited paper, "The Role of the Information Analysis Center in Quality Assurance," American Society of Information Services, Knoxville, Tennessee, June 13, 1982

Member, ORNL Honors and Awards Advisory Committee

Member, Biogeochemical Cycles and Climate Impact Assessment Working Groups of ICSU/UNEP Scientific Committee on Problems of the Environment (SCOPE) (1982)

D. L. Moses

Reviewer, *Nuclear Science and Engineering*

Reviewer, *Nuclear Technology*

D. K. Olsen

Chairman, Committee on Proton Accelerator for ORELA Replacement

Referee, *Nuclear Science and Engineering*, proposal reviews, NSF

Invited paper, "Review of (n,n' γ) Cross-Section Data on Actinide Nuclei," OECD Special-1st Meeting on Fast Neutron Scattering on Actinide Nuclei, Paris, France, 1981

Invited lectures, "ORELA Thorium Cycle Cross Section Measurements,"

CBNM Geel, November 20, 1981, and AERE Harwell, November 30, 1981

R. W. Peelle

Chairman, Cross Section Evaluation Working Group (CSEWG) Data Uncertainty and Correlation Subcommittee
 Member, Technical Program Committee, Sixth International Conference on Radiation Shielding, May 16-20, 1983
 Chairman, Workshop for Fissile and Fertile Nuclides, April 1982 NEANDC Specialists' Meeting on Neutron Capture Cross Sections, Argonne National Laboratory
 Referee, *The Physical Review, Nuclear Science and Engineering*

F. G. Perey

U. S. Member, Nuclear Energy Agency Nuclear Data Committee (NEANDC)
 Chairman, Neutron Standards Subcommittee, NEANDC
 Member, Department of Energy Nuclear Data Committee
 Member, INDC Standards Committee

R. B. Perez

Member, Program Committee, Reactor Physics Division, American Nuclear Society
 Best Paper Award, American Nuclear Society, Reactor Physics Division, "On the ENDF/B Unresolved Resonance Region Formalism Representation," with G. de Saussure, J. L. Munoz-Cobos, J. Barhen, and R. Q. Wright, November 1981
 Invited paper, "Representation of the Neutron Cross Sections of Several Fertile and Fissile Nuclei in the Resonance Regions," Spring Meeting of the American Nuclear Society, Miami, Florida, with G. de Saussure (June 1981)
 Professor, Nuclear Engineering Department, University of Tennessee
 Referee, *Nuclear Science and Engineering*
 Referee, National Science Foundation

W. A. Rhodes

Award for Technical Excellence, American Nuclear Society, Radiation Protection and Shielding Division, June 1981

R. W. Rossin

Member, Panel on Reference Nuclear Data, representative for American Nuclear Society, Radiation Protection and Shielding Division
 Member, ANS 6.2.2, Shielding Benchmark Subcommittee
 Member, ANS 6.1.2, Shielding Cross Sections Committee
 Chairman, Shielding Testing and Applications Subcommittee, Cross Section Evaluation Working Group (CSEWG)

R. T. Santoro

Member, Executive Committee, Radiation Protection and Shielding Division, American Nuclear Society
 Member, Honors and Awards Committee, American Nuclear Society, Fusion Energy Division
 Technical Advisor, TFTR Steering Committee, Princeton Plasma Physics Laboratory; Tritium Breeding Experiments, Princeton Plasma Physics Laboratory
 Invited lecture, "Integral Experiment Program at ORNL," presented to Japanese Nuclear Society, Tokyo, Japan, December 1980

D. L. Selby

Member, Executive Board, American Nuclear Society, Local Section
 Member, WATtec Sponsors Committee
 Program Chairman, ANS session of WATtec Conference, 1982
 Member, ANS Speaker's Bureau
 Invited presentation, American Nuclear Society, "Energy and Public Understanding Workshop," Los Angeles, California, June 9, 1982
 ORNL Speaker's Bureau presentation, Greeneville High School

C. O. Slater

Member, Program Committee, American Nuclear Society, Radiation Protection and Shielding Division

R. R. Spencer

Member, ^{252}Cf π Workshop of Cross Section Evaluation Working Group (CSEWG)
 Referee, *Nuclear Science and Engineering*
 Referee, *The Physical Review*

D. K. Trubey

ANS representative to ANSI Committee N-13, Radiation Protection
 Chairman, ANS-6, Radiation Protection and Shielding Subcommittee, Standards Committee, American Nuclear Society
 Chairman, ANS-6.4.3 Standards Working Group, Gamma-Ray Attenuation Data, ANS-6.5 Shielding Glossary, American Nuclear Society
 Member, ANS-6.5 Shielding Glossary, American Nuclear Society
 Member, Working Group on Code Interfaces and Formats, Department of Energy Safety Analysis Data Coordinating Group
 Lecturer, NRC-sponsored Applied Health Physics Course conducted by ORAU, May 1981

D. R. Vondy

Member, Mathematics and Computation Division Program Committee, American Nuclear Society
 Member, M&C Benchmark Committee, American Nuclear Society
 Member, ANS-10 Standards Committee, American Nuclear Society
 Alternate Member, ANSI Standards Committee, X-3
 Alternate Member, Fortran Committee, X3J3
 Member, Department of Energy, Office of Reactor Development and Technology, Physics and Safety Section, Large Core Methods Evaluation Committee and Cooperative Computer Coding Committee
 Lecturer, Tennessee Industries Week, "Computational Methods in Nuclear Reactor Analysis," University of Tennessee

C. R. Weisbin

Member, U. S. Department of Energy Assessment Team on CO₂ and Climate Director, Carbon Dioxide Information Center
 Member, 1982 Reactor Physics Program Committee, American Nuclear Society
 Member, TVA/ORNL Task Force; representation for Fuel and Core Design
 Chairman, Data Testing and Applications Committee, Cross Section Evaluation Working Group (CSEWC)

Member, Physics of Reactor Design Standards Committee, ANS-19
 Member, Codes and Formats, Fission Products and Shielding, CSEWG
 Invited lecture, "Sensitivity and Uncertainty Analysis for Reactor Physics in the Reactor Safety and Quality Engineering Laboratory," presented at The Technion, Haifa, Israel, December 30, 1981
 Invited lecture, "Sensitivity and Uncertainty Analysis in Reactor Physics," presented at Hebrew University, Jerusalem, Israel, January 3, 1982

L. W. Weston

Chairman, General Purpose Evaluation Subcommittee, Cross Section Evaluation Working Group (CSEWG)
 Invited paper, "Review of Fast-Neutron Capture Cross Sections of the Higher Plutonium Isotopes and ^{241}Am ," NEAND/NEACRP Specialists' Meeting on Fast-Neutron Capture Cross Sections, Argonne National Laboratory, April 1982
 Invited paper, "Error Assessment from White Source Measurements," Workshop on Covariance Methods and Their Application to General-Purpose Evaluations (CSEWG), Oak Ridge, May 1982
 Referee, *Nuclear Science and Engineering*

M. L. Williams

Member, Thermal Reactor Data Testing Subcommittee, Cross Section Evaluation Working Group (CSEWG)
 Lecturer, Tennessee Industries Week, University of Tennessee, "Computational Methods in Nuclear Reactor Analysis"
 Referee, *Nuclear Science and Engineering*
 Referee, Proposal Reviews, National Science Foundation

B. A. Worley

Member, ANS Program Committee, Reactor Physics Division
 Coordinator, EFD United Way Campaign, 1980, 1981, 1982

ENGINEERING PHYSICS DIVISION SEMINARS AT ORNL

Current seminar coordinators are R. R. Spencer and B. A. Worley. G. de Saussure was one of the seminar coordinators during the first half of the period covered by this report. The following seminars were held during 1981-1982.

- F. S. Alsmiller, Engineering Physics Division, "A Phenomenological Model for Prediction of the End Products of Medium-Energy (≤ 3 -GeV) Fission and Comparison with Experiment"
- R. G. Alsmiller, Jr., Engineering Physics Division, "Neutron Production by Medium-Energy Protons in Thick Uranium Targets and Comparisons with Experimental Data"
- R. G. Alsmiller, Jr., Engineering Physics Division, "Adjoint Sensitivity Theory and Its Application to the Long-Term Energy Analysis Program (LEAP) Model 22C" (Joint Seminar with Energy Division)
- S. M. Austin, Michigan State University, "Creation of Light Elements and Whether the Universe Will Expand Forever" (Joint Seminar with Physics Division)
- G. Bartholomew, B. M. Townes, and S. O. Schriber, Atomic Energy of Canada, Ltd., Chalk River Nuclear Laboratory, "Canadian Accelerator Breeder Program"
- H. Beer, Federal Republic of Germany, "Branching in S-Process Nucleosynthesis and Their Astrophysical Applications"
- C. D. Bowman, National Bureau of Standards, "Possibilities for Accelerator-Based Neutron Research Using the Pigmi Technology"
- J. K. Dickens, Engineering Physics Division, "Decay Heat: Fission Product Yields and Reactor Antineutrino Spectra, Pearls Among the Pearls"
- M. Divadeenam, Brookhaven National Laboratory, "Doorway State Interpretation of Neutron and Proton Resonances"
- G. F. Flanagan, Engineering Physics Division, "Results of the DOE Industrial Fact-Finding Group (R&D LWR Safety Needs)"
- D. V. Gopinath, Safety Research Laboratory, Kalpakkam, India, "An Analytical Solution for the Practical Radiation Transport Problems Using Finite Transforms"
- D. V. Gopinath, Safety Research Laboratory, Kalpakkam, India, "Secondary Sources in Gamma-Ray Transport"
- M. C. G. Hull, Engineering Physics Division, "Sensitivity Analysis of the Impact of CO₂ Accumulation on the Climate"

C. D. Heising-Goodman, Massachusetts Institute of Technology, "A Comparative Assessment of Events of Advanced Reactor Fuel Cycles -- Preventing Things That Go Bang in the Night" (Joint Seminar with Health and Safety Research Division)

E. A. Knapp, Los Alamos National Scientific Laboratory, "Linacs and Storage Rings at Los Alamos"

George Lawrence, Accelerator Technology Division, Los Alamos National Laboratory, "Design of the Los Alamos Proton Storage Ring and Possible Options"

George Lawrence, Los Alamos National Scientific Laboratory, "Design of the Los Alamos Proton Storage Ring and Possible ORNL Options"

F. C. Maienschein, Engineering Physics Division, "ORNL Computer Steering Committee Plans and Possible Impacts on EPD"

B. F. Maskewitz, Engineering Physics Division, "General Session"

D. L. Moses, Engineering Physics Division, "A Proposed Method for Inverse Reactivity Measurements in Large Nuclear Power Reactors"

S. Mosko and J. Benjamin, Physics Division, "Coping with SF₆ Hazards"

C. Overbey, National Highway Traffic Safety Administration, "Human Factors and Heavy Truck Safety - Ride Quality Research"

F. G. Perey, Engineering Physics Division, "Logical Inference and Bayesian Statistics" (Joint Seminar with Physics Division)

F. G. Perey, Engineering Physics Division, "Sessions of Technical Presentations"

R. B. Perez, Engineering Physics Division, "Cardiograma: A Stochastic, Semiempirical Methodology for Power Reactor Surveillance and Diagnostics"

J. Persensky, National Bureau of Standards - Nuclear Regulatory Commission, "Methods in Human Factors Research"

M. Reiser, University of Maryland, "Advanced High Current Accelerator Concepts"

Y. Ronen, Beer-Sheva, Israel, "A Probabilistic Rejection Test for Multivariable Sensitivity Analysis"

J. Siegel, BDM Corporation, Oak Ridge, "Analysis of Time Dependence of Radiation-Induced Conductivity and Dielectrics"

M. Slabbekorn, Information Division, Graphic Arts, "Publication Quality Computer Graphics"

P. Slovic, Decision Research, Division of Perceptronics, "Risk Assessment: Behavioral Research and Social Implications"

- M. Steinberg, P. Grand, J. Powell, and H. Takahashi, Brookhaven National Laboratory, "APEX Nuclear Fuel Cycle and Spallator for Production of LWR Fuel and Elimination of Radioactive Waste"
- D. A. Swenson, Los Alamos National Laboratory, "The Pigma Technology: A New Breed of Proton Linac"
- V. R. R. Uppuluri, Computer Sciences Division, "Mathematical and Statistical Problems in Risk Analysis"
- D. R. Vondy, Engineering Physics Division, "On Validating Major Computer Codes"
- G. Young, Physics Division, "Overview of the Proposed Neutron-Antineutron Oscillation Experiment"

PUBLICATIONS AND PAPERS PRESENTED AT SCIENTIFIC MEETINGS

(See also Vol. II Bibliography for additional publications for L. E. Bartine, B. L. Broadhead, T. J. Burns, S. N. Cramer, M. B. Emmett, W. W. Engle, Jr., G. F. Flanagan, D. T. Ingersoll, J. O. Johnson, H. E. Klee, E. M. Oblow, J. V. Pace, III, W. A. Rhoades, D. L. Selby, J. R. White, and L. R. Williams)

PUBLICATIONS*

ABBOTT, L. S., D. T. INGERSOLL, F. J. MUCKENTHALER, AND C. O. SLATER

"Review of ORNL-TSF Shielding Experiments for the Gas-Cooled Fast Breeder Reactor Program," ORNL-5805 (January, 1982). (5.32)

ALSMILLER, F. S., R. G. ALSMILLER, JR., T. A. GABRIEL, R. A. LILLIE, AND J. BARISH**

"A Phenomenological Model for Particle Production from the Collision of Nucleons and Pions with Fissile Elements at Medium Energies," ORNL/TM-7528 (March, 1981); *Nucl. Sci. Eng.* 79, 147-161 (1981). (5.24)

ALSMILLER, R. G., JR., AND J. BARISH**

"Neutron-Photon Multigroup Cross Sections for Neutron Energies ≤ 400 MeV," ORNL/TM-7818 (June, 1981); *Nucl. Sci. Eng.* 80, 448-452 (1982). (5.27)

ALSMILLER, R. G., JR., J. BARISH,** J. M. BARNES,** AND R. T. SANTORO

"Neutron Production by Protons (190-268 MeV) in a Water-Cooled Tantalum Target," ORNL/TM-7949 (September, 1981); *Nucl. Sci. Eng.* 80, 452-455 (1982). (5.26)

ALSMILLER, R. G., JR., J. BARISH**, AND R. L. CHILDS**

"Skyshine at Neutron Energies < 400 MeV," *Part. Accel.* 11(3), 131-141 (1981). (1980)

ALSMILLER, R. G., JR., J. BARISH**, J. DRISCHLER, J. HORWEDEL**, J. L. LUCIUS,** AND J. McADOO**

"Adjoint Sensitivity Theory and Its Application to LEAP Model 22C," ORNL/TM-7789 (June, 1981). (6.3)

ALSMILLER, R. G., JR., T. A. GABRIEL, J. BARISH**, AND F. S. ALSMILLER

"Neutron Production by Medium-Energy (< 1.5 GeV) Protons in Thick Uranium Targets," ORNL/TM-7527 (January, 1981); *Nucl. Sci. Eng.* 79, 162-166 (1981). (5.25)

*Numbers shown in parentheses following the publication description correspond to an abstract in this report. In most cases, abstracts not included here were published in a previous progress report, and the year is indicated.

**Not a member of the Engineering Physics Division.

ALSMILLER, R. G., JR., T. A. GABRIEL, AND B. L. BISHOP**

"Calculated Differential π^+ Production Spectra at Large Angles from 28 GeV Protons on a Thick Iridium Target and Comparisons with Experimental Data RNL/TM-7647 (January, 1981); *Nucl. Instr. and Meth.* 185, 253-255 (1981). (5.22)

AUCHAMPAUGH, G. F.,** S. PLATTARD,** N. W. HILL,** G. de SAUSSURE, R. B. PEREZ, AND J. A. HARVEY**

"K-Components for the ~ 1.4 -, ~ 1.6 -, and ~ 1.7 -MeV Structures in the Fission of $^{232}\text{Th} + n$," Los Alamos Report LA-8671 (July, 1981). (1.6)

BARHEN, J., D. G. CACUCI, J. J. WAGSCHAL,** M. A. BJERKE,** AND C. B. MUL-
LINS**

"Uncertainty Analysis of Time-Dependent Nonlinear Systems: Theory and Application to Transient Thermal-Hydraulics" *Nucl. Sci. Eng.* 81, 23 (1982). (4.10)

BARHEN, J., AND W. ROTHENSTEIN**

"OZMA - A Code to Calculate Resonance Reaction Rates in Reactor Lattice: Using Resonance Profile Tabulations," EPRI Project 709-2, Report NP-926 (February, 1981). (1980)

BARISH, J.,** T. A. GABRIEL, F. S. ALSMILLER, AND R. G. ALSMILLER, JR.

"HETFIS - High-Energy Nucleon-Meson Transport Code with Fission," ORNL/TM-7882 (July, 1981). (3.7)

BARKS, D. B.,** E. J. KOZINSKY,** AND S. ECKEL**

"Nuclear Power Plant Control Room Task Analysis: Pilot Study for Pressurized Water Reactors," NUREG/CR-2598 (ORNL/Sub/79-40432/1) (May, 1982). (7.4)

BARNES, J. M.,** R. T. SANTORO, R. A. LILLIE, AND R. G. ALSMILLER, JR.

"Isa-Response Contours in the ETF Neutral Beam Injectors," ORNL/TM-7783 (May, 1981). (5.10)

BELL, Z. W.

"ISOLR - An Isometric Plotting Package for Use with ORPLOT, ORNL/TM-8192 (May 1982). (1.22)

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